United States Nuclear Regulatory Commission

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NUREG-2193
Supplement 1

Safety Evaluation Report

Related to the License Renewal of Davis-Besse Nuclear Power Station

Docket Number 50-346

FirstEnergy Nuclear Operating Company

Supplement 1

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Manuscript Completed: August 2015
Date Published: April 2016

Office of Nuclear Reactor Regulation

ABSTRACT

This document is a supplemental safety evaluation report (SSER) for the license renewal application (LRA) for Davis-Besse Nuclear Power Station (Davis-Besse) as submitted by FirstEnergy Nuclear Operating Company (FENOC or the applicant). By letter dated August 27, 2010, FENOC submitted its LRA to the U.S. Nuclear Regulatory Commission (NRC) for renewal of the Davis-Besse operating licenses for an additional 20 years. The NRC staff (the staff) issued a safety evaluation report (SER) related to the license renewal of Davis-Besse Nuclear Power Station, dated September 3, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13248A267), which summarizes the results of its review of the LRA for compliance with the requirements of Title 10, Part 54, of the *Code of Federal Regulations* (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

This SSER documents the staff's review of supplemental information provided by the applicant since the issuance of the SER. This information includes annual updates required by 10 CFR 54.21(b) and updated information and commitments in response to the recent industry operating experience. This SSER supplements portions of SER Sections 1, 2, 3, 4, and Appendices.

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ABBREVIATIONS AND ACRONYMS

ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
AERM	aging effect requiring management
AFW	auxiliary feedwater
A/LAI	applicant/licensee action items
ALARA	as low as is 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&W	Babcock & Wilcox
BWST	borated water storage tank
CASS	cast austenitic stainless steel
CFR	*Code of Federal Regulations*
CLB	current licensing basis
CRGT	control rod guide tube
CSE	copper/copper sulfate reference electrode
CSS	core support shield
CUF	cumulative usage factor
CUI	corrosion under insulation
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
EQ	Equipment Qualification

FACE	Full Apparent Clause Evaluation (Report)
FD	flow distributor
FENOC	FirstEnergy Nuclear Operating Company
FIV	flow-induced vibration
FMECA	failure modes, effects and criticality analysis/analyses
GALL	Generic Aging Lessons Learned
GL	generic letter
HDPE	high-density polyethylene
HPCI	high-pressure coolant injection
HPI	high-pressure injection
HTH	high-temperature annealed and aged condition heat treatment
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
I&E	inspection and evaluation
IASCC	irradiation-assisted stress corrosion cracking
IGA	intergranular attack
IMI	incore monitoring instrumentation
IPA	integrated plant assessment
ISI	inservice inspection
LAI	licensee action items
LCB	lower core barrel
LOCA	loss-of-coolant accident
LRA	license renewal application
LR-ISG	license renewal interim staff guidance
LTS	lower thermal shield

MFW	main feedwater
MIC	microbiologically-influenced corrosion
MRP	Materials Reliability Program
MUR	measurement uncertainty recapture
mV	millivolt(s)
NACE	National Association of Corrosion Engineers
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
ohm-cm	ohm-centimeter(s)
OTSG	once-through steam generator
PIV	post-indicating valves
PM	preventive maintenance
ppm	part(s) per million
psid	pound(s) per square inch differential
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RFO	refueling outage
RG	regulatory guide
RVI	reactor vessel internals
RVIIP	reactor vessel internals inspection plan

SBO	station blackout
SBODG	station blackout diesel generator
SCC	stress corrosion cracking
SCV	steel containment vessel
SEI	Structural Engineering Institute
SER	safety evaluation report
SRP-LR	“Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Revision 2”
SSC	systems, structures, and components
SSER	supplemental safety evaluation report
SSHT	surveillance specimen holder tube
TR	Topical Report
TLAA	time-limited aging analysis
TS	technical specification(s)
UCB	upper core barrel
USAR	updated safety analysis report
UT	ultrasonic testing
UTS	upper thermal shield
VT-3	visual examination

SECTION 1

INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This document is a supplemental safety evaluation report (SSER) for the license renewal application (LRA) for Davis-Besse Nuclear Power Station (Davis-Besse), as submitted by FirstEnergy Nuclear Operating Company (FENOC or the applicant). By letter dated August 27, 2010, FENOC submitted its LRA to the U.S. Nuclear Regulatory Commission (NRC) for renewal of the Davis-Besse operating licenses for an additional 20 years. The NRC staff (the staff) issued a safety evaluation report (SER) related to the license renewal of Davis-Besse dated September 3, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13248A267), which summarizes the results of its review of the LRA for compliance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

This SSER documents the staff's review of additional information provided by the applicant since the staff's issuance of the SER in September 2013. This information includes annual updates required by 10 CFR 54.21(b) and updated information and commitments in response to the recent industry operating experience. This SSER supplements portions of SER Sections 1, 2, 3, 4, and Appendices.

The following sections, unless otherwise noted, have been updated and supersede the corresponding sections of the SER.

1.4 Interim Staff Guidance

License renewal is a living program. The staff, industry, and other interested stakeholders gain experience and develop lessons learned with each renewed license. The lessons learned address the staff's performance goals of maintaining safety, improving effectiveness and efficiency, reducing regulatory burden, and increasing public confidence. License renewal interim staff guidance (LR-ISG) is documented for use by the staff, industry, and other interested stakeholders until incorporated into such license renewal guidance documents as NUREG–1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Revision 2, December 2010 (SRP-LR), and NUREG–1801, "Generic Aging Lessons Learned Report," Revision 2, December 2010 (GALL Report).

The applicant revised several aging management programs (AMPs) and aging management review (AMR) items in several Table 2s in order to address recently issued LR-ISGs. The staff's evaluation of associated changes to AMPs and AMR items in the Table 2s are addressed in the corresponding sections of this SSER. For example, revised and new enhancements to the Buried Piping and Tanks Inspection Program are addressed in SSER Section 3.0.3.2.3. Changes to the AMR items in the Table 2s are addressed in SSER Sections (e.g., 3.2, 3.3, 3.4).

Table 1.4-1 Current License Renewal Interim Staff Guidance

ISG Issue (Approved ISG Number)	Purpose	SSER Section
"Changes to the Generic Aging Lessons Learned (GALL) Report Revision 2, Aging Management Program (AMP) XI.M41, Buried and Underground Piping and Tanks" (LR-ISG-2011-03)	See Section 1.4.1	See SSER Section 3.0.3.2.3
"Wall Thinning Due to Erosion Mechanisms" (LR-ISG-2012-01)	See Section 1.4.2	See SSER Section 3.0.3.5
"Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion under Insulation" (LR-ISG-2012-02)	See Section 1.4.3	See SSER Sections 3.0.3.4.1, 3.0.3.3.7, 3.0.3.2.8, 3.0.3.2.1, and 3.0.3.2.5

1.4.1 LR-ISG-2011-03, "Changes to the Generic Aging Lessons Learned (GALL) Report Revision 2 Aging Management Program (AMP) XI.M41, 'Buried and Underground Piping and Tanks'"

The staff issued LR-ISG-2011-03 on August 2, 2012, in order to address changes to GALL Report AMP XI.M41, "Buried and Underground Piping and Tanks," associated with inspection recommendations for plants not utilizing a cathodic protection system, increases in inspection sample size when adverse conditions are detected, acceptance criteria for cathodic protection surveys, and other miscellaneous topics. The staff's evaluation of the applicant's changes to its Buried Piping and Tanks Inspection Program to address this ISG is documented in SSER Section 3.0.3.2.3.

1.4.2 LR-ISG-2012-01, "Wall Thinning Due to Erosion Mechanisms"

The staff issued LR-ISG-2012-01, "Wall Thinning Due to Erosion Mechanisms," on May 1, 2013, in order to address loss of material due to various erosion mechanisms. This LR-ISG generated new AMR items and modified GALL Report AMP XI.M17, "Flow-Accelerated Corrosion." By letter dated June 23, 2014, the applicant provided its evaluation of LR-ISG-2012-01 and concluded that an additional monitoring program was not needed to manage erosion, flashing, or cavitation, based on reviews of plant operating experience. The staff's evaluation of the applicant's review for this LR-ISG is documented in SSER Section 3.0.3.5.

1.4.3 LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation"

The staff issued LR-ISG-2012-02, on November 22, 2013, in order to address aging effects associated with recurring internal corrosion, fire water systems, atmospheric storage tanks, corrosion under insulation, and other changes to the GALL Report and SRP-LR. The changes addressed in LR-ISG-2012-02 are discussed in the following sections.

1.4.3.1 Recurring Internal Corrosion

Part A of LR-ISG-2012-02 addresses loss of material due to recurring internal corrosion. By letter dated February 19, 2014, the applicant provided its evaluation of LR-ISG-2012-02, Part A, and provided a new LRA Section 3.3.2.2.16, "Loss of Material Due to Recurring Internal Corrosion." The staff's evaluation of the applicant's review for loss of material due to recurring internal corrosion is documented in SSER Section 3.0.3.4.1.

1.4.3.2 Representative Minimum Sample Sizes for Periodic Inspections

Part B of LR-ISG-2012-02 addresses the representative minimum sample size for periodic inspections in GALL Report AMP XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components." By letter dated February 19, 2014, the applicant provided its evaluation of LR-ISG-2012-02, Part B, and made changes to LRA Sections A.1.41 and B.2.41, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program." The staff's evaluation of changes to the applicant's Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program is documented in SSER Section 3.0.3.3.7.

1.4.3.3 Flow Blockage of Water-Based Fire Protection System Piping

Part C of LR-ISG-2012-02 addresses flow blockage of water-based fire protection system piping in GALL Report AMP XI.M27, "Fire Water System." By letter dated February 19, 2014, the applicant provided its evaluation of LR-ISG-2012-02, Part C, and made changes to LRA Sections A.1.18 and B.2.18, "Fire Water Program." The applicant also revised LRA Table 3.3.2-14 to address flow blockage and managing aging effects associated with the fire water storage tank. The staff's evaluation of changes to the applicant's Fire Water Program is documented in SSER Section 3.0.3.2.8.

1.4.3.4 Revisions to the Scope of Inspection Recommendations of GALL Report AMP XI.M29, "Aboveground Metallic Tanks"

Part D of LR-ISG-2012-02 addresses revisions to the recommended inspection scope of GALL Report AMP XI.M29, "Aboveground Metallic Tanks." By letter dated February 19, 2014, the applicant provided its evaluation of LR-ISG-2012-02, Part D, and replaced LRA Sections A.1.2 and B.2.2, "Aboveground Steel Tanks Inspection Program," in their entirety. The staff's evaluation of the applicant's changes to the Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1.

1.4.3.5 Corrosion Under Insulation

Part E of LR-ISG-2012-02 addresses corrosion under insulation. By letter dated February 19, 2014, the applicant provided its evaluation of LR-ISG-2012-02, Part E, and revised LRA Sections A.1.15 and B.2.15, "External Surfaces Monitoring Program." The staff's evaluation of changes to the applicant's External Surfaces Monitoring Program is documented in SSER Section 3.0.3.2.5.

1.5 Summary of Open Items

The staff does not have any changes or updates to this section of the SER.

1.7 Summary of Proposed License Conditions

Following the staff's review of the LRA, including subsequent information and clarifications provided by the applicant, the staff identified three proposed license conditions.

The first license condition requires the information in the updated safety analysis report (USAR) supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the LRA review process and supplemented by Appendix A of the "Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station," to be part of the USAR, which will be updated in accordance with 10 CFR 50.71(e), following the issuance of the renewed license. As such, the applicant may make changes to the programs and activities described in the USAR supplement, provided the applicant evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

The second license condition states that the applicant's USAR supplement submitted pursuant to 10 CFR 54.21(d), as supplemented by Appendix A of the "Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station," describes certain future programs and activities to be completed before the period of extended operation.

(a) The applicant shall implement those new programs and enhancements to existing programs no later than October 22, 2016 (i.e., no later than 6 months prior to the period of extended operation).

(b) The applicant shall complete those activities as noted in Commitment Nos. 22, 23, 24, 38, 41, 54, and 55 no later than October 22, 2016.

The applicant shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

LRA Appendix A, "Updated Safety Analysis Report Supplement," Table A-1, "Davis-Besse License Renewal Commitments," contains commitments for license renewal and an associated schedule for when the applicant plans to implement or complete the commitments. The staff noted that, through the commitments in LRA Appendix A, Table A-1, the applicant will implement new programs, will implement enhancements to existing programs, and will also complete inspection or testing activities. The staff also noted that the Davis-Besse current license expires on April 22, 2017. Therefore, the applicant's implementation schedule for some commitments, as provided originally in LRA Section Appendix A, Table A-1, may conflict with the implementation schedule intended by the generic second license condition described above. By letter dated March 26, 2013, the staff issued RAI A.1-1, Part (1), requesting that the applicant identify those commitments to implement new programs and enhancements to existing programs and state when the implementation of these programs will be completed. In addition, RAI A.1-1, Part (2), requested that the applicant identify those commitments to complete inspection or testing activities and state when the completion of these inspection and testing activities will occur. The applicant responded to RAI A.1-1 in a letter dated March 26, 2013.

In response to RAI A.1-1, Part (1), the applicant identified Commitment Nos. 1 through 14, 16 through 21, 25, 27 through 32, 34, 40, 45 through 47, and 49, as those commitments associated with implementation of new programs and enhancements to existing programs.

The applicant stated that these commitments will be completed no later than October 22, 2016. As part of its response, the applicant also provided LRA Amendment 40, which revised the implementation schedule in LRA Appendix A, Table A-1, for these commitments, to state that they will be completed no later than October 22, 2016. In response to RAI A.1-1, Part (2), the applicant identified Commitment Nos. 22 through 24, 38, 41, 43, 44, and 48, as those commitments associated with inspection and testing activities. The applicant stated that these commitments will be completed no later than October 22, 2016. The applicant also revised the implementation schedule in LRA Appendix A, Table A-1, to state that these commitments will be completed no later than October 22, 2016.

The staff finds the applicant response to RAI A.1-1, Part (1), acceptable because the applicant identified those commitments that implement new programs and enhancements to existing programs and revised the implementation schedule on LRA Appendix A, Table A-1, to complete these commitments 6 months before the period of extended operation, which is consistent with the proposed second license condition. The staff finds the applicant response to RAI A.1-1, Part (2), acceptable because the applicant identified those commitments to complete inspection or testing activities and revised the implementation schedule on LRA Appendix A, Table A-1, consistent with the proposed second license condition, to state that these commitments will be implemented 6 months before the period of extended operation. Therefore, the staff's concerns described in RAI A.1-1, Parts (1) and (2), are resolved.

The third license condition requires testing of surveillance capsules for the period of extended operation to meet the test procedures and reporting requirements of American Society of Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. All pulled capsules shall be properly maintained for testing, and any changes to storage requirements must be approved by the NRC. All pulled and tested capsules, unless discarded before August 31, 2000, shall be placed in storage to be saved for possible future reconstitution and use.

SECTION 2

STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

2.2 Plant-Level Scoping Results

2.2.3 Staff Evaluation

Summary of Technical Information in the Application. By letter dated September 20, 2013, the applicant supplemented the LRA as required by 10 CFR 54.21(b). In this supplement, FENOC provided information related to a new steam generator storage facility that was not previously included as part of the August 27, 2010, LRA submittal. In the original LRA, the applicant provided the results of applying the license renewal scoping criteria for structures in Table 2.2-3. If a system or structure, in whole or in part, met one or more of the license renewal scoping criteria, the system or structure was evaluated as within the scope of license renewal for Davis-Besse. The applicant stated in the September 20, 2013 LRA supplement letter that the new steam generator storage facility is an 18,000-square-foot structure that will be used to house and protect the replacement steam generators until they are installed in the plant. This structure is located outside the protected area and, therefore, is not within the scope of license renewal.

The applicant also stated that LRA Table 2.2-3, "License Renewal Scoping Results for Structures," had been revised to address the new structure.

In addition, FENOC included information related to new foundations added to the Davis-Besse switchyard to support new electrical breakers. Specifically, the applicant stated in the supplement letter that the Davis-Besse switchyard had been reconfigured in order to facilitate breaker maintenance, by extending the J (East) and K (West) buses and adding two new 345 kilovolts (kV) breakers. The two 345 kV breakers are connected on either side of the new location of the Ohio Edison line, with one breaker between the Ohio Edison line and the J Bus and one breaker between the Ohio Edison line and the K Bus. The applicant also stated that the new switchyard configuration will enable the buses to be maintained under maintenance or outage conditions, resolving concerns regarding power interruption and short outage duration time limits imposed during line or breaker maintenance activities.

Staff Evaluation. The staff has completed the evaluation related to the new steam generator storage facility, as described in the supplemental information provided by the applicant on September 20, 2013. The staff compared the original review, as detailed in SER Section 2 from September 3, 2013, to the proposed changes detailed by the applicant in the supplement. The staff performed the review in accordance with the regulations and requirements of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and the guidance contained in NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR), Section 2.1, "Scoping and Screening Methodology," and SRP-LR Section 2.2, "Plant-Level Scoping Results."

The staff reviewed the information related to the new steam generator storage facility, since the applicant did not identify it as being within the scope of license renewal to verify whether the systems and structures have any intended functions requiring their inclusion within the scope of license renewal.

Based on its review, the staff determined that the new steam generator storage facility is not within the scope of license renewal because the structure does not meet any of the scoping criteria of 10 CFR 54.4 as described below:

(1) safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events

(2) all nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of 10 CFR 54.4

(3) all systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63)

2.2.4 Conclusion

LRA Table 2.2-3, "License Renewal Scoping Results for Structures" has been revised to address the new structure. There are no changes or updates to this section of the SER.

2.4 Scoping and Screening Results: Structures

2.4.12 Yard Structures

2.4.12.1 Summary of Technical Information in the Application

The staff noted that, by letter dated June 23, 2014, the applicant provided additional information related to an additional yard structure after the issuance of the SER. The additional structure below is added to the 2.4.12.1 list in the SER.

- SBO Components foundations and Structures in the Yard and Switchyard including Startup Transformers 01 and 02; Bus-Tie Transformers; 345-kV Switchyard circuit breakers ACB34560, ACB34561, ACB34562, ACB34563, ACB34564, 81-B-65, 81-B-66 and 81-B-67; Relay House, Switchyard and Yard Towers for 345-kV distribution; "J" and "K" buses – Seismic Class II.

2.4.12.2 Staff Evaluation

There are no changes or updates to this section of the SER.

2.4.12.3 Conclusion

There are no changes or updates to this section of the SER.

2.5 Scoping and Screening Results: Electrical and Instrumentation and Controls

This section documents the staff's review of the applicant's scoping and screening results for electrical and instrumentation and control (I&C) systems. Specifically, this section discusses electrical and I&C component commodity groups.

In accordance with the requirements of 10 CFR 54.21(a)(1), the applicant must list passive, long-lived systems, structures, and components (SSCs) within the scope of license renewal and subject to an AMR. To verify that the applicant properly implemented its methodology, the staff's review focused on the implementation results. This focus allowed the staff to confirm that there were no omissions of electrical and I&C system components that meet the scoping criteria and are subject to an AMR.

The staff's evaluation of the information in the LRA was the same for all electrical and I&C systems. The objective was to determine whether the applicant has identified, in accordance with 10 CFR 54.4, components and supporting structures for electrical and I&C systems that appear to meet the license renewal scoping criteria. Similarly, the staff evaluated the applicant's screening results to verify that all passive, long-lived components were subject to an AMR in accordance with 10 CFR 54.21(a)(1).

In its scoping evaluation, the staff reviewed the applicable LRA sections and the request for additional information (RAI) responses, focusing on components that have not been identified as within the scope of license renewal. The staff reviewed the Davis-Besse updated safety analysis report (USAR) for each electrical and I&C system to determine whether the application has omitted, from the scope of license renewal, components with intended functions delineated under 10 CFR 54.4(a).

After its review of the scoping results, the staff evaluated the applicant's screening results. For those SSCs with intended functions, the staff sought to determine whether (1) the functions are performed with moving parts or a change in configuration or properties or (2) the SSCs are subject to replacement after a qualified life or specified time period, as described in 10 CFR 54.21(a)(1). For those SSCs meeting neither of these criteria, the staff sought to confirm that these SSCs were subject to an AMR, as required by 10 CFR 54.21(a)(1).

2.5.1 Electrical and Instrumentation and Controls Commodity Groups

2.5.1.1 Summary of Technical Information in the Application

LRA Section 2.5 describes the electrical and I&C systems. The integrated plant assessment (IPA) approach for the review of the electrical and I&C components that are in scope of license renewal eliminates the need to uniquely identify each individual component and its specific location and precludes improper exclusion of components from an AMR.

The applicant's IPA scoping process includes all plant electrical and I&C systems within the scope of license renewal unless they are specifically scoped out. The IPA screening process groups all in-scope electrical and I&C components in commodity groups and identifies those electrical commodity groups that are subject to an AMR by applying 10 CFR 54.21(a)(1)(i) and 10 CFR 54.21(a)(1)(ii). The applicant determined the in-scope electrical components required for compliance with the Station Blackout (SBO) Rule (10 CFR 50.63), as well as their corresponding intended functions, through a review of the Davis-Besse current licensing basis (CLB), with consideration of the requirements of 10 CFR 54.4(a)(3) and the guidance provided in the SRP-LR.

LRA Table 2.5-1, "Electrical and I&C System Components Subject to Aging Management Review," identifies the following electrical and I&C component and commodity groups subject to AMR along with their license renewal intended functions:

- non-environmentally qualified (EQ) insulated cables and connections includes non-EQ electrical penetration assemblies, non-EQ cable connections (metallic parts) – conduct electricity
- non-EQ sensitive, high-voltage, low-level signal instrument cables and connections – conduct electricity
- non-EQ medium-voltage power cables – conduct electricity
- switchyard bus and connections – conduct electricity
- transmission conductors and connections – conduct electricity
- high-voltage insulators – insulation and support

Electrical and I&C components, such as thermocouples, radiation detectors, flow elements, and electrical heaters having pressure boundary intended functions, are assessed in LRA Section 2.3, "Scoping and Screening Results: Mechanical."

Components that support or interface with the electrical and I&C components are assessed in LRA Section 2.4, "Scoping and Screening Results: Structural."

2.5.1.2 Staff Evaluation

The staff reviewed LRA Section 2.5, the annual updates to LRA Section 2.5 in LRA Amendments 46 and 50, and Davis-Besse USAR Chapters 7 and 8, using the evaluation methodology described in the guidance in SRP-LR Section 2.5, "Scoping and Screening Results: Electrical and Instrumentation and Controls Systems."

During its review, the staff evaluated the system functions described in the LRA and USAR to verify that the applicant has not omitted, from the scope of license renewal, any components with intended functions delineated under 10 CFR 54.4(a). The staff then reviewed those components that the applicant identified as within the scope of license renewal to verify that the applicant has not omitted any passive and long-lived components subject to an AMR, in accordance with the requirements of 10 CFR 54.21(a)(1).

In LRA Section 2.5.3, "Elimination of Component Commodity Groups With No License Renewal Intended Functions," the applicant provided the electrical and I&C component that do not perform a license renewal function and are excluded from AMR.

The applicant excluded uninsulated ground conductors from AMR because uninsulated ground conductors do not perform a license renewal intended function at Davis-Besse. The applicant clarified that uninsulated ground conductors limit equipment damage and provide personnel protection; their failure cannot cause the loss of a safety function; and they are not relied upon in safety analyses or plant evaluations to perform any function consistent with the requirements of 10 CFR 54.4(a)(3). The staff reviewed the USAR and found that uninsulated ground conductors are not credited in the Davis-Besse design basis. Therefore, the staff concludes that the exclusion of uninsulated ground conductors from AMR is acceptable.

The applicant excluded fuse holders from AMR because Davis-Besse fuse holders are either part of active electrical panels or are located in circuits that perform no license renewal intended function. The applicant made this determination based on a review of Davis-Besse electrical drawings, the fuse documentation, and other engineering documents. Based on this information, the staff finds that the exclusion of fuse holders from AMR is consistent with NUREG-1801, Revision 1, “Generic Aging Lessons Learned Report” (GALL Report) and is, therefore, acceptable.

The applicant excluded cable tie-wraps from AMR because cable tie-wraps are not within the scope of license renewal at Davis-Besse. The applicant clarified that cables tie-wraps are used to bundle wires and cables together to keep the wire and cable runs neat and to restrain cables and wires within raceway to facilitate cable installation at Davis-Besse. The applicant stated that cable tie-wraps have no current license basis requirements at Davis-Besse; they are not required for maintaining cable ampacity, minimum bending radius, or cables within vertical raceways; and they are not required for seismic analysis. Based on this information and the review of the Davis-Besse USAR, the staff finds that the exclusion of cable-tie wraps from AMR is acceptable.

In LRA Section 2.5.6.2, “Station Blackout Recovery Path Evaluation Boundaries,” the applicant identified the SBO license renewal offsite power recovery path boundaries and the in-scope components for SBO. The Davis-Besse in-scope SBO recovery path boundary drawing is provided in LRA Figure 2.5-1.

Regulations in 10 CFR 54.4(a)(3) require that all systems, structures, and components (SSCs) relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for SBO (10 CFR 50.63) be included within the scope of license renewal. SRP-LR Section 2.5.2.1.1 provides the guidance to identify electrical and I&C systems components that are relied upon to meet the requirements of the SBO Rule for license renewal. This includes equipment that is required to cope with an SBO (e.g., alternate ac power sources) meeting the requirements in 10 CFR 54.4(a)(3) and the plant system portion of the offsite power system, including switchyard 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 system, and the associated control circuits and structures, that is used to connect the plant to the offsite power source. In addition, General Design Criteria 17 of 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” requires that electric power from the transmission network to the onsite electric distribution system is supplied by two physically independent circuits to minimize the likelihood of their simultaneous failure. SSCs that are relied upon to meet the requirements of the SBO Rule in both circuits are to be included within the scope of license renewal.

By letters dated September 20, 2013 (LRA Amendment 46), and June 23, 2014 (LRA Amendment 50), the applicant provided annual updates of the LRA that include revisions to Section 2.5. The revisions summarized changes to the Davis Besse switchyard that affect the in-scope components in the SBO power recovery path.

In LRA Amendment 46, the applicant stated that the Davis-Besse switchyard was reconfigured to facilitate breaker maintenance by extending the J (East) and K (West) buses, adding two new 345 kV breakers, and removing existing disconnect ABS34625. Specifically, the two new 345 kV breakers are connected on either side of the new location of the Ohio Edison line, with one breaker between the Ohio Edison line and the J Bus and one breaker between the Ohio Edison line and the K Bus. The applicant also stated that the new switchyard configuration will enable the buses to be maintained under maintenance or outage conditions, resolving concerns regarding power interruption and short outage duration time limits imposed during line or breaker maintenance activities. This configuration change provides greater operational flexibility, power transfer capability, and less operational risk. As a result of the switchyard reconfiguration, the applicant revised the last two paragraphs of LRA Section 2.5.6.2, LRA Figure 2.5-1, and LRA Table A-1, "Davis-Besse License Renewal Commitments."

In LRA Amendment 50, the applicant stated that the Davis-Besse switchyard was reconfigured to install a fourth 345-kV transmission line, the "Hayes Line," to relocate the Ohio Edison line and rename it as the "Beaver Tower C" line, and to add a new 345 kV circuit breaker (81-B-65). The Beaver Tower C line is now connected between the new circuit breaker 81-B-65 and existing circuit breaker ACB34564. The applicant also stated that new foundation was added to the Davis-Besse switchyard to support the new breaker and associated structures. As a result, the applicant provided a second revision to the second to last paragraph of LRA Section 2.5.6.2 and LRA Figure 2.5-1.

During the review of LRA Amendment 50, the staff noted that Davis-Besse relies on an SBO diesel generator (SBODG) system to satisfy the 10 CFR 54.4(a)(3) scoping criteria for the SBO regulated event. The applicant evaluated the in-scope mechanical components of the SBODG system in LRA Section 2.3 and the in-scope structural items (SBODG building) in LRA Section 2.4. However, the applicant neither evaluated the in-scope electrical components of the SBODG in LRA Section 2.5 nor included the SBODG in LRA Figure 2.5-1. By letter dated August 19, 2014, the staff issued RAI 2.5.6.2a, requesting the applicant to provide the scoping and screening results of SBODG system electrical components and revise LRA Figure 2.5-1.

In its response letter dated September 16, 2014, the applicant provided LRA Amendment 53 including an update to LRA Figure 2.5-1 and the first two paragraphs of LRA Section 2.5.6.2. The applicant stated that the emergency diesel generators (EDGs) systems are safety-related systems that perform license renewal intended functions meeting the requirements under 10 CFR 54.4(a)(1); therefore, LRA Section 2.5.6.2 is revised to state that the cable and connectors from the EDGs and SBODG systems provide connection to the onsite power system meeting the requirements of 10 CFR 54.4. The applicant included the cable and connectors from the SBODG and EDGs systems in the commodity groups subject to an AMR and added the SBODG and the EDGs to the SBO offsite power recovery path in LRA Figure 2.5-1.

No change was required for LRA Table 2.5-1 since no new components types were added.

The revised paragraphs in LRA Section 2.5.6.2 and the revised LRA Figure 2.5-1 supersede those in the original LRA. In the revised LRA Section 2.5.6.2, the applicant included within the scope of license renewal all components starting from transmission line circuit breakers ACB34560, ACB34561, ACB34562, ACB34563, ACB34564, 81-B-65, 81-B-66, 81-B-67

supplying the startup transformers 01 and 02 down to the 4.16 kV essential buses “C1” and “D1” and all components from and including the SBODG to the 4.16 kV essential buses “C1” and “D1.” The startup transformers 01 and 02 provide the in-scope power pathways into the plant and to the safety buses. As shown in revised LRA Figure 2.5-1, the startup transformers 01 and 02 step down the voltage from 345 kV to 13.8 kV to supply the 13.8 kV switchgear buses “A” and “B”, then the bus-tie transformers “AC” and “BD” step down the voltage from 13.8 kV to 4.16 kV to supply the 4.16 kV essential buses “C1” and “D1” through the 4.16 kV switchgear buses “C2” and “D2.” The circuit from the SBODG to the 4.16 kV essential buses “C1” and “D1” is through the 4.16 kV switchgear buses “D3” and “D2.”

The switchyard 345 kV “J” and “K” buses and connections, the transmission conductors and connections, high-voltage insulators, and the control circuits and protective relays for the switchyard circuit breakers (and the equipment associated with the “J” and “K” buses) are within the scope of license renewal. The applicant stated that the in-scope structural items (towers and foundations) and the switchyard relay house (where the switchyard control circuits and relays located) are evaluated in LRA Section 2.4. Components that are subject to AMR are included in LRA Table 2.5-1.

Based on the review of this information and the Davis-Besse USAR, the staff concludes that the scoping is consistent with the guidance in SRP-LR Section 2.5.2.1.1. The staff’s concern described in RAI 2.5.6.2a is resolved.

In LRA Section 2.5.5, “Electrical and I&C Component Commodity Groups Requiring an Aging Management Review,” the applicant evaluated the electrical and I&C component commodity groups that require AMR. These commodity groups are listed in LRA Table 2.5-1 along with their intended functions. The staff reviewed the commodity groups and did not identify any omissions of components from those that are subject to an AMR.

2.5.1.3 Conclusion

The staff reviewed the LRA Section 2.5, LRA Amendments 46, 50, and 53, and Davis-Besse USAR to determine whether the applicant identified all SSCs within the scope of license renewal and to determine whether the applicant had identified all components subject to an AMR. On the basis of its review, the staff concludes that the applicant adequately identified the electrical and I&C systems components within the scope of license renewal as required by 10 CFR 54.4(a), and those subject to an AMR, as required by 10 CFR 54.21(a)(1).

SECTION 3

AGING MANAGEMENT REVIEW RESULTS

3.0 Applicant's Use of the Generic Aging Lessons Learned (GALL) Report

3.0.3 Aging Management Programs (AMPs)

3.0.3.1 AMPs Consistent with the GALL Report

3.0.3.1.8 Flow-Accelerated Corrosion Program

Summary of Technical Information in the Application. In response to a steam line failure event in May 2015, the applicant provided supplemental operating experience information in LRA Amendment 58, by letter dated June 12, 2015. Based on issues identified during its investigation into the steam line failure event, the applicant revised LRA Table A-1 to include new license renewal Commitment No. 55 to improve and maintain the fidelity of the data in the Flow-Accelerated Corrosion Program.

Staff Evaluation. There are no changes or updates to this section of the SER.

Operating Experience. By letter dated June 12, 2015, the applicant revised LRA Section B.2.19, as a result of its investigation and corrective actions related to a steam line failure in May 2015. The applicant clarified that the corrective actions associated with an earlier steam leak in 2006 had enhanced the program by providing second level verification of CHECWORKS™ data to improve the software model quality, had focused on single-phase and two-phase flow aspects and did not include flow orifice input parameter verification. The steam line failure in 2015 was attributed to a data entry error in the CHECWORKS™ model for the restricting orifice, immediately upstream of the failed elbow. The applicant determined that the input error occurred during the original computer model development in the late 1980s and resulted in a nonconservative wear rate prediction from the inception of the Flow-Accelerated Corrosion Program. Corrective actions from this event included verifying all critical design inputs to the current CHECWORKS™ models, enhancing documentation requirements for changes to CHECWORKS™ models, and updating procedures for evaluating known accelerated wear rate configurations.

Based on its audit and review of the application, as modified on June 12, 2015, the staff finds that operating experience related to the applicant's program demonstrates that implementation of the program has resulted in the applicant taking appropriate corrective actions and that it can adequately manage the detrimental effects of aging within the scope of the program. The staff confirmed that the "operating experience" program element satisfies the criteria in SRP-LR Section A.1.2.3.10; therefore, the staff finds it acceptable.

USAR Supplement. Although the applicant did not revise the USAR supplement in LRA Amendment 58, the staff notes that in commitment 55 the applicant committed to performing several actions to improve and maintain the fidelity of the data in the Flow-Accelerated Corrosion Program, in response to the steam failure event in May 2015 and the resulting Root Cause Evaluation Corrective Actions. The applicant will review and validate the data inputs and

will create and maintain a document of the validation results as a quality record. The applicant committed to implementing these actions prior to October 22, 2016. The staff determined that the previous information in the USAR supplement continues to be an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. There are no changes or updates to this section of the SER.

3.0.3.1.10 Inservice Inspection (ISI) Program—IWE

Summary of Technical Information in the Application. The applicant provided additional information related to the Inservice Inspection (ISI) Program—IWE after the issuance of the SER. By letter dated June 23, 2014, the applicant stated that, during the Cycle 18 (Spring 2014) refueling outage, FENOC performed core bores to access the inside surface of the embedded steel containment vessel (SCV) described in Phase 1 of Commitment No. 39 related to addressing the potential for borated water degradation of the steel containment. The applicant also stated that there was no evidence of the presence of borated water in the concrete or in contact with the SCV, and the ultrasonic testing (UT) thickness measurement of the SCV performed at that location indicated that the thickness was above the nominal value of 1.5 inches. The applicant further stated that the compressive strength of the core samples tested was above design values and no aging effects were identified that required entry into the Corrective Action Program. The applicant also stated that it completed Phase 1 of Commitment No. 35, with acceptable results.

Staff Evaluation. The staff evaluated the core bore results of Phase 1 of Commitment No. 39 and noted that there was no evidence of the presence of borated water in the concrete or degradation of the inaccessible portion of the SCV due to borated water at the locations of the core bores. Therefore, the staff determines that its plan to impose a license condition associated with Commitment No. 39, as described in SER Section 3.0.3.1.10 (pages 3-47), dated September 2013, is not necessary and is deleted from this section of the SER.

Operating Experience. There are no changes or updates to this section of the SER.

USAR Supplement. LRA Section A.1.22 provides the USAR supplement for the ISI Program—IWE. The staff reviewed this USAR supplement description of the program and noted that it conforms to the recommended description for this type of program, as described in SRP-LR, Table 3.5-2. The staff also noted that, by letter dated June 23, 2014, the applicant amended the LRA to state that it completed Phase 1 of Commitment Nos. 35 and 39. The staff determined that the information in the USAR supplement, as amended, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. There are no changes or updates to this section of the SER.

3.0.3.1.18 Steam Generator Tube Integrity Program

Summary of Technical Information in the Application. LRA Section B.2.38 describes the existing Steam Generator Tube Integrity Program as consistent with GALL Report AMP XI.M19, “Steam Generators.” The applicant stated that the Steam Generator Tube Integrity Program is credited for aging management of cracking, denting, loss of material, and reduction in heat transfer of the steam generator (SG) tubes, as well as cracking of tube plugs and tube support plates. The applicant further stated that the Steam Generator Tube Integrity Program is performed as part of the overall Steam Generator Management Program; the program is based on technical specifications (TS) requirements, and the program is implemented in accordance

with Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines." The applicant further stated that the Steam Generator Tube Integrity Program includes secondary-side examinations to assist in the verification of tube integrity and the condition of the tube support plates. Additionally, the applicant stated that the Steam Generator Tube Integrity Program is a combination condition monitoring and mitigation program.

Staff Evaluation. During its audit, the staff reviewed the applicant's claim of consistency with the GALL Report. The staff also reviewed the plant conditions to determine if they are bounded by the conditions for which the GALL Report was evaluated. The staff compared elements one through six of the applicant's program to the corresponding elements of GALL Report AMP XI.M19. As discussed in the Audit Report, the staff confirmed that these elements are consistent with the corresponding elements of GALL Report AMP XI.M19. Based on its audit, the staff finds that elements one through six of the applicant's Steam Generator Tube Integrity Program are consistent with the corresponding program elements of GALL AMP XI.M19 and are, therefore, acceptable.

Operating Experience. LRA Section B.2.38 summarizes operating experience related to the Steam Generator Tube Integrity Program. The staff reviewed this information and interviewed the applicant's technical personnel to confirm that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL Report. During the audit, the staff independently confirmed that the applicant adequately incorporated and evaluated operating experience related to this program.

The applicant stated that, during each refueling outage (RFO), SG degradation assessments are performed in accordance with the provisions of NEI 97-06, "Steam Generator Program Guidelines," dated August 2005 and the Electric Power Research Institute (EPRI) pressurized water reactor (PWR) SG examination guidelines. These industry guidelines are based, in part, on operating experience and inspection results from other operating PWRs. Degradation assessment topics include SG tube degradation mechanisms, inspection and expansion requirements, tube repair criteria, structural limits, guidelines for testing, and chemical cleaning provisions.

The Davis-Besse original SGs were replaced during the Cycle 18 refueling outage (Spring 2014). The applicant stated that the nuclear station is currently in the first cycle of operation following SG replacement and has no identified tube degradation mechanisms.

The staff reviewed operating experience information, in the application and during the audit, to determine if the applicable aging effects and industry and plant-specific operating experience were reviewed by the applicant and are evaluated in the GALL Report. As discussed in the audit report, the staff conducted an independent search of the plant operating experience information to determine if the applicant adequately incorporated and evaluated operating experience related to this program. The applicant's operating experience demonstrates that plant-specific and industry acceptable practices are implemented and that the program is able to manage the aging effects of cracking, denting, loss of material, and reduction in heat transfer of the SG tubes, as well as cracking of tube plugs and tube support plates. During its review, the staff found no operating experience to indicate that the applicant's program would not be effective in adequately managing aging effects during the period of extended operation.

Based on its audit and review of the application, the staff finds that operating experience related to the applicant's program demonstrates that it can adequately manage the detrimental effects of aging within the scope of the program and that implementation of the program has resulted in

the applicant taking appropriate corrective actions. The staff confirmed that the "operating experience" program element satisfies the criterion in SRP-LR Section A.1.2.3.10; therefore, the staff finds it acceptable.

USAR Supplement. LRA Section A.1.38 provides the USAR supplement for the Steam Generator Tube Integrity Program. The staff reviewed this USAR supplement description of the program and noted that it conforms to the recommended description for this type of program, as described in SRP-LR Table 3.1-2. The staff determined that the information in the USAR supplement is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its review of the applicant's Steam Generator Tube Integrity Program, the staff finds all program elements consistent with the GALL Report. The staff concludes that the applicant demonstrated that the effects of aging will be adequately managed so that the intended function(s) will remain consistent with the current licensing basis (CLB) for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.2 AMPs Consistent with the GALL Report with Exceptions or Enhancements

3.0.3.2.1 Aboveground Steel Tanks Inspection Program

Summary of Technical Information in the Application. LRA Section B.2.2 describes the existing Aboveground Steel Tanks Inspection Program as consistent, with enhancements, with GALL Report AMP XI.M29, "Aboveground Steel Tanks." Subsequent to the submittal of the LRA, the staff issued LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation," which revised several AMPs, including the guidance for AMP XI.M29. By letter dated February 19, 2014, the applicant revised LRA Section B.2.2 in its entirety, based on its review of LR-ISG-2012-02 AMP XI.M29. The description of the applicant's revised program and the staff's evaluation of the applicant's revised AMP are discussed below.

In a letter dated February 19, 2014, the applicant described its amended Aboveground Steel Tanks Inspection Program as an existing program that, with enhancements, will be consistent with the GALL Report AMP XI.M29, as revised by LR-ISG-2012-02. The applicant stated that the Aboveground Steel Tanks Inspection Program manages the effects of loss of material and cracking on the outside and inside surfaces of aboveground tanks constructed on concrete or soil. The applicant also stated that the tanks included in the program are the steel diesel fuel oil outdoor storage tank, stainless steel borated water outdoor storage tank, and steel condensate indoor storage tanks. The applicant further stated that the program is a condition monitoring program that consists of periodic visual inspections of tank external surfaces and volumetric examinations of tank bottoms. The applicant stated that additional opportunistic tank bottom inspections would be performed whenever the tanks are drained. The applicant also stated that the program credits coatings or protective paint on external surfaces of steel tanks as a preventive measure to mitigate corrosion. In addition, the program relies on periodic inspections to monitor degradation of the coatings or protective paint.

Staff Evaluation. The staff noted that the applicant's amendment placed the fire water storage tank in the scope of its Fire Water Program, which is consistent with the staff's revised guidance. The staff's review of the applicant's Fire Water Program is discussed in

Section 3.0.3.2.8 of this SSER. The staff also noted that the applicant amended its program to include the two indoor condensate steel storage tanks in the scope of its amended Aboveground Steel Tanks Inspection Program, which is also consistent with the revised guidance in LR-ISG-2012-02.

The staff noted that LRA Section 2.3.3.21 states that the miscellaneous liquid radwaste system satisfies the scoping criteria for 10 CFR 54.4(a)(2). The staff also noted that license renewal drawing M-037C, "Clean Liquid Radioactive Waste System," states that the clean waste receiver tanks are in scope and have a nominal capacity of 103,000 gallons. During its review of the changes to the Aboveground Steel Tanks Program described by the applicant's letter dated February 19, 2014, the staff noted that the clean waste receiver tanks may meet the revised guidance but were not included in the scope of the program. By letter dated, July 7, 2014, the staff issued RAI 3.0.3.4.3-1, requesting that the applicant state whether the clean waste receiver tanks should be in the scope of LR-ISG-2012-02, AMP XI.M29. The staff requested that the applicant: (a) revise the program to include the clean water receiver tanks or (b) state the basis for why it is not necessary to include these tanks in the scope of the Aboveground Steel Tanks Inspection Program, and (c) state whether there are other indoor tanks that should be within the scope of the Aboveground Steel Tanks Inspection Program.

In its response dated July 29, 2014, the applicant stated that the top of the clean waste receiver tanks have a design pressure of 15 psig. The applicant further stated that because these indoor tanks have a design pressure above atmospheric, they are not within the scope of its Aboveground Steel Tanks Inspection Program. The staff finds the applicant's response to RAI 3.0.3.4.3-1 acceptable because the clean waste receiver tanks do not meet the scoping criteria to be included within the applicant's Aboveground Steel Tanks Inspection Program. The staff noted that the scoping criteria of GALL Report AMP XI.M29, as revised by LR-ISG-2012-02, requires that indoor tanks that have a storage capacity greater than 100,000 gallons, and a design pressure near atmospheric, be included within the scope of the program. The staff also noted that LRA Table 3.3.1, item 3.3.1-91, addresses stainless steel and steel with stainless steel cladding, piping, piping components, and piping elements exposed to treated borated water, which are being managed for loss of material due to pitting and crevice corrosion. The staff further noted that the applicant also applies this item to the clean waste receiver tanks; therefore, these components, although not within the scope of the Aboveground Steel Tanks Inspection Program, will be managed for aging. The staff's evaluation of components associated with item 3.3.1-91 is discussed in SER Section 3.3.2.1.24. The staff's concerns described in RAI 3.0.3.4.3-1 are resolved.

During the audit, the staff noted that the borated water storage tank (BWST) is insulated and is located in an environment that could result in stress corrosion cracking (SCC) of the stainless steel tank (e.g., located within 1/2 mile of a salt-treated highway, or other sources of chlorides). The staff also noted that the applicant's response to RAI B.2.2-2, dated May 24, 2011, states that: "(a) the outdoor air environment could result in an AERM for the BWST; (b) the polyurethane foam insulation installed on the BWST is limited to leach less than 1000 ppm chlorides; and (c) the Aboveground Steel Tanks Inspection Program was revised to manage SCC for the BWST."

Based on its review of Licensee Event Report No. 346-1982-001, Rev. 2, dated October 11, 1984, the staff noted that the insulation for the BWST was added a number of years after the plant was placed in service. The staff also noted that Regulatory Guide (RG) 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," Figure 1, recommends an upper limit of chlorides and fluorides dependent on the sodium and silicate content of the insulation.

Based on the applicant's response to RAI B.2.2-2 in relation to the chloride content of the insulation, the staff does not have adequate information to conclude that the leachable chlorides and fluorides from the BWST insulation, coupled with possible chloride contaminants present prior to the application of the insulation, would not result in SCC.

In addition, the insulation appears to be that which could be described as tightly adhering insulation. LR-ISG-2012-02 AMP XI.M29 does not require removal of tightly adhering insulation unless there is evidence of damage to the moisture barrier. The revised Aboveground Steel Tanks Inspection Program does not include inspections related to corrosion under insulation. It is not clear to the staff whether the inspections related to corrosion under insulation in the External Surfaces Monitoring Program include inspections of the BWST insulation. The number and periodicity of inspections in LR-ISG-2012-02 AMP XI.M29 are generally in alignment with those for LR-ISG-2012-02 AMP XI.M36, "External Surfaces Monitoring of Mechanical Components"; however, the recommended inspection locations are different. The staff needed clarification on the specific inspections and periodicity of inspections, which the applicant will use to manage loss of material due to pitting and crevice corrosion, as well as cracking due to SCC on the exterior surfaces of the BWST.

By letter dated July 7, 2014, the staff issued RAI 3.0.3.4.3-2, requesting that the applicant do the following:

(1) Confirm whether the insulation installed on the BWST is considered as tightly adhering insulation and impermeable to moisture.

(2) State whether inspections of the BWST insulation will be conducted under the External Surfaces Monitoring Program. If this is the case, state the extent and periodicity of inspections, the inspection methods, and how inspection locations will be selected. If no inspections will be conducted, state the basis for why there is reasonable assurance that the BWST will perform its current licensing-basis intended functions during the period of extended operation.

(3) State whether visual and surface examinations sufficient to detect loss of material due to pitting and crevice corrosion, and cracking were conducted on the external surfaces of the BWST prior to installing the insulation. If they were not conducted, state the basis for why bare metal inspections would not have to be conducted prior to the period of extended operation.

(4) Provide a summary of plant-specific operating experience related to the integrity of the BWST insulation. If there have been instances of damage to the insulation such that moisture could have penetrated to the surface of the BWST, state what inspections have been conducted on the bare metal surfaces of the tank. If none were conducted, state the basis for why bare metal inspections would not have to be conducted prior to the period of extended operation.

By letter dated July 29, 2014, the applicant provided its response to RAI 3.0.3.4.3-2. In its response to Part 1 of RAI 3.0.3.4.3-2, the applicant stated that the exterior accessible surfaces of the tank are painted and, with the exception of the roof, some nozzles, and the bottom, also insulated. The applicant also stated that the insulation for the BWST consists of multiple layers of coating and insulation; therefore, it is considered tightly adhering and impermeable to moisture.

The staff finds the applicant's response to RAI 3.0.3.4.3-2, Part 1, acceptable because the applicant clarified that the insulation for the BWST is considered tightly adhering and impermeable to moisture penetration. The staff's concerns described in RAI 3.0.3.4.3-2, Part 1, are resolved.

In its response to Part 2 of RAI 3.0.3.4.3-2, the applicant stated that the Aboveground Steel Tanks Inspection Program would be used to inspect the insulation for the BWST. The applicant also stated that the inspection periodicity, methods, and locations would be consistent with the GALL Report AMP XI.M29, as revised by LR-ISG-2012-02. As part of its response, the applicant revised LRA Sections A.1.2, B.2.2, and LRA Table A-1 accordingly. The staff finds the applicant's response to RAI 3.0.3.4.3-2, Part 2, acceptable because the applicant revised the Aboveground Steel Tanks Inspection Program to specifically include inspections for the exterior surfaces of the insulation for the BWST, consistent with the staff's revised guidance for GALL Report AMP XI.M29. The staff's concerns described in RAI 3.0.3.4.3-2, Part 2, are resolved.

In its response to Part 3 of RAI 3.0.3.4.3-2, the applicant stated that it could not locate any documentation that confirmed whether any surface inspections were conducted on the BWST prior to the application of the insulation, which could detect loss of material due to pitting, cracking, and crevice corrosion. The applicant also stated that the Aboveground Steel Tanks Inspection Program will be enhanced to specifically include bare metal inspection of the BWST exterior surfaces prior to the period of extended operation. As part of its response, the applicant revised LRA Sections A.1.2, B.2.2, and LRA Table A-1 accordingly. The staff finds the applicant's response to RAI 3.0.3.4.3-2, Part 3, acceptable because (a) the applicant revised the Aboveground Steel Tanks Inspection Program to specifically include inspections of the exterior bare metal surfaces of the BWST for loss of material and cracking and (b) these inspections would be completed prior to the start of extended operation. The staff's concerns described in RAI 3.0.3.4.3-2, Part 3, are resolved.

In its response to Part 4 of RAI 3.0.3.4.3-2, the applicant stated that a 2011 condition report described coating damage to the insulation of the BWST. The applicant also stated that 120 square feet of the butyl rubber coating was detached (delaminated) from the polyurethane foam insulation. The applicant further stated that the insulation was not damaged, and there was no evidence of any moisture penetration to the BWST. The applicant stated that there were no records to indicate any instances of insulation damage or delamination, or bare metal inspections of the BWST. The applicant also restated that, as discussed in its response for Part 3 of this RAI, the enhanced Aboveground Steel Tanks Program includes a bare metal inspection of the exterior surfaces of the BWST prior to the period of extended operation.

The staff finds the applicant's response to RAI 3.0.3.4.3-2, Part 4, acceptable because (a) the applicant provided the available operating experience related to the insulation for the BWST, which indicated that there have been no known instances of damage or evidence of moisture intrusion or penetration to the foam insulation, (b) the exterior coating applied to the BWST and the multilayered nature of the insulation should provide protection against moisture intrusion, (c) the exterior surfaces of the insulation will be inspected, and (d) bare metal surface examinations performed prior to the period of extended operation will provide confirmation of the adequacy of the AMP. The staff's concerns described in RAI 3.0.3.4.3-2, Part 4, are resolved.

Based on its review of the applicant's Aboveground Steel Tanks Inspection Program as amended by LRA Amendment 48, dated February 19, 2014, and revised by letter dated July 29, 2014, response to RAI 3.0.3.4.3-1, and response to RAI 3.0.3.4.3-2, the staff finds that

elements one through six of the applicant's program, with the enhancements, are consistent with the corresponding program elements of LR-ISG-2012-02 AMP XI.M29. Therefore, the staff finds the applicant's Aboveground Steel Tanks Inspection Program acceptable.

Operating Experience. The staff does not have any changes or updates to this section of the SER.

USAR Supplement. The staff reviewed LRA Section A.1.2, which provides the applicant's USAR supplement for the Aboveground Steel Tanks Inspection Program, as amended by letters dated February 19, 2014, and July 29, 2014. The staff reviewed this USAR supplement description of the program and noted that it conforms to the recommended description for this type of program, as described in LR-ISG-2012-02, Table 3.0-1. The staff also noted that the applicant committed (Commitment No. 1) to enhancing the existing Aboveground Steel Tanks Inspection Program prior to entering the period of extended operation, in accordance with the enhancements described above.

The staff determined that the information in the USAR supplement, as amended, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its review of the applicant's Aboveground Steel Tanks Inspection Program, the staff determined that those program elements for which the applicant claimed consistency with the GALL Report are consistent. In addition, the staff reviewed the enhancements and confirmed that their implementation, through Commitment No. 1, prior to the period of extended operation, would make the existing AMP consistent with the GALL Report AMP, as revised by LR-ISG-2012-02, to which it was compared. The staff concludes that the applicant demonstrated that the effects of aging will be adequately managed so that the intended function(s) will remain consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.2.3 Buried Piping and Tanks Inspection Program

Summary of Technical Information in the Application. Subsequent to the submittal of the LRA, the staff issued LR-ISG-2011-03, "Changes to the Generic Aging Lessons Learned (GALL) Report Revision 2 Aging Management Program (AMP) XI.M41, 'Buried and Underground Piping and Tanks.'" By letter dated September 20, 2013, the applicant submitted changes to its Buried Piping and Tanks Inspection Program to incorporate the changes recommended in LR-ISG-2011-03.

Staff Evaluation. The staff's evaluation of the changes to the applicant's Buried Piping and Tanks Inspection Program follows.

Extent of Fuel Oil Piping Inspections. By letter dated September 20, 2013, the applicant amended the extent of piping inspections (i.e., 2 percent of the buried in-scope piping containing hazardous materials) for buried fuel oil piping to address the changes in recommendations issued in LR-ISG-2011-03. The applicant stated that the extent of piping inspections will be consistent with LR-ISG-2011-03, Table 4a, "Inspections of Buried Pipe." The staff noted that Table 4a recommends the extent of inspections based on the material type and availability and effectiveness of preventive actions (i.e., cathodic protection availability and effectiveness, coatings, and backfill) and not on the contents of the piping.

The staff finds this change acceptable because it is consistent with LR-ISG-2011-03 in regard to the extent of piping inspections.

Cathodic Protection Acceptance Criteria. By letter dated September 20, 2013, the applicant amended the acceptance criteria for cathodic protection ground surveys to address the changes in recommendations issued in LR-ISG-2011-03. As amended, the cathodic protection acceptance criteria are negative 850 mV relative to a copper/copper sulfate reference electrode (CSE) instant off or negative 100 mV minimum polarization, and limiting the potential to be not more negative than 1,200 mV.

The staff noted that LR-ISG-2011-03, Table 6a, "Cathodic Protection Acceptance Criteria," recommends that:

> The 100 mV polarization criterion is limited to electrically isolated piping sections or areas of grounded piping where the effects of mixed potentials are shown to be minimal. When the 100 mV criterion is utilized in lieu of the - 850 mV CSE criterion for steel piping, or where copper or aluminum components are protected, applicants must explain in the application why the effects of mixed potentials are minimal and why the most anodic metal in the system is adequately protected.

In addition, Table 6a recommends that polarized potentials not be greater than negative 1,200 mV to avoid potentially detrimental effects (e.g., coating disbondment) associated with excessive overprotection. By letter dated February 11, 2014, the staff issued RAI 2011-03-1, requesting that the applicant state the basis for use of the negative 100 mV minimum polarization criterion.

In its response dated March 11, 2014, the applicant stated that the negative 100 mV minimum polarization criterion is applied to the manways and vents located at the top of the two EDG fuel oil storage tanks. The tanks are installed above grade elevation with tornado missile protection provided by a truncated pyramid of structural backfill built around the tanks (mound-buried). The negative 850 mV polarized potential criterion is being met at all locations except the top of the mound, which is approximately 13 feet above grade. At all other locations that are cathodically protected, the negative 850 mV polarized potential instant off acceptance criterion will be applied. To ensure that the man ways and vents at the top of the mound over the EDG fuel oil storage tanks are not degrading due to the lower polarization criterion, UT thickness measurements will be performed prior to entering the period of extended operation and every 10 years during the period of extended operation to ensure that the metal thickness in those areas remains satisfactory.

The staff noted that the applicant revised LRA Section B.2.7 and Commitment No. 3 to reflect the above changes.

The staff finds the applicant's response acceptable because, in the locations where the negative 100 mV polarization will be used, the applicant will periodically confirm that the cathodic protection is providing effective protection by volumetric wall thickness measurements. These inspections will provide a direct indication that the effects of mixed potentials have been minimal and that, therefore, the program will be consistent with LR-ISG-2011-03. The staff's concern described in RAI 2011-03-1 is resolved. The staff also finds the applicant's cathodic protection acceptance criteria acceptable because they are consistent with LR-ISG-2011-03, ensuring that the effectiveness of the cathodic protection system can be determined.

Sample Size Expansion Criteria. By letter dated September 20, 2013, the applicant amended the sample size expansion criteria to address the changes in recommendations issued in LR-ISG-2011-03. The applicant added a requirement that, if adverse indications are detected, the inspection sample sizes, within the affected piping categories, are initially doubled, and if adverse conditions are discovered in the expanded sample, the size of the follow-on inspections is determined by establishing the extent of condition and extent of cause. The applicant stated that scheduling of additional examinations will be based on the severity of the degradation identified and commensurate with the consequences of a leak or loss of function, but in all cases, the expanded sample inspection will be completed within the 10-year interval in which the original adverse indication was identified. The staff finds this change acceptable because it is consistent with LR-ISG-2011-03 in regard to the extent of piping inspections, ensuring that an appropriate number of additional inspections will be conducted when adverse conditions are detected.

Changes in Classification of Piping. Prior to issuance of LR-ISG-2011-03, GALL Report AMP XI.M41 recommended that the number of inspections to be conducted for a specific material of buried in-scope piping was, in part, based on its contents and function (i.e., code class, safety-related, contains hazardous materials) under the "detection of aging effects" program element description. In its response to RAI B.2.7-1(12) dated May 24, 2011, the applicant stated that the components cited in LRA Table 3.3.2-12, row 102, have an internal environment of fuel oil. However, LR-ISG-2011-03, as stated in Table 4a, bases the extent of the inspection recommendations on the material type and availability and effectiveness of preventive actions (i.e., cathodic protection availability and effectiveness, coatings, and backfill) and not on the contents of the piping. Therefore, the response to RAI B.2.7-1(12) is no longer technically relevant. As stated above, in subsection *Extent of Fuel Oil Piping Inspections*, the applicant's extent of inspections related to buried fuel oil piping is appropriately based on Table 4a of LR-ISG-2011-03.

Underground Piping Volumetric Inspections. By letter dated September 20, 2013, the applicant amended the underground inspection requirements of its program to address the changes in recommendations issued in LR-ISG-2011-03. The applicant stated that volumetric inspections of underground piping will not be conducted. The staff noted that the recommendation to perform volumetric examinations of underground piping was removed from the scope of GALL Report AMP XI.M41 by LR-ISG-2011-03 because the purpose of volumetric examinations is to detect aging effects occurring on the internal surfaces of the piping, whereas the purpose of AMP XI.M41 is to manage aging effects associated with the external surfaces of the components. In a prior response dated May 24, 2011, the applicant had amended its program to include visual inspections of the external surfaces of in-scope underground piping. The staff finds this change acceptable because conducting visual examinations of the external surfaces of in-scope underground piping is consistent with LR-ISG-2011-03.

The applicant amended Enhancement No. 2 and added 12 new enhancements, Enhancement Nos. 3 through 14.

Enhancement 2. By letter dated September 20, 2013, the applicant amended Enhancement No. 2. The applicant stated that the extent of piping inspections will be consistent with LR-ISG-2011-03, Table 4a, "Inspections of Buried Pipe." Table 4a recommends the extent of inspections be based on the material type and condition of preventive actions (i.e., cathodic protection availability and effectiveness, coatings, and backfill). On the basis of its review, the staff finds the applicant's change to Enhancement No. 2 acceptable because, when it is

implemented prior to the period of extended operation, it will make the program's extent of inspections consistent with the recommendations in LR-ISG-2011-03.

Enhancement 3. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "parameters monitored or inspected" program element. The applicant stated that it will conduct annual potential surveys of the cathodic protection system. The applicant also stated that the cathodic protection voltage and current will be monitored monthly. The applicant further stated that voltage, current, and ground potential readings will be trended and evaluated for adverse trends. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will make the program consistent with the recommendations in LR-ISG-2011-03, and because the proposed parameters and frequency of testing will be sufficient to demonstrate the availability and effectiveness of the cathodic protection system.

Enhancement 4. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant stated that it will monitor the activity of the jockey pump or equivalent parameter on at least a monthly interval. The applicant also stated that, when unresolved changes in jockey pump activity are observed, a flow test will be conducted by the end of the next refueling outage. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will make the program consistent with the recommendations in LR-ISG-2011-03, and the proposed monitoring and followup testing will be effective at detecting leakage in buried fire protection piping.

Enhancement 5. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant stated that it will select buried pipe inspection locations based on risk. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will make the program consistent with the recommendations in LR-ISG-2011-03, and inspections of the highest risk locations will ensure that the most critical piping locations are inspected.

Enhancement 6. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant added a requirement to inspect the EDG fuel oil storage tanks prior to the period of extended operation. The applicant stated that the inspection will be either (a) a visual inspection of at least 25 percent of each tank and include at least some portion of the tank top and bottom, or (b) an internal inspection consisting of UT measurements with at least one measurement per square foot of the surface of the tanks. The applicant also stated that these inspections are not required if it is demonstrated that the tanks are cathodically protected in accordance with National Association of Corrosion Engineers (NACE) SP0169-2007, "Control of External Corrosion on Underground or Submerged Metallic Piping Systems," or NACE RP0285-2002, "Corrosion Control of Underground Storage Tank Systems by Cathodic Protection." In addition, as described above in the response to RAI 2011-03-1, the applicant amended this enhancement to include the UT measurements of the man ways and vents for the EDG fuel oil storage tanks.

The staff noted that the applicant has committed (Commitment No. 44) to cathodically protecting the EDG fuel oil storage tanks prior to entering the period of extended operation. The staff also noted that LR-ISG-2011-03, Table 4c, "Inspections of Buried Tanks for all Inspection Periods," recommends that tank inspections not be required if the tank is cathodically protected in accordance with certain provisions of footnote 3 of the Table. However, in the event that the

tanks are not cathodically protected (due to a future commitment change), the staff considers the incorporation of the Table 4c inspections into the commitment necessary to ensure that aging will be managed throughout the period of extended operation. By letter dated February 11, 2014, the staff issued RAI 2011-03-2, requesting that the applicant revise Enhancement No. 6 to require that, if cathodic protection is not provided, tank inspections meet the recommendations of Table 4c, or state the basis for why tank inspections would not be required during the period of extended operation.

In its response dated March 11, 2014, the applicant stated that cathodic protection was installed and has been operational for the in-scope buried EDG fuel oil storage tanks and associated fuel oil supply piping from the storage tanks to the EDGs in the auxiliary building since April 2012. In addition, the buried service water system piping and diesel fuel oil piping from the aboveground diesel oil storage tank to the auxiliary building have been cathodically protected. As such, the applicant stated that it completed Commitment No. 44.

The staff noted that the applicant revised LRA Section B.2.7 and Commitment No. 3 to state that, if the cathodic protection system for the EDG fuel oil storage tanks meets the availability criteria in Table 4c., inspections would not need to be conducted. Otherwise, inspections will be performed in accordance with Table 4c.

The staff finds the applicant's response, closure of Commitment No. 44, and enhancement acceptable because cathodic protection has been installed as committed and the tank inspections will be conducted if the cathodic protection system does not meet the availability and effectiveness goals described in the footnotes to Table 4c. The staff's concern described in RAI 2011-03-2 is resolved.

Enhancement 7. As amended by letters dated May 24, 2011, and September 20, 2013, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant stated that it will conduct a visual inspection of the underground piping within the borated water piping trench during each 10-year period, beginning no sooner than 10 years prior to entry into the period of extended operation. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will make the program consistent with the recommendations in LR-ISG-2011-03, Table 4b, "Inspections of Underground Piping for all Inspection Periods," and visual inspections of underground pipe can detect the loss of material and cracking on the external piping surfaces.

Enhancement 8. As amended by letters dated May 24, 2011, and September 20, 2013, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant stated that, if adverse conditions are detected, it will conduct inspection sample-size expansions as quoted above in the response to RAI B.2.7-1(10). On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, the inspection sample-size expansion criteria will be consistent with LR-ISG-2011-03 and will ensure that an appropriate number of additional inspections will be conducted when adverse conditions are detected.

Enhancement 9. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant stated that it will conduct inspections of buried fire protection system bolting when the bolting becomes accessible during opportunistic or focused inspections. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of

extended operation, it will be consistent with the "scope of program" element of LR-ISG-2011-03, which states that the program manages loss of material due to corrosion of buried piping system bolting.

Enhancement 10. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "detection of aging effects" program element. The applicant stated that it will conduct inspections of buried piping using visual (VT-3 or equivalent) inspection methods, and the inspections will encompass a minimum of 10 linear feet of piping, with all surfaces of the pipe exposed. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will be consistent with LR-ISG-2011-03 and will ensure that visual examinations are conducted on a sufficiently lengthy segment of piping in order to detect degradation such as damage to coatings or loss of material.

Enhancement 11. As amended by letters dated May 24, 2011, September 20, 2013, and March 11, 2014, LRA Section B.2.7 states an enhancement to the "acceptance criteria" program element. The applicant stated that the acceptance criteria for the cathodic protection system performance will be as stated above in the response to RAIs B.2.7-1(3) and 2011-03-1. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, the cathodic protection acceptance criteria will be consistent with LR-ISG-2011-03 and will ensure that the effectiveness of the cathodic protection system can be determined.

Enhancement 12. As amended by letters dated May 24, 2011, and September 20, 2013, LRA Section B.2.7 states an enhancement to the "acceptance criteria" program element. The applicant stated that:

> 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 Coating Inspector Program Level 2 or 3 inspector qualification or an individual has attended the Electric Power Research Institute (EPRI) Comprehensive Coatings Course and completed the EPRI Buried Pipe Condition Assessment and Repair Training Computer Based Training Course. Where damage to the coating has been evaluated as significant and the damage was caused by non-conforming backfill, an extent of condition evaluation should be conducted to ensure that the as-left condition of backfill in the vicinity of observed damage will not lead to further degradation.

On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will be consistent with LR-ISG-2011-03 and will ensure that adverse conditions are evaluated by individuals qualified to make effective decisions and that an extent-of-condition evaluation is conducted to determine the extent of nonconforming backfill.

Enhancement 13. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "acceptance criteria" program element. The applicant stated that, if 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. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to

the period of extended operation, it will be consistent with LR-ISG-2011-03 and will ensure that there is reasonable assurance that loss of material is adequately managed.

Enhancement 14. As amended by letter dated May 24, 2011, LRA Section B.2.7 states an enhancement to the "acceptance criteria" program element. The applicant stated that an acceptance criterion will be established to state that changes in jockey pump activity (e.g., longer run periods, more frequent starts) or equivalent parameters that are attributed to leakage from buried piping will not be acceptable. On the basis of its review, the staff finds the applicant's enhancement acceptable because, when it is implemented prior to the period of extended operation, it will be consistent with LR-ISG-2011-03, which allows jockey pump monitoring, or an equivalent parameter, to be used to demonstrate that buried fire water system piping will meet its current licensing-basis intended function(s).

Based on its review of the above changes the staff finds that elements one through six of the applicant's Buried Piping and Tanks Inspection Program, with acceptable enhancements, are consistent with the corresponding program elements of LR-ISG-2011-03 and, therefore, are acceptable.

Operating Experience. The staff noted that LR-ISG-2011-03 states that a 10-year search of plant-specific operating experience should be conducted if cathodic protection is not provided. The staff also noted that the applicant has committed (Commitment No. 41) to cathodically protecting the EDG fuel oil storage tanks and in-scope fuel oil and service water piping prior to entering the period of extended operation. The staff concludes that the applicant's original search of 5 years of plant-specific operating experience is sufficient because cathodic protection will be provided for buried in-scope components prior to the period of extended operation.

As addressed in the original SER, by letter dated April 20, 2011, the staff issued RAI B.2.7-1(6), requesting that the applicant state the basis for having reasonable assurance that the planned inspections represent an adequate quantity to identify coating damage and holidays before leaks occur. In its response to RAI B.2.7-1(6) dated May 24, 2011, the applicant cited the increase in inspections in the time frame from 10 years prior to the period of extended operation to the end of extended operation from two to six inspections and that the inspection locations will be selected based on previous examination results, trending, risk ranking, and areas of cathodic protection failures or gaps as the basis for establishing reasonable assurance that the buried in-scope components will meet their current licensing-basis intended functions. As amended by letter dated September 20, 2013, the extent of the applicant's buried pipe inspections will instead be consistent with LR-ISG-2011-03, Table 4a., which recommends that the extent of inspections be based on the material type and condition of preventive actions (i.e., cathodic protection availability and effectiveness, coatings, and backfill). The staff continues to find the applicant's response to RAI B.2.7-1(6) acceptable because the number of inspections will be consistent with LR-ISG-2011-03 Table 4a, which states that the number of inspections is increased based on the condition of preventive actions (i.e., cathodic protection availability and effectiveness, coatings, and backfill).

USAR Supplement. As amended by letter dated September 20, 2013, LRA Section A.1.7 provides the USAR supplement for the Buried Piping and Tanks Inspection Program. The staff reviewed this USAR supplement description of the program and noted that it conforms to the recommended description for this type of program, as described in LR-ISG-2011-03, Table 3.0-1. The staff determines that the information in the USAR supplement, as amended, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its review of the changes to the applicant's Buried Piping and Tanks Inspection Program, the staff determined that those program elements for which the applicant claimed consistency with the GALL Report, as revised by LR-ISG-2011-03, are consistent. Also, the staff reviewed the enhancements and confirmed that their implementation, through Commitments No. 3 and No. 44, prior to the period of extended operation, would make the existing AMP consistent with the GALL Report AMP to which it was compared. The staff concludes that the applicant demonstrated that the effects of aging will be adequately managed so that the intended function(s) will remain consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21.

3.0.3.2.5 External Surfaces Monitoring Program

Summary of Technical Information in the Application. Subsequent to the submittal of the LRA, the staff issued LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation," which revised GALL Report AMPs associated with managing loss of material due to corrosion under insulation. By letter dated February 19, 2014, the applicant provided the results of its review and changes to the LRA associated with the staff's recommendations in LR-ISG-2012-02 Section E, "Corrosion Under Insulation," and associated appendices. The staff's evaluation of these changes follows. The staff's evaluation of the applicant's revisions to its Aboveground Steel Tanks Inspection Program to address corrosion under insulation for outdoor tanks is documented in SSER Section 3.0.3.2.1.

Staff Evaluation. The staff's evaluation of the changes to the applicant's External Surfaces Monitoring Program follows.

By letter dated February 19, 2014, the applicant revised the program description, and "scope of program" element of its External Surfaces Monitoring Program to also state that outdoor insulated components, and indoor insulated components exposed to condensation (because the in-scope component is operated below the dew point), have portions of the insulation inspected or removed to determine whether the exterior surface of the component is degrading or has the potential to degrade. The applicant also revised the "detection of aging effects" program element to address the selection of sample locations, sample size, method of inspection, and frequency of inspections for inspections associated with corrosion under insulation.

The staff finds the applicant's changes to the External Surfaces Monitoring Program acceptable because the details associated with inspection locations, sample size, methodology, and frequency are consistent with the staff's inspection and sampling recommendations of AMP XI.M36, as revised by LR-ISG-2012-02.

The staff also reviewed the portions of the "scope of program," and "detection of aging effects" program elements associated with the additional enhancements of the applicant's program, to determine if the program will be adequate to manage the aging effects for which it is credited. The staff's evaluation of these enhancements follows.

Enhancement 7. As amended by letter dated February 19, 2014, LRA Section B.2.15 states an enhancement to the "scope of program" program element.

The applicant stated that the program will be enhanced to include inspection of outdoor insulated components, and indoor insulated components exposed to condensation (because the in-scope component is operated below the dew point).

The staff reviewed this enhancement against the corresponding program elements in GALL Report AMP XI.M36, as modified by LR-ISG-2012-02. On the basis of its review, the staff finds this enhancement acceptable because the revised scope is consistent with AMP XI.M36, as modified by LR-ISG-2012-02.

Enhancement 8. As amended by letter dated February 19, 2014, LRA Section B.2.15 states an enhancement to the "detection of aging effects" program element. The applicant stated that the program will be enhanced to include details associated with the selection of sample locations, sample size, method of inspection, and frequency of inspections to address corrosion under insulation.

The staff reviewed this enhancement against the corresponding program elements in GALL Report AMP XI.M36, as modified by LR-ISG-2012-02. On the basis of its review, the staff finds this enhancement acceptable because the revised scope is consistent with AMP XI.M36, as modified by LR-ISG-2012-02.

Based on its review of the applicant's letter dated February 19, 2014, the staff finds that elements one through six of the applicant's External Surfaces Monitoring Program are consistent with the corresponding program elements of GALL Report AMP XI.M36, as modified by LR-ISG-2012-02 and, therefore, are acceptable.

USAR Supplement. As amended by letter dated February 19, 2014, LRA Section A.1.15 provides the USAR supplement for the External Surfaces Monitoring Program. The staff reviewed this USAR supplement description of the program and noted that it conforms to the recommended description for this type of program, as described in SRP-LR Table 3.0-1, as modified by LR-ISG-2012-02.

The staff determined that the information in the USAR supplement is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its review of the proposed changes to the External Surfaces Monitoring Program, as amended by letter dated February 19, 2014, the staff determined that those program elements for which the applicant claimed consistency with AMP XI.M36, as revised by LR-ISG-2012-02, are consistent. The staff concludes that the applicant demonstrated that the effects of aging will be adequately managed so that the intended function(s) will remain consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.2.8 Fire Water Program

Summary of Technical Information in the Application. Subsequent to the submittal of the LRA, the staff issued LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation," which revised several AMPs, including the guidance for AMP XI.M27. By letter dated February 19, 2014, (LRA Amendment 48) the applicant revised LRA Section B.2.18, based on its review of LR-ISG-2012-02 AMP XI.M27.

Staff Evaluation. The staff's evaluation of LRA Amendment 48 is as follows.

Flow Blockage as an Aging Effect. The staff noted that the applicant amended LRA Table 3.3.2-14 to include flow blockage (in addition to loss of material) as an AERM for multiple component types and materials exposed to raw water. The staff finds this acceptable because the addition of flow blockage as an aging effect is consistent with LR-ISG-2012-02 AMP XI.M27.

Fire Water Storage Tank. The staff noted that the applicant deleted the AMR item for the fire water storage tank exposed to outdoor air being managed for loss of material by the Aboveground Steel Tanks Inspection Program in LRA Table 3.3.2-14 because aging effects associated with this tank will be managed by the Fire Water Program. The staff also noted that, as stated in Enhancement No. 5, the applicant will conduct inspections and tests in accordance with LR-ISG-2012-02 Table 4a, "Fire Water System Inspection and Testing Recommendations." This table includes inspections of the external surfaces of the fire water storage tank; therefore, the staff finds the deletion of the AMR item acceptable.

Followup Wall Thickness Examinations Due to Detection of Surface Irregularities. The staff noted that the applicant's changes to the Fire Water Program did not state whether the internal visual inspections used to detect loss of material would be capable of detecting surface irregularities that could be indicative of wall loss below nominal pipe wall thickness due to corrosion and corrosion product deposition. In addition, the program did not address the actions that would be taken if such irregularities are detected. The staff also noted that the Fire Water Program did not state what actions would be taken if the presence of sufficient foreign organic or inorganic material to obstruct pipe or sprinklers is detected during pipe inspections. By letter dated July 7, 2014, the staff issued RAI B.2.18-2 requesting that the applicant state the basis for not including the above information in the Fire Water Program,

In its response dated July 29, 2014, the applicant revised the Fire Water Program to include two new enhancements (Enhancement Nos. 7 and 8) to address the staff's concern described in RAI B.2.18-2. The staff's evaluation of these enhancements is described below.

In LRA Amendment 48, the applicant deleted Enhancement Nos. 1, 2, and 4, amended Enhancement No. 3, and added Enhancement Nos. 5 through 8. The staff's evaluation of these changes is documented as follows.

Enhancement 1. LRA Section B.2.18 states an enhancement to the "parameters monitored or inspected" and "detection of aging effects" program elements to add a program requirement to perform periodic UT for wall thickness of representative above-ground water suppression piping that is not periodically flow tested but contains, or has contained, stagnant water. As amended by letter dated February 19, 2014, the applicant deleted this enhancement. The staff finds the deletion of this enhancement acceptable because the inspections and tests recommended in LR-ISG-2012-02 AMP XI.M27 Table 4a, described in Enhancement No. 5, augmented inspections and tests described in Enhancement No. 6, and followup ultrasonic examinations when internal visual inspections reveal surface irregularities that could be indicative of wall loss below nominal pipe wall thickness are sufficient to establish reasonable assurance that the fire water system will be able to perform its CLB intended function(s) during the period of extended operation.

Enhancement 2. LRA Section B.2.18 states an enhancement to the "detection of aging effects" program element to add a program requirement to perform at least one opportunistic or focused visual inspection of the internal surface of buried fire water piping and of similar above-ground

fire water piping within the 5-year period prior to the period of extended operation. As amended by letter dated February 19, 2014, the applicant deleted this enhancement. The staff finds the deletion of this enhancement acceptable because the inspections and tests recommended in LR-ISG-2012-02 AMP XI.M27 Table 4a described in Enhancement No. 5, augmented inspections and tests described in Enhancement No. 6, and followup ultrasonic examinations when internal visual inspections reveal surface irregularities that could be indicative of wall loss below nominal pipe wall thickness, are sufficient to establish reasonable assurance that the fire water system will be able to perform its CLB intended function(s) during the period of extended operation.

Enhancement 3. As amended by letter dated February 19, 2014, LRA Section B.2.18 states an enhancement to the "detection of aging effects" program element to add a program requirement to perform representative sprinkler head sampling or replacement prior to 50 years of service and at 10-year intervals thereafter, in accordance with the 2011 Edition of National Fire Protection Association (NFPA) 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems," or until there are no untested sprinkler heads that will see 50 years of service through the end of the period of extended operation. The staff reviewed this enhancement against the corresponding program element in GALL Report AMP XI.M27. GALL Report AMP XI.M27 states that 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. The staff finds the applicant's enhancement acceptable because the applicant will inspect the sprinkler heads or replace them prior to 50 years of service, which is consistent with the recommendations in GALL Report AMP XI.M27.

Enhancement 4. LRA Section B.2.18 states an enhancement to the "detection of aging effects" program element to add a program requirement to perform opportunistic fire water supply and water-based suppression system internal inspections each time one of these systems is opened for repair or maintenance. As amended by letter dated February 19, 2014, the applicant deleted this enhancement. The staff finds the deletion of this enhancement acceptable because the inspections and tests recommended in LR-ISG-2012-02 AMP XI.M27 Table 4a, described in Enhancement No. 5, augmented inspections and tests described in Enhancement No. 6, and followup ultrasonic examinations when internal visual inspections reveal surface irregularities that could be indicative of wall loss below nominal pipe wall thickness, are sufficient to establish reasonable assurance that the fire water system will be able to perform its CLB intended function(s) during the period of extended operation.

Enhancement 5. As amended by letter dated February 19, 2014, LRA Section B.2.18 states an enhancement to the "detection of aging effects" program element to include the inspections and tests recommended in LR-ISG-2012-02 Table 4a. The staff reviewed this enhancement against the corresponding program element in LR-ISG-2012-02 AMP XI.M27. The AMP recommends the inspections and tests in Table 4a in order to detect potential loss of material and flow blockage. The staff finds the applicant's enhancement acceptable because it is consistent with LR-ISG-2012-02 AMP XI.M27.

Enhancement 6. As amended by letter dated February 19, 2014, LRA Section B.2.18 states an enhancement to the "detection of aging effects" program element to include augmented inspections and tests of water-based fire protection system piping segments that are normally dry but periodically subject to flow that either cannot be drained or will allow water to collect. The inspections and tests of these piping segments will consist of the following:

- A flow test or flush sufficient to detect potential flow blockage will be conducted in each 5-year interval beginning 5 years prior to the period of extended operation. Alternatively, a visual inspection of 100 percent of the internal surface of piping segments will be conducted.
- Volumetric wall thickness inspections will be conducted on 20 percent of the length of the piping segments in each 5-year interval of the period of extended operation. Measurement points are obtained sufficient to ensure that each potential aging effect can be identified (e.g., general corrosion, microbiologically-influenced corrosion (MIC)). The 20 percent of piping that is inspected in each 5-year interval will be in different locations than previously inspected piping.
- Further tests or inspections will not be conducted if the 100-percent internal visual inspections are acceptable and the piping segment is not subsequently wetted.

The staff reviewed this enhancement against the corresponding program element in LR-ISG-2012-02 AMP XI.M27. The AMP recommends these inspections and tests to ensure that potential corrosion product buildup due to accelerated corrosion as a result of the water and air mixture in the piping will not result in flow blockage. The staff finds the applicant's enhancement acceptable because it is consistent with LR-ISG-2012-02 AMP XI.M27.

Enhancement 7. As amended by letter dated July 29, 2014, LRA Section B.2.18 states an enhancement to the "parameters monitored or inspected" program element to require that, "when visual inspections are used to detect loss of material, the inspection technique is capable of detecting surface irregularities that could indicate wall loss to below nominal pipe wall thickness due to corrosion and corrosion product deposition. Where such irregularities are detected, followup volumetric wall thickness examinations are performed. The staff reviewed this enhancement against the corresponding program element in LR-ISG-2012-02 AMP XI.M27. The AMP recommends that the visual inspections be capable of detecting surface irregularities and followup volumetric wall thickness examinations be performed to ensure that, where loss of material is detected, the remaining wall thickness will be adequate to ensure that the fire water system will be capable of performing its current licensing-basis intended function(s) during the period of extended operation. The staff finds the applicant's enhancement acceptable because it is consistent with LR-ISG-2012-02 AMP XI.M27.

Enhancement 8. As amended by letter dated July 29, 2014, LRA Section B.2.18 states an enhancement to the "acceptance criteria" program element to require that, "if the presence of sufficient foreign organic or inorganic material to obstruct pipe or sprinklers is detected during pipe inspections, the material is removed and its source is determined and corrected."

The staff reviewed this enhancement against the corresponding program element in LR-ISG-2012-02 AMP XI.M27. The AMP recommends that evidence of foreign material capable of obstructing pipe or sprinklers be removed and the source determined and corrected in order to ensure that there is reasonable assurance that flow blockage will not occur in the fire water system. The staff finds the applicant's enhancement acceptable because it is consistent with LR-ISG-2012-02 AMP XI.M27.

Based on the inclusion of Enhancement Nos. 7 and 8, the staff's concern described in RAI B.2.18-2 is resolved. In addition, based on its review of the applicant's response to RAI B.2.18-2 by letter dated February 19, 2014, the staff finds that elements one through six of the applicant's Fire Water Program, with acceptable enhancements, are consistent with the corresponding program elements of LR-ISG-2012-02 AMP XI.M27, and, therefore, are acceptable.

Operating Experience. The staff does not have any changes or updates to this section of the SER.

USAR Supplement. As amended by letter dated February 19, 2014, LRA Section A.1.18 provides the USAR supplement for the Fire Water Program. The staff reviewed this USAR supplement description of the program and noted that it conforms to the recommended description for this type of program, as described in LR-ISG-2012-02, Table 3.0-1. The staff also noted that the applicant committed (Commitment No. 10), as amended by letter dated July 29, 2014, to enhancing the existing Fire Water Program prior to entering the period of extended operation in accordance with the enhancements described above.

The staff determines that the information in the USAR supplement, as amended, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. The staff does not have any changes or updates to this section of the SER.

3.0.3.2.11 One-Time Inspection

Summary of Technical Information in the Application. The applicant amended the One-Time Inspection Program subsequent to the issuance of the SER. The revisions are discussed below in the "Staff Evaluation" section.

Staff Evaluation. The staff's evaluation of the applicant's proposed One-Time Inspection Program is documented in SER Section 3.0.3.2.11. By letter dated June 23, 2014, the applicant revised the One-Time Inspection Program to delete the enhancement associated with inspections to detect cracking due to cyclic loading in the stainless steel makeup pump casings in the makeup and purification system. The staff's evaluation of this item is documented in SSER Section 3.3.2.2.4, item 3.

Operating Experience. The staff does not have any changes or updates to this section of the SER.

USAR Supplement. The staff does not have any changes or updates to this section of the SER.

Conclusion. The staff does not have any changes or updates to this section of the SER.

3.0.3.2.15 Structures Monitoring Program

Summary of Technical Information in the Application. The applicant amended the Structures Monitoring Program subsequent to the issuance of the SER. The revisions are discussed below in the "USAR Supplement" section.

Staff Evaluation. The staff does not have any changes or updates to this section of the SER.

Operating Experience. The staff does not have any changes or updates to this section of the SER.

USAR Supplement. LRA Section A.1.39 provides the USAR supplement for the Structures Monitoring Program.

In LRA Appendix A, "Updated Safety Analysis Report Supplement," the applicant provided the USAR supplement for the Structures Monitoring Program. The staff reviewed the USAR

supplement sections and noted that they conform to the recommended description for these types of programs, as described in SRP-LR Table 3.5-2. The staff also noted that the applicant committed (Commitment No. 20) to enhancing the Structures Monitoring Program prior to April 22, 2017. The staff also noted that, by letter dated June 23, 2014, the applicant amended the LRA to state that it completed Commitment No. 33, Phase 1, Actions 1 and 2; and Phase 2, Action 1.

The staff determined that the information in the USAR supplement is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. The staff does not have any changes or updates to this section of the SER.

3.0.3.2.16 Water Control Structures Inspection

Summary of Technical Information in the Application. The applicant amended the Water Control Structures Inspection Program subsequent to the issuance of the SER. The revisions are discussed below in the "USAR Supplement" section.

Staff Evaluation. The staff does not have any changes or updates to this section of the SER.

Operating Experience. The staff does not have any changes or updates to this section of the SER.

USAR Supplement. LRA Section A.1.40 provides the USAR supplement for the Water Control Structures Inspection Program. In LRA Appendix A, "Updated Safety Analysis Report Supplement," the applicant provided the USAR supplement for the Water Control Structures Inspection Program. The staff reviewed the USAR supplement sections and noted that they conform to the recommended description for these types of programs, as described in SRP-LR Table 3.5-2. The staff also noted that the applicant committed (Commitment No. 21) to enhancing the Water Control Structures Inspection Program prior to entering the period of extended operation. The staff also noted that, by letter dated June 29, 2015, the applicant amended the LRA to state that it completed Commitment No. 48.

The staff determined that the information in the USAR supplement is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. The staff does not have any changes or updates to this section of the SER.

3.0.3.3 AMPs Not Consistent with or Not Addressed in the GALL Report

3.0.3.3.6 PWR Reactor Vessel Internals Program

Summary of Technical Information in the Application. LRA Section B.2.32 includes the PWR Reactor Vessel Internals Program, which is defined as a new, plant-specific AMP for the LRA. The applicant stated that the PWR Reactor Vessel Internals Program will manage the following aging effects for the reactor vessel internals (RVI) components at Davis-Besse.

- changes in component dimensions due to void swelling or distortion
- cracking due to flaw initiation, flaw growth, SCC/intergranular attack (SCC/IGA), and irradiation-assisted SCC (IASCC)
- loss of preload due to stress relaxation

- reduction in fracture toughness due to radiation and thermal embrittlement
- loss of material due to wear

A full description of the AMP for managing aging in the RVI components is given in the Summary of Technical Information in the Application subsection of Section 3.0.3.3.6 in the SER. The RVI components at Davis-Besse were designed and fabricated by the Babcock and Wilcox Company (B&W, now owned by AREVA).

By letter dated April 21, 2015, the applicant amended the LRA and submitted the inspection plan for the RVI components in order to fulfill the conditions and criteria specified in LRA Commitment No. 15, which was included as a commitment in USAR Supplement Table A-1. In this commitment, the applicant committed to submitting the RVI inspection plan (RVIIP) to the NRC for review and approval by April 22, 2015. The RVIIP is documented in nonproprietary AREVA Report No. ANP-3920, Revision 1, "Reactor Vessel Internals Inspection Plan for Davis-Besse Nuclear Plant Unit No. 1 – Licensing Report," which was included as an enclosure in the letter of April 21, 2015.

The letter of April 21, 2015, also included the following AREVA reports that contain the background criteria for the RVIIP:

- Nonproprietary AREVA NP Licensing Report No. ANP-3285, Revision 0, "Confirmation of Stress Relief for the DB-1 [Davis-Besse] Core Support Structure Upper Flange Weld" (ADAMS Accession No. ML15113B134).
- Proprietary AREVA NP Licensing Report No. ANP-3359P, Revision 0, "Davis-Besse License Renewal Scope and MRP-189, Revision 1 Comparison," which is a controlled document that the staff has determined meets the NRC's withholding requirements in 10 CFR 2.390 and is being withheld from disclosure to members of the general public in accordance with those requirements.
- Nonproprietary AREVA NP Licensing Report No. ANP-3359NP, Revision 0, "Davis-Besse License Renewal Scope and MRP-189, Revision 1 Comparison" (ADAMS Accession No. ML15113B134), which is the nonproprietary version of Proprietary Report ANP-3359P, Revision 0, and which may be accessed by the general public.

The applicant stated that the RVI components at Davis-Besse were assessed according to their intended functions and the probability of inducing specific aging effects in the components.

The applicant stated that the components were then assessed for aging effect impacts on the intended functions of the components and then grouped into the categories for "primary," "expansion," "existing program," and "no additional measures" categories for the components, as defined in MRP-227-A. The applicant stated that the MRP-227-A protocols do not credit any "existing program" requirements for management of RVI components in B&W-designed plants. Therefore, the applicant clarified that the program does not implement any "existing program" inspections of the RVI components at Davis-Besse, other than the TS-defined surveillance requirements that apply to the vent valve assemblies and are implemented at Davis-Besse.

Evaluation. The NRC's evaluation of the PWR Reactor Vessel Internals Program (LRA AMP B.2.32) is documented in SER Section 3.0.3.3.6. The program is based on inspection and evaluation (I&E) guidelines that are provided in EPRI MRP Technical Report (TR) No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," dated January 2012 (ADAMS Accession Nos. ML12017A193 for the

transmittal letter to the NRC and ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195, and ML12017A199 for the final report), and EPRI MRP No. 1018292, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Revision 1)," which was issued in 2009.

This supplemental evaluation provides the staff's assessment of any changes to the previous program element's criteria for the PWR Reactor Vessel Internals Program that were proposed and included in the applicant's letter of April 21, 2015, including any impacts on the program elements induced by the I&E protocols defined in the RVIIP. This supplemental evaluation also includes a review of the applicant's bases for resolving those applicant/licensee action items (A/LAIs) that were issued in the NRC's supplemental evaluation for the MRP-227-A report and apply to the design of RVI components in B&W-designed reactors, which apply to the specific design of the RVI components at Davis-Besse.

Evaluation of AMP Program Element Nos. 1 – 4 and 7 – 9

The staff noted that, in the RVIIP, the applicant's definitions for "primary," "expansion," "existing programs," and "no additional measures" category components were consistent with those defined in MRP-227-A.[1] The staff confirmed that, consistent with this AMP and the RVIIP, the applicant will implement specific visual or UT inspections of the RVI components (as defined in MRP-227-A) in order to monitor and manage the following aging effects that are applicable to the RVI components: (a) loss of material, (b) cracking, (c) loss of fracture toughness, (d) loss of preload, and (e) changes in component dimensions/distortion.

The staff verified that the criteria for the "scope of program," "preventive actions," "parameters monitored," "detection of aging effects," "corrective actions," "confirmation process," and "administrative controls" program elements had not changed from the previous version of these elements in the FENOC letter of Sept. 16, 2011 (ADAMS Accession No. ML11264A059). Therefore, the staff concludes that the previous evaluations of "the scope of program," "preventative actions," "parameters monitored," "detection of aging effects," "corrective actions," "confirmation process," and "administrative controls" program elements remain valid, as previously documented in SER Section 3.0.3.3.6.

Evaluation of AMP Program Element No. 5, "Monitoring and Trending"

The staff determined that the applicant's "monitoring and trending" element remained unchanged in the letter of April 21, 2015, and that the RVIIP will implement all I&E criteria for "primary" category components defined in Table 4-1 of the MRP-227-A report and for "expansion" category components defined in Table 4-4 of the MRP-227-A report, with the exceptions of the deviations that the applicant had identified in the RVIIP for the vent valve assemblies and for nickel alloy weld locations in the lower grid and upper grid assemblies. Other than these changes, the staff reconfirmed that the previous "monitoring and trending" element had not changed from the previous version of this element in the FENOC letter of

[1] In MRP-227-A, the EPRI MRP did not apply any "existing program" criteria for aging management of B&W-designed RVI components. The applicant states that it will need to perform ASME Section XI-defined inspections of any RVI components that are defined as removable core support structure components (i.e., ASME Examination Category B-N-3 components), as required by 10 CFR 50.55a and the Table IWB-2500-1 requirements in the ASME Code Section XI. The applicant acknowledges that these requirements apply even if the components were placed into the "no additional measures" category in accordance with MRP-227-A methodology.

September 16, 2011. Therefore, the staff concludes that the previous evaluation of the "monitoring and trending" element, as previously documented in SER Section 3.0.3.3.6, remains valid, with the exception of the deviations identified for the vent valve assemblies and nickel alloy locations in the lower grid and upper grid assemblies. The staff evaluates these deviations in the staff's evaluation of A/LAI #2 in this section, subsection *Evaluation of the Applicant's Bases for Resolving Those A/LAIs That Apply to the RVI Components at Davis-Besse as a B&W-Designed Light Water Reactor Facility*.

Evaluation of AMP Program Element No. 6, "Acceptance Criteria"

The applicant's "acceptance criteria" for the AMP and RVIIP are provided in Table 5-1 of the MRP-227-A report, with the exception of the changes to acceptance criteria for evaluating those nickel alloy component locations in the lower grid and upper grid assemblies and for evaluating components in the vent valve assemblies. Other than these changes, the staff reconfirmed that the previous "acceptance criteria" element had not changed from the previous version of this element in the FENOC letter of September 16, 2011. Therefore, the staff concludes that the previous evaluation of the "acceptance criteria" element, as previously documented in SER Section 3.0.3.3.6, remains valid, with the exception of the acceptance criteria that relate to the identified deviations from MRP-227-A. The staff evaluates the changes to the acceptance criteria for nickel alloy locations in the lower grid and upper grid assemblies in the staff's evaluation of A/LAI #2 in this section, subsection *Evaluation of the Applicant's Bases for Resolving Those A/LAIs That Apply to the RVI Components at Davis-Besse as a B&W-Designed Light Water Reactor Facility*. The staff evaluates the basis for using TS 5.5.4 to establish appropriate acceptance criteria for the vent valve assemblies in Section 3.1.2.1.5 of this SSER and in the staff's evaluation of A/LAI #2 in this section, subsection *Evaluation of the Applicant's Bases for Resolving Those A/LAIs That Apply to the RVI Components at Davis-Besse as a B&W-Designed Light Water Reactor Facility*.

Evaluation of AMP Program Element No. 10, "Operating Experience"—Past RVI Inspection Summary and Updated "Operating Experience" Evaluation.

The staff noted that the applicant stated that the following inspections were performed during the current operating period:

- Vent valve tests and inspections are performed every 24 months (i.e., once every refueling outage [RFO]) in accordance with applicable requirements in TS Section 5.5.4. The applicant stated that past operating experience did not identify any relevant conditions other than the following: (a) raised/backed out jackscrew bushing observed during RFO 14 in 2006, (b) vent valve seating discoloration observed during RFO 16 in 2010, and (c) during RFO 17 in 2012, one vent valve exceeded the stay-open force acceptance criterion.

  The staff reviewed this operating experience against the stated intended function of the vent valve assemblies, as identified and discussed in nonproprietary TR No. ANP-3359NP, Revision 0. The staff noted that the intended function of B&W-designed vent valve assemblies is to open and achieve a fully open position during a postulated loss of coolant accident (LOCA) such that an adequate emergency core cooling system coolant flow through the reactor core is ensured during the design basis accident conditions. AREVA states that the opening of the valve prevents any back-pressure across the reactor core that could otherwise impede reactor flow into the core. Regarding the operating experience associated with the jackscrew bushing, the staff noted that the applicant's re-inspection of the jackscrew bushings during RFO 16

did not reveal any indications of raised/backed out bushings during the visual inspections that were performed in 2010. Thus, the staff noted that the applicant has not had any issues with jackscrew positions since 2010.

With regard to the discoloration that was observed in the vent valve seat in 2010, the staff noted that the discoloration observed in the vent valve seat did not affect the ability of the vent valve assembly to meet its intended function, as tested and confirmed during the TS-required surveillance test performed during RFO 16. Therefore, the staff concludes that this operating experience would not affect any of the elements in the PWR Reactor Vessel Internals Program or the ability of the AMP to manage aging during the period of extended operation.

Regarding the operating experience on the force needed to open the vent valves during the surveillance test, the applicant stated it performed several subsequent surveillance test cycles of the impacted vent valve assembly to confirm valve operability. The applicant stated that the subsequent surveillance tests did not result in any conditions where an excessive holding force would be needed to place the vent valve in a fully open valve disc position.

The staff noted that the vent valve surveillance test requirement in TS 5.5.4, Section c., limits the vertical force needed to place the valve configuration in a fully open position to a maximum value of 400 lbs of force. Thus, the staff noted that the subsequent surveillance tests of the impacted vent valve assembly confirmed that the valve would fully open when a vertical force of 400 lbs or less was applied to the impacted vent valve assembly.

Thus, based on this review, the staff noted that the applicant had appropriately reassessed the applicable operating experience for its impact on the intended function of the vent valve assemblies and that there was not any unresolved operating experience that could affect the intended function of vent valve assemblies during subsequent operations of the plant. Thus, the staff did not observe any vent valve assembly operating experience that, if left unresolved, could potentially result in a need for amending TS 5.5.4 under the TS change requirements of 10 CFR 54.22 or to adjusting the AMP's augmented inspection protocols for upper and lower vent valve retaining rings during the period of extended operation.

- <u>Ultrasonic Test (UT) inspections are of the upper core barrel (UCB), lower core barrel (LCB), lower thermal shield (LTS), upper thermal shield (UTS), flow distributor (FD), and surveillance specimen holder tube (SSHT) bolts</u>.

  The applicant stated that no recordable indications were observed during the UT inspections of these bolts in either 1984 or 1990. A more detailed operating experience evaluation of the inspections performed on the UCB, LCB, UTS, LTS, FD, and SSHT bolts and heat treatments for replaced Alloy X-750 bolts has been given in the staff's evaluation of the "operating experience" program element for the applicant's PWR Reactor Vessel Internals Program, as documented in SER Section 3.0.3.3.6. In the summary of that evaluation, the staff concluded that there have not been any relevant flaw or material degradation indications from past inspections of the UCB, LCB, FD, UTS, LTS, or SSHT bolts that would cause the structural integrity of the bolts to be called into question. Therefore, the staff concludes that the applicant's bases for performing UT inspections of the UCB, LCB, and FD bolts (as "primary category"

components for the program) and potentially performing UT inspections of the UTS, LTS, and SSHT bolts (as "expansion" category components for the program) remains acceptable without need for change under A/LAI #2.

- Core clamping measurements were taken of the differential height between the top of the plenum rib pads and the reactor vessel seating surface.

  The applicant stated that it performed this one-time measurement to satisfy the EPRI MRP physical measurement recommendations defined in Section 4.3.1 of the MRP-227-A, which recommends that a one-time physical measurement be performed of the differential distance from the top of the plenum rib pads in the plenum cover assembly to the reactor vessel seating surface.

  The staff also noted that the applicant's program will call for the applicant to implement VT-3 visual inspections of the plenum cover weldment rib pads, plenum cover support flange, and core support shield (CSS) to flange once every 10 years. The staff noted that: (a) the applicant's augmented physical measurement criteria associated with these inspections are consistent with EPRI MRP's physical measurement criteria for these components defined in Table 4-1 of MRP-227-A, and (b) the applicant has already completed its one-time activity to perform the physical measurement of this differential distance. In Table 5-1 of the MRP-227-A report, the EPRI MRP sets the following acceptance criteria on the performance of these differential measurements:

  > The measured differential height from the top of plenum rib pads to the vessel seating surface shall average less than 0.004 inches when compared to the as-built configuration.

  By letter dated June 5, 2015, the applicant amended the LRA to: (a) identify the number of physical measurements that were performed to measure the top of plenum rib pads-to-vessel seating surface distances, and (b) summarize the difference of the average value of these measurements from the height that was documented for this parameter in the original design documents. The applicant stated that a total of eight physical measurements were performed of this height parameter and that the average height differed by less than 0.004 inches from the top of the plenum rib pads to the reactor vessel seating surface height documented in the original design records. The applicant also stated that the physical measurement readings were uniform and did not reveal any evidence of wear in these reactor vessel internals components.

  The staff noted that the supplemental information provided by the applicant provides sufficient demonstration that there have not been any changes to the component configurations or degradation of the plenum rib pads that otherwise, if detected, could potentially change the criteria of the "primary" category inspections that will be applied to the components in accordance with MRP-227-A. Therefore, based on this review, the staff concludes that the applicant has demonstrated that the subsequent VT-3 visual examinations of the plenum rib pads during the period of extended operation are justified as proposed in MRP-227-A because: (a) the applicant has completed the physical measurements of the components in accordance with the guidelines in MRP-227-A, (b) the physical measurement results have not shown evidence of wear in the components or changes to the configurations of these components, (c) the physical measurement results support the conclusion that the visual inspection criteria in MRP-227-A for inspecting these components do not need to be adjusted, and

(d) consistent with the RVIIP, the applicant will implement the applicable visual inspections of the plenum rib pads during the period of extended operation, as recommended in MRP-227-A.

The operating experience related to one-time physical measurements of the plenum rib pads is closed, and the applicant has completed its commitment to perform a one-time set of physical measurements of the plenum rib pad-to-reactor seating surface height.

Evaluation of the Applicant's Bases for Resolving Those A/LAIs That Apply to the RVI Components at Davis-Besse as a B&W-Designed Light Water Reactor Facility.

In the staff's supplemental evaluation of the MRP-227-A report dated December 16, 2011 (ADAMS Accession No. ML11308A770), the staff included the following A/LAIs that an applicant implementing MRP-227-A report would need to address as part of its LRA. The B&W PWR applicable A/LAIs are A/LAI #s 1, 2, 4, 6, 7, and 8. A/LAIs #s 3 and 5 are applicable only to the design of Combustion Engineering or Westinghouse PWRs and do not apply to Davis-Besse.

The staff verified that the applicant had provided acceptable responses for resolving A/LAI #8, Subitems 1-4, in the applicant's letter of September 16, 2011 (ADAMS Accession No. ML11264A059). The applicant provided updated responses to A/LAI #s 1, 2, 4, 6, 7, and 8 in the letter of April 21, 2015.

The staff's evaluations and bases for accepting the applicant's responses to A/LAI #8, Subitems 1-4, are documented in SER Section 3.0.3.3.6. The evaluations that follow provide the staff's assessments of the applicant's bases for resolving the requests in A/LAI #s 1, 2, 4, 6, 7, and 8, Subitem 5, which were not resolved during the staff's initial review of the PWR Reactor Vessel Internals Program, as previously documented in SER Section 3.0.3.3.6.

Applicant's Response to Resolve A/LAI #1

In A/LAI #1, the staff asked the applicant to submit an evaluation demonstrating that the failure modes, effects and criticality analyses (FMECAs) and functionality analyses for B&W-designed internals in MRP-227-A are bounding for the design of RVI components at its facility or else to identify the process that will be used to identify differences in the design of their components from that assessed in MRP-227-A. The applicant stated that the design of the RVI components at Davis-Besse is bounded by the design and operating history assumptions for B&W-designed reactors, as defined in MRP-227-A, the FMECA report (MRP-190), and the B&W design functionality analysis (MRP-229, Revision 3).

The staff previously evaluated the basis for concluding that the analysis in MRP-277-A was bounding for the assessment of the RVI components at Davis-Besse in SER Section 3.0.3.3.6. In addition, the staff verified that, in the letter of April 21, 2015, the applicant identified those differences in the plant's RVI design from that assumed and assessed in the MRP-227-A report or its background reports, and that the applicant has resolved these differences as part of the applicant's basis for resolving the request in A/LAI #2. Thus, the staff concludes that the applicant has appropriately demonstrated that the analytical assumptions in MRP-227-A are bounding for the design of the RVI components at Davis-Besse, with the exception of those design deviations that the applicant has identified in its response to A/LAI #2. The staff evaluates the applicant's basis for reconciling these design differences in the staff's evaluation of the applicant's response to A/LAI #2, which follows in the next subsection.

Applicant's Response to Resolve A/LAI #2

In A/LAI #2, the staff asked the applicant to compare the design information for the RVI components at the plant to the corresponding information for the components in Tables 4-1 and 4-2 of MRP-189, Revision 1, and in Table 4-4 of MRP-191. Using the results of this comparison, the staff asked the applicant to identify any changes that would need to be made to the inspection criteria for the components in MRP-227-A.

The applicant stated that the methods in MRP-189, Revision 1, were used to evaluate the aging mechanisms and resultant aging effects on the components that could result in risk-significant degradation and to determine whether any necessary modifications of the program defined in MRP-227-A would be necessary. The applicant also stated that it performed a comparison of the RVI design at Davis-Besse to the component information and criteria in Tables 4-1 and 4-2 of MRP-189, Revision 1, and Table 4-4 of MRP-191. The applicant determined that all RVI components at Davis-Besse were in conformance with the aging assessments for B&W-designed RVI components in these reports, with the exception of the following components: (a) vent valve miscellaneous locking device parts, and (b) nickel alloy locations in the lower grid assembly.

In regard to the deviation for the vent valve assembly locking devices, the applicant stated that any aging effects that could potentially induce a failure of the original vent valve miscellaneous locking device parts are adequately managed by other existing plant programs. The applicant stated that this includes applicable programmatic controls in the American Society of Mechanical Engineers (ASME) Code Section XI and the applicable vent valve testing and inspection requirements for the vent valve assemblies that are included and implemented in accordance with the plant's TS.

The staff reviewed the TS requirements and the MRP-227-A "primary" category inspection criteria for the vent valve assemblies to determine whether the criteria would adequately manage the components in the vent valve assemblies, including any locking devices in the assemblies. The staff noted that the TS for Davis-Besse include TS #5.5.4, "Reactor Vessel Internals Vent Valve Program," which requires the applicant to implement the following testing and inspection activities of the vent valve assemblies on a 24-month surveillance frequency:

- Verify by visual inspection that the valve body and valve disc exhibit no abnormal degradation.
- Verify the valve is not stuck in an open position.
- Verify by manual actuation that the valve is fully open when a force less than or equal to 400 lbs is applied vertically upward.

The staff also noted that the PWR Reactor Vessel Internals Program and MRP-227-A "primary" category inspection protocols would call for the applicant to perform inspections of the vent valve top and bottom retaining rings on a 10-year inspection frequency. The staff noted that the collective set of visual inspections that would be implemented in accordance with the requirements of TS 5.5.4 and the PWR Reactor Vessel Internals Program for vent valve upper and lower retaining rings would not necessarily cover visual inspections of any locking devices that are part of the vent valve assemblies. Therefore, the staff reviewed additional information that was provided in Non-Proprietary AREVA Report No. ANP-3359NP, Revision 0, "Davis Besse Reactor Vessel Internals, License Renewal Scope and MRP-189, Revision 1 Comparison," to determine whether implementation of the

surveillance testing requirements in TS 5.5.4 could accomplish alternative aging management criteria for locking devices in the vent valve assemblies. These vent valve assemblies are designed to fully open in order to relieve pressure and ensure proper emergency coolant flow during a postulated LOCA event.

In Appendix A.1 of the SRP-LR, the NRC defines that performance monitoring programs test the ability of a structure or component to perform its intended function(s) and are one category of AMPs that may be used to manage the effects of aging. The staff noted that the surveillance requirements in TS 5.5.4 could be considered adequate performance monitoring program activities if the surveillance requirements were capable of indicating a potential vent valve assembly problem such that the applicant's Corrective Action Program would initiate an investigation of the issue, including any potential failure issues with the vent valve locking devices. The staff noted that TS 5.4.4 does require the applicant to perform a surveillance test requirement (i.e., performance monitoring requirement) to manually actuate the vent valve discs to the fully open position when a vertical force is applied to them. The staff determined that implementation of the applicable TS requirements would provide adequate performance monitoring activities for the vent valve assembly components (including applicable locking devices) because:

(a) Implementation of the surveillance requirements would be used to verify that the vent valve discs would be capable of fully opening when subjected to a flow-induced force, as would be needed during a postulated LOCA event.

(b) The applicant's Corrective Action Program would initiate an investigation of any vent valve assembly problem that had prevented the valve disc from fully opening at the required force during performance of the TS surveillance test requirement.

(c) Consistent with the program element criteria in SRP-LR Section A.1.2.3 for performance monitoring programs, this constitutes sufficient performance monitoring protocols for identifying any potential vent valve locking device issues that could potentially prevent the valve disc from achieving its intended fully open position requirement, which is necessary to be achieved during a postulated LOCA event.

Based on this review, the staff finds that the vent valve surveillance requirements in TS 5.5.4, when taken into account with the RVIIP's "primary component" inspection criteria for the vent valve upper and lower retaining rings, provide for adequate management of the vent valve assembly components (including applicable locking devices) because the staff has confirmed that the combination of condition monitoring activities (i.e., inspection activities) and surveillance testing activities (i.e., performance monitoring activities) meet the criteria for implementing performance monitoring and condition monitoring activities in Appendix A.1 of the SRP-LR and are acceptable to manage potential aging effects in the vent valve assembly components during the period of extended operation. The staff also finds that the MRP-227-A I&E protocols, as applied to the vent valve retaining ring components under the RVIIP, do not need to be further adjusted in accordance with the request in A/LAI #2.

In regard to the exceptions for the nickel alloy components in the lower grid and upper grid assemblies, the staff noted that the applicant proposed alternative "primary" and "expansion" categories for the components. Specifically, the staff noted that, in Appendix D and the Tables of TR No. ANP-3290, Revision 1, the applicant identified the following two deviations

from the generic design of the lower grid and upper grid assemblies assumed in MRP-227-A:

- Regarding inclusion of nickel alloy dowel-to-guide block welds as “primary” category components for the B&W-designed lower grid assemblies in MRP-227-A, the applicant stated the original dowel and dowel weld were removed from the Davis-Besse design during initial fabrication and replaced with a dowel, dowel cap, and dowel welds made from stainless steel materials. The applicant also stated that the guide blocks were welded directly to the lower grid forging and that the lower grid assembly design at Davis-Besse therefore no longer includes nickel alloy dowel-to-guide block welds.
- Regarding inclusion of nickel alloy dowel-to-upper grid fuel assembly support pad welds as “expansion” category components for the upper grid assembly in MRP-227-A, the applicant stated that this type of weld configuration does exist at Davis-Besse, even though it was previously thought not to exist in the plant design. The applicant also verified that the dowel-to-lower grid fuel assembly support pad welds in the upper grid assembly are made from nickel alloy weld materials.

The applicant stated that, since these two component items are linked in MRP-227-A, alternative “primary” and “expansion” category criteria need to be defined for the Davis-Besse program that will meet the same objective or level of conservatism for the original “primary” and “expansion” category links proposed for lower grid and upper grid assemblies in MRP-227-A. The applicant proposed the following alternative augmented inspection criteria to achieve this objective:

- elevation of the nickel alloy dowel-to-lower grid fuel assembly support pad welds in the lower grid assembly from “expansion” category components in MRP-227-A to “primary” category components for the RVIIP, with the components serving as the new lead indicators for nickel alloy locations in the lower grid and upper grid assemblies, noting that, for these components, the applicant stated that VT-3 type of visual inspections will be performed on the welds no later than two refueling outages from the beginning of the period of extended operation, with subsequent VT-3 inspections to be performed on a 10-year re-inspection basis
- identification of the nickel alloy dowel-to-upper grid fuel assembly support pad welds in the upper guide assembly as the appropriate “expansion” category component links for the “primary” category inspections that will be performed on the nickel alloy dowel-to-lower grid fuel assembly support pad welds in the lower grid assembly
- criteria for initiating inspections of the nickel alloy dowel-to-upper grid fuel assembly support pad welds in the upper grid assembly (as the “expansion” category components) if confirmed evidence of relevant conditions is noted at two or more dowel-to-lower grid fuel assembly support weld locations in the lower grid assembly, and for completing the expanded inspections of the dowel-to-upper grid fuel assembly support pad welds in the upper grid assembly by the completion of the subsequent refueling outage

The staff noted that the generic program for B&W-designed RVI components in MRP-227-A assumes that nickel alloy dowel-to-guide block welds are present in the plant’s lower grid assembly design, which is not the case for the modified design of the lower grid assembly at Davis-Besse.

Therefore, the staff noted that the proposed changes to the “primary” and “expansion” category component designations for nickel alloy welds in the lower grid and upper grid assemblies were appropriate because they accomplish the following general objectives:

(a) assign new alternative nickel alloy component locations (i.e., the nickel alloy dowel-to-lower grid fuel assembly support pad welds in the lower grid assembly) that will serve as the new leading “primary” category nickel alloy component locations in the lower grid and upper grid assemblies in order to account for the fact that the current design no longer includes any lower grid assembly dowel-to-guide block welds made from nickel alloy materials

(b) revise the current “expansion” category criteria for the nickel alloy dowel-to-upper grid fuel assembly support pad welds, such that the potential for performed expanded inspections of the components will now be linked to the results of the primary category” inspections that will be performed on the nickel alloy dowel-to- fuel assembly support pad welds in the lower grid

(c) establish appropriate “primary component” inspection result threshold criteria that will be used to initiate expanded inspections of the nickel alloy dowel-to-upper grid fuel assembly support pad welds in the upper grid assembly based on the results of the primary category inspections that will be performed on the nickel alloy dowel-to-upper grid fuel assembly support pad welds in the lower grid assembly during the period of extended operation

Based on this review, the staff finds that the changes in the “primary” and “expansion” category inspection criteria for these nickel alloy weld locations are acceptable because the staff has verified that the changes are consistent with: (a) the actual design of nickel alloy components in the lower grid and upper grid assemblies of the plant, and (b) the general intent of the criteria in MRP-227-A to establish lead “primary” and “expansion” category criteria for nickel alloy locations in B&W-designed lower grid and upper grid assemblies.

Based on this review, the staff finds that the applicant’s bases for resolving the requests in A/LAI #2 to be comprehensive and complete and A/LAI #2 is resolved with respect to the applicant’s PWR Reactor Vessel Internals Program and RVIIP.

Applicant’s Response to Resolve A/LAI #4

According the request in A/LAI #4, the applicant would need to implement enhanced visual inspections (EVT-1 inspections) of the upper flange weld in the core support structure no later than two refueling outages from the beginning of the period of extended operation and on a 10-year re-inspection basis, if the applicant could not provide evidence that the weld was stress relieved during initial weld fabrication. Therefore, in A/LAI #4, the staff asked applicants of B&W-designed PWRs to provide sufficient confirmation that the upper flange welds in the core support structures had been appropriately stress relieved as part of the processes that were used to fabricate the weld.

The applicant’s bases for responding to A/LAI #4 are given in Section 6.2.3.2 of TR No. ANP-3290, Revision 1, as supplemented with specific details in TR No. 3285, Revision 0, “Confirmation of Stress Relief for the DB-1 [Davis-Besse] Core Support Structure Upper Flange Weld.” The applicant stated that the original fabrication records for the RVI components confirm that the upper flange weld in the core support structure was

stress relieved as part of the processes that were used to fabricate the weld. By letter dated May 20, 2015, the applicant amended the RVIIP submittal and stated that the fabrication record confirming stress relief of the core support structure upper flange weld is given in Enclosure B of AREVA Record No. 51-9191898-000, "Reactor Vessel Internals Welds Stress Relief Records Search for the Operating 177-FA B&W Units," dated October 30, 2012.

The staff noted that TR No. ANP-3285, Revision 0, provides the details and summarizes the results of the applicant's fabrication record search. Specifically, the staff noted that the technical report indicates that the applicant did a search of the fabrication records for all double V-groove, double U-groove, and J-groove configured RVI welds listed in Table 4-2 of the MRP-189, Revision 1, report and that the fabrication records demonstrate that the welds were stress relieved during weld fabrication using a low frequency, vibratory stress equalization process. Therefore, the information in TR No. ANP-3285, Revision 0, as supplemented in the applicant's letter of May 20, 2015, provides sufficient demonstration that the upper flange weld in the core support structure was appropriately stress relieved immediately after the time of weld fabrication and provides an adequate basis for concluding that this weld will not need to be inspected under the augmented inspection protocols in MRP-227-A.

The staff did note that the upper flange weld in the core support structure may be defined by the applicant as a weld that is required to be inspected in accordance with Examination Category B-N-3 requirements of ASME Code Section XI for removable core support structure components. The staff noted that, if the core support structure upper flange weld is defined as an Examination Category B-N-3 component, the weld will be required to be inspected in accordance with Examination B-N-3 requirements specified in Table IWB-2500-1 of ASME Code Section XI and that the MRP-227-A protocols would not relieve the applicant of its obligations to perform these inspections in accordance the applicable ASME Code Section XI requirements, as invoked by 10 CFR 50.55a(g)(4). For these types of components, ASME Code Section XI requires all accessible surfaces of the removable core support structures to be inspected using ASME-defined VT-3 visual inspection methods once every 10-year ISI interval.

Based on this review, the staff concludes that the applicant has provided adequate demonstration that it does not need to implement augmented EVT-1 visual inspections of the core support structure upper flange weld under the I&E protocols in MRP-227-A because the applicant has provided sufficient demonstration that the core support structure upper flange weld has been appropriately stress relieved immediately after fabrication of the weld. However, the staff also concludes that, if this weld is part of the removable core support structure for the facility, the applicant will continue to be required to perform the appropriate ASME Section XI Examination Category B-N-3-defined VT-3 inspections of the weld (i.e., implement the appropriate ISI requirements) during each applicable 10-year ISI interval for the reactor unit. The request in A/LAI #4 is resolved with respect to the applicant's PWR Reactor Vessel Internals Program and RVIIP.

Applicant's Response to Resolve A/LAI #6

A/LAI #6 relates to the basis for managing aging in the following B&W-designed "expansion" category RVIs that are either inaccessible to inspection or for which adequate inspection methods have yet to be developed by the U.S. nuclear power generation industry: (a) axial and girth welds in the core barrel, (b) former plates, (c) external baffle-to-baffle bolts and

their locking devices, (d) core barrel-to-former bolts and their locking devices, and (e) internal baffle-to-baffle bolts. Specifically, the staff asked B&W applicants to justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation or by proposing a scheduled replacement of the components. As part of their applications to implement the approved version of MRP-227, the staff asked the applicants to provide a justification for the continued operability of each of the inaccessible or uninspectable components and, if necessary, provide, for NRC review and approval, their plan for replacing the components.

The staff noted that, in the letter of April 21, 2015, as supplemented in the letter of May 20, 2015, the applicant updated its basis for managing aging in these "expansion" category components and amended USAR Supplement Table A-1 to include Commitment No. 52, in which the applicant committed to submitting the plant-specific analysis or replacement schedule for these components to the staff for review and approval. The staff evaluates the applicant's new commitment and basis for resolving A/LAI #6 in the USAR Supplement subsection of this supplemental evaluation.

Applicant's Response to Resolve A/LAI #7

A/LAI #7 relates to the basis for managing the loss of fracture toughness due to thermal aging or neutron irradiation embrittlement in B&W-designed RVIs that are made from either cast austenitic stainless steel (CASS), martensitic stainless steel, or precipitation hardened stainless steel materials. In A/LAI #7, the staff recommended that applicants of B&W-designed reactors develop plant-specific analyses that will be applied to the incore monitoring instrumentation (IMI) guide tube assembly spiders and control rod guide tube (CRGT) spacer castings or to additional RVI components that may be fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel materials in order to demonstrate that the components will maintain their functionality during the period of extended operation. The staff stated that the plant-specific analysis for the components should be consistent with the plant's CLB and the need to maintain the functionality of the components under all conditions of operation for the CLB. The staff asked the applicant to provide the plant-specific analysis as part of its submittal to apply the MRP-227-A report to the CLB for its facilities (i.e., in this case, as part of the LRA).

The staff noted that, in the letter of April 21, 2015, as supplemented in the letter of May 20, 2015, the applicant updated its basis for managing neutron irradiation embrittlement and thermal aging embrittlement in RVI components made from CASS, martensitic stainless steel, or precipitation hardened stainless steel materials and amended USAR Supplement Table A-1 to include Commitment No. 53. In this commitment, the applicant committed to submitting a plant-specific analysis for RVI components made from these materials to the staff for review and approval. The staff evaluation of the applicant's new commitment and basis for resolving A/LAI #7 is documented in the USAR Supplement subsection of this supplemental evaluation.

Applicant's Response to Resolve A/LAI #8, Subitem 5

In A/LAI #8, Subitem 5, the staff addressed the need to identify all analyses that qualify as time-limited aging analyses (TLAAs) for RVI components. Specifically, under the requirements of 10 CFR 54.21(c)(1), the applicant is required to identify all analyses in the CLB that conform to the definition of a TLAA in 10 CFR 54.3(a) and to include an evaluation of the TLAAs in the LRA. The staff stated that MRP-227-A does not specifically address the

resolution of TLAAs that may apply to an applicant's RVI components. Therefore, in A/LAI #8, Subitem 5, the staff asked PWR applicants implementing the MRP-227-A report to evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that apply to their RVI components, and if so, to include and evaluate them in their LRAs in accordance the requirements of 10 CFR 54.21(c)(1)(i), (ii), or (iii).

In A/LAI #8, Subitem 5, the staff included additional request criteria for the CLBs that may include cumulative usage factor (CUF) TLAAs for specific RVI components in the plant designs. For those CLBs that did include these types of TLAAs, the staff stated that the applicant may use the PWR Reactor Vessel Internals Program as the basis for accepting the CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses will be periodically inspected for fatigue-induced cracking in the components during the period of extended operation. In this case, the staff stated that the periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, the staff stated that acceptance of the CUF TLAAs should be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with the requirement in 10 CFR 54.21(c)(1)(iii), using the applicant's program that corresponds to AMP X.M1, "Fatigue Monitoring," in the GALL Report, Revision 2. To satisfy the evaluation requirements of ASME Code Section III, Subsections NG-2160 and NG-3121, the staff stated that the existing fatigue CUF analysis evaluation should address how the effects of the reactor coolant system (RCS) water environment will be factored into the basis for accepting the TLAA, in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii).

The applicant stated that, for RVI components with CUF analyses, the TLAAs will be accepted in accordance with 10 CFR 54.21(c)(1)(iii) and that fatigue-induced cracking of the components will be adequately managed during the period of extended operation using the PWR Reactor Vessel Internals Program. The applicant further stated that fatigue-induced cracking will be managed by inspections defined for the components in MRP-227-A, which have been incorporated into the RVIIP.

The applicant stated that, in LRA Amendments 15 and 24, LRA Sections A.1.32 and B.2.32 were revised to state that the program includes management of the TLAA for reduction in fracture toughness of the RVI components and that the TLAA will be managed in accordance with the implementation of the MRP-227-A guidelines, as amended by the MRP-227-A SE, including all activities associated with the FENOC responses to plant-specific action items identified in Section 4.2 of the supplemental evaluation. The applicant stated that, in the SER for the LRA, the NRC staff determined that the PWR Reactor Vessel Internals Program is an acceptable basis for accepting the reduction of ductility TLAA, in accordance with 10 CFR 54.21(c)(1)(iii), and for managing changes in the ductile fracture toughness properties of the RVI components during the period of extended operation. However, the applicant stated that the applicable analysis in Appendix E of TR No. BAW-10008, Part 1, Revision 1, will require an update for the period of extended operation. Therefore, in Commitment No. 54, the applicant committed to submitting the updated evaluation under this A/LAI at least 6 months prior to entering into the period of extended operation (i.e., by October 22, 2016).

The applicant stated that two flow-induced vibration (FIV) analyses for RVI components were performed as part of the CLB and qualify as TLAAs for the LRA. The applicant stated that the FIV analysis of the RV internals and the incore instrument nozzles was dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and has been demonstrated to remain valid for the period of extended operation. The applicant stated that the FIV analysis

(i.e., high-cycle CUF analysis) for the reactor vessel surveillance capsule holder tubes was dispositioned in accordance with 10 CFR 54.21(c)(1)(ii) and has been projected to the end of the period of extended operation.

The staff verified that the LRA includes the following TLAAs for the RVI components at Davis-Besse: (a) reduction of ductile fracture toughness TLAA for the RVI components, as discussed and evaluated in LRA Section 4.2.7, (b) FIV endurance limit analyses for the RVI components and the incore instrumentation nozzles, as discussed and evaluated in LRA Section 4.3.2.2.2.2 (as amended), (c) the low-cycle fatigue CUF analyses for replaced UCB bolts, LCB bolts, and LTS bolts performed in accordance with a low-cycle metal fatigue method (i.e., low-cycle CUF analysis), as discussed and assessed in LRA Section 4.3.2.2.2.1, and (d) high-cycle FIV CUF analysis for the reactor vessel surveillance capsule specimen tubes, as discussed and evaluated in LRA Section 4.3.2.2.2.3 (as amended).

The staff verified that the evaluation of the FIV endurance limit analysis for the RVI components and incore instrumentation nozzles would remain unchanged, as documented in SER Section 4.3.2.2.2. For this analysis, the staff reconfirmed that the analysis remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The staff verified that the evaluation of the high-cycle CUF analysis for the replaced reactor vessel surveillance capsule specimen tubes would remain unchanged, as documented in SER Section 4.3.2.2.3. For this analysis, the staff reconfirmed that the analysis has been adequately projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The staff also verified that consideration of environmental effects associated with the reactor coolant does not need to be factored into the high-cycle CUF analysis because the analysis is based on an assessment of vibrations and not on an assessment of the impacts that cumulative design transient occurrences and environmental effects will have on low-cycle fatigue calculations.

For the reduction of ductile fracture toughness TLAA, the staff noted that the applicant changed its basis for accepting the TLAA in accordance with 10 CFR 54.21(c)(1)(iii) from that evaluated in SER Section 4.2.7. In this SER section, the staff previously accepted the applicant's basis for using the PWR Reactor Vessel Internals Program to accept the reduction of ductility TLAA in accordance with 10 CFR 54.21(c)(1)(iii) and to manage potential reductions in the ductile fracture toughness properties of the RVI components during the period of extended operation. However, in its letter of April 21, 2015, the applicant stated that it would update the reduction-of-ductility analysis in TR No. BAW-10008 to project the analysis to the end of the period of extended operation. The staff noted that the applicant committed to submitting the updated analysis to the NRC for review and approval at least 6 months prior to entering into the period of extended operation (i.e., by October 22, 2016). Thus, the staff determined that an updated evaluation of this TLAA was necessary in order to assess the applicant's updated basis for accepting the TLAA in accordance with 10 CFR 54.21(c)(1)(iii). The staff evaluates the change in the basis for dispositioning the reduction of ductility TLAA in accordance with 10 CFR 54.21(c)(1)(iii) in Section 4.2.7 of this SSER.

For the low-cycle CUF analyses that were performed for the replaced UCB bolts, LCB bolts, and LTS bolts, the applicant amended the TLAA evaluation for these components in a letter dated June 05, 2015. The applicant stated that a combination of the PWR Reactor Vessel Internals Program (LRA AMP B.2.32) and the Fatigue Monitoring Program (B.2.18) will be

used to ensure that fatigue-induced cracking will be adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii). The staff evaluates the basis for accepting these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) in Section 4.3.2.2.2 of this SSER.

The staff did not identify any other analyses in the CLB that would qualify as TLAAs for the RVI components, in accordance with the definition for TLAAs in 10 CFR 54.3(a). Based on this review, the staff finds that the applicant has identified those TLAAs that apply to the RVI components at Davis-Besse and appropriately addressed A/LAI #8, Subitem 5, through the applicant's identification of these TLAAs in the LRA. A/LAI #8, Subitem 5, is resolved with respect to the PWR Reactor Vessel Internals Program and the RVIIP for Davis-Besse.

USAR Supplement

The applicant's USAR supplement summary description for its PWR Reactor Vessel Internals Program is documented in Section A.1.32 of the LRA, which was amended in a letter dated March 9, 2012, and subsequently by letter May 20, 2015. This section supplements and updates the staff's previous evaluation of the USAR supplement summary description in LRA Section A.1.32, as documented in the USAR Supplement subsection of SER Section 3.0.3.3.6.

In the USAR Supplement subsection of SER Section 3.0.3.3.6, the staff stated that USAR Supplement Table A-1 in the LRA included LRA Commitment No. 15, which committed the applicant to submitting the inspection plan for the RVI components at Davis-Besse to the staff for review and approval no later than 2 years after issuance of the renewed operating license or 2 years prior to the beginning of the period of extended operation (i.e., April 22, 2015), whichever is earlier. The staff noted that the applicant's submittal of the inspection plan for the RVI components fulfills Commitment No. 15 because: (a) the applicant had submitted the inspection plan to the NRC for review and approval by April 22, 2015, (b) at the time of the submittal of the inspection plan, the staff had yet to issue a renewed operating license for the Davis-Besse facility, and (c) based on (a) and (b), the submittal of the RVI inspection plan on April 21, 2015, satisfies the time of submittal condition for submitting the inspection plan by April 22, 2015. Commitment No. 15 on the LRA is closed.

The staff also noted that the submittal of the RVI inspection plan added three new commitments to USAR Supplement Table A-1, which were included in the USAR Supplement relative to the applicant's activities for implementing of the PWR Reactor Vessel Internals during the period of extended operation:

(1) The applicant included Commitment No. 52 in order to address the NRC's action request in A/LAI #6. The applicant committed to the performance of analyses justifying the acceptability of the core barrel cylinder, including vertical and circumferential seam welds, former plates, external baffle-to-baffle bolts and their locking devices, core barrel-to-former bolts and their locking devices, and internal baffle-to-baffle bolts for continued operation through the period of extended operation or to the replacement of these component items. The applicant also committed to submitting either the detailed analyses or replacement schedule to the NRC for review and approval within 1 year of any degradation that is detected in the "primary" category components linked to these "expansion" category" components and exceeds the acceptance criteria for the applicable aging effects in the MRP-227-A report.

The staff noted that a commitment was appropriate to resolve this A/LAI because the applicant would not need to implement an assessment of one of the referenced "expansion" category components unless degradation was detected in the linked "primary" category component. The staff also noted that the applicant would only need to submit the analysis or replacement schedule of the referenced "expansion" category component to the NRC for review and approval if an applicable monitored aging effect in the linked "primary" category component was determined to exceed the limit for that aging effect in Table 5-1 of the MRP-227-A report. Therefore, based on this review, the staff finds that Commitment No. 52, as placed on USAR Table A-1, is acceptable because it is consistent with the guidelines approved in MRP-227-A and therefore resolves the request in A/LAI #6.

(2) The applicant included Commitment No. 53 in order to address the NRC's action request in A/LAI #7. The applicant stated that it is developing a plant-specific analysis to demonstrate that the IMI guide tube assembly spiders, CRGT spacer castings, and additional RVI component items made from CASS, martensitic stainless steel, or martensitic precipitation-hardened stainless steel materials (e.g., CSS vent valve top and bottom retaining rings) will maintain their functionality during the period of extended operation. The applicant stated that the analysis will: (a) consider the possible loss of fracture toughness in these component items due to thermal embrittlement and/or irradiation embrittlement and may also need to consider limitations on accessibility for inspection and the resolution or sensitivity of the inspection techniques, and (b) will be consistent with the CLB for Davis-Besse and the need to maintain the functionality of the component items being evaluated under all licensing basis conditions of operation. The applicant also committed to submitting the plant-specific analyses to the NRC for review and approval at least 1 year prior to the MRP-227-A implementation date for inspecting the applicable "primary" category component items.

The staff noted the action in Commitment No. 53 was consistent with the criteria in the MRP-227-A report and the NRC's criteria for resolving A/LAI #7, as defined in the December 16, 2011, supplemental evaluation for MRP-227-A. Therefore, based on this review, the staff finds that Commitment No. 53, as placed on USAR Table A-1, is acceptable because: (a) it is consistent with the guidelines approved in MRP-227-A and resolves the request in A/LAI #7, and (b) the applicant will submit the analysis to the NRC for review and approval at least 1 year prior to the MRP-227-A implementation date for inspecting the applicable "primary" category component items.

(3) The applicant included Commitment No. 54 in order to address the adequacy of the applicant's RVI reduction of ductility TLAA, as identified in response to A/LAI #8, Subitem 5. The applicant committed to the submittal of an evaluation for the period of extended operation that will assess the effect of irradiation on the mechanical properties and deformation limits of those RV internals that were evaluated for the current term in Appendix E of TR No. BAW-10008, Part 1, Revision 1. The applicant also committed to submitting the evaluation for NRC review and approval by October 22, 2016. The staff evaluates Commitment No. 54 in Section 4.2.7 of this SSER, which provides the staff's updated evaluation of the basis for accepting the reduction-of-ductility TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

Based on the above evaluation, the closure of LRA Commitment No. 15, and the inclusion of Commitment Nos. 52, 53, and 54, the staff finds that the information in the USAR supplement,

as amended, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its technical review of the applicant's PWR Reactor Vessel Internals Program, the staff concludes that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.3.7 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program

Summary of Technical Information in the Application. Subsequent to the submittal of the LRA, the staff issued LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation," which revised several GALL Report AMPs, including the guidance for AMP XI.M38. By letter dated February 19, 2014, the applicant provided the results of its review and changes to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program associated with the recommendations in LR-ISG-2012-02, Section B, "Representative Minimum Sample Size for Periodic Inspections" in GALL Report AMP XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components."

Staff Evaluation. To ensure that the GALL Report AMP XI.M38 inspections include a representative sample size, the staff issued LR-ISG-2012-02, which recommends a periodic minimum sample size, frequency, and inspection location for each material, environment, and aging effect combination for in-scope components. The revision included a provision to inspect 20 percent of each representative population of in-scope components, with a maximum sample size of 25 components, in each 10-year period during the period of extended operation.

By letter dated February 19, 2014, the applicant amended LRA Section B.2.41 to reflect the results of its review related to LR-ISG-2012-02, Section B. The applicant revised the program description of its Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program. The revised LRA Section B.2.41 program description now also states that, as a minimum, every 10 years, starting from the period of extended operation, a representative sample of 20 percent of each representative population (defined as components that have the same material, environment, and aging effect combination) or a maximum of 25 components per representative population will be inspected. The revision to the program description also states that, where practical, the inspection will include a representative sample of the system population and will focus on the bounding or lead components most susceptible to aging because of time in service and the severity of the operating conditions. The revised program description also states that this minimum sample size will not override the opportunistic basis of the AMP.

The applicant amended the "scope of program" element to also include periodic inspections of a representative sample size. The applicant also revised the "detection of aging effects" program element to make it consistent with its revised program descriptions as discussed above.

Finally, the applicant amended the "monitoring and trending" program element in its entirety. The revised element now states the following:

> Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program uses standardized monitoring and trending activities to track degradation. Deficiencies are documented using approved processes and procedures such

that results can be trended. Inspections are performed at frequencies identified in "Detection of Aging Effects" program element.

The staff finds the applicant's revisions to its Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program acceptable because the resulting sample size, inspection locations, frequency, and "monitoring and trending" program element are consistent with the recommendations provided in AMP XI.M38, as revised by LR-ISG-2012-02.

USAR Supplement. The staff reviewed the changes to LRA Section A.1.41 as amended by letter dated February 19, 2014, and noted that the applicant's program description is consistent with the recommended description in SRP-LR Table 3.0-1, as revised by LR-ISG-2012-02.

The staff determines that the information in the USAR supplement, as amended, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its review of the proposed changes to the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program, as amended by letter dated February 19, 2014, the staff determined that those program elements for which the applicant claimed consistency with AMP XI.M38, as revised by LR-ISG-2012-02, are consistent. The staff concludes that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.3.9 Shield Building Monitoring Program

Summary of Technical Information in the Application. Subsequent to the issuance of the SER dated September 3, 2013, the applicant amended the plant-specific AMP in LRA Section B.2.43, "Shield Building Monitoring Program," in response to the staff's requests for additional information to address the potential impact of more recent operating experience related to the shield building laminar cracking. As a result of indications of laminar crack propagation in the shield building identified by the applicant during baseline inspections conducted in August/September of 2013, the staff noted that additional information was needed to evaluate the potential impact of this operating experience on the shield building AMP. By letters dated July 3, 2014, and January 28, 2015, the applicant updated the "detection of aging effects," "monitoring and trending," "acceptance criteria," and "operating experience" program elements and the Shield Building Monitoring Program section of the USAR supplement to account for this recent plant-specific operating experience, as discussed in the staff evaluation below.

Staff Evaluation. The staff reviewed the updated program elements of the applicant's revised plant-specific Shield Building Monitoring Program against the acceptance criteria for the corresponding elements as stated in SRP-LR Section A.1.2.3. The staff's review focused on how the applicant's program manages aging effects of the shield building laminar cracking during the period of extended operation through the effective incorporation of these program elements. This review did not address the adequacy of the CLB, or the impacts of laminar cracking on the licensing basis that is ensured through processes outside of license renewal. Additional information on that review following the issuance of the SER can be found in the staff's Inspection Report (IR) 05000346/2013009, "Design and Licensing Basis of the Shield Building," dated May 12, 2014, IR 05000346/2014008, "Inspection of Apparent Cause Evaluation Efforts for Propagation of Laminar Cracking in Reinforced Concrete Shield Building

and Closure of Unresolved Item Involving Shield Building Laminar Cracking Licensing Basis," dated May 28, 2015, and other associated documents.

During its review, the staff asked followup RAIs and the applicant revised the Shield Building Monitoring Program. Unless otherwise noted, the staff's evaluation relates to the final version of the AMP found in the applicant's RAI response letter dated November 20, 2012, as amended by LRA Amendments 51 and 54 provided by letters dated July 3, 2014, and January 28, 2015, respectively, to account for the plant-specific operating experience of laminar crack propagation identified in August/September 2013. The followup RAIs and the responses provided after the issuance of the SER dated September 3, 2013, are summarized briefly below, followed by the staff's evaluation of the revised program elements as found in the final version of the AMP and clarified in the RAI responses.

- RAI B.2.43-4 issued by letter dated April 15, 2014: RAI B.2.43-4 requested information, with sufficient technical detail, of any modifications or enhancements that may be made to the Shield Building Monitoring Program or the Structures Monitoring Program to account for plant-specific operating experience related to: (1) discovery of previously unidentified cracks in the shield building core bore holes (also referred to as "core holes" or "bore holes") during the baseline inspection conducted in August/September 2013; and (2) broken/cracked rebar found, in February 2014, at several mechanical splice locations during hydro-demolition activities to create a temporary construction opening in the shield building to support the SG replacement.
- Response to RAI B.2.43-4 provided by letter dated July 3, 2014: The applicant provided information that several of the previously unidentified cracks resulted from propagation of existing laminar cracks by the "ice wedging" phenomenon. The applicant provided modifications to the Shield Building Monitoring Program with regard to increased number, location, and increased frequency, and a strategy for monitoring core holes for laminar cracks and its propagation. The response included LRA Amendment 51, which revised LRA Sections A.1.43, "Shield Building Monitoring Program," and B.2.43 to reflect the changes in the core hole monitoring sample, schedule, strategy, summary, and evaluation results incorporating the plant-specific operating experience that identified propagation of existing laminar cracks in 2013. The applicant further stated that the "rebar failure did not represent an aging management issue" because the broken or cracked rebar resulted from stress conditions induced by the physical process of hydro-demolition during the creation of the temporary construction opening, which was not an aging effect. By letter notification dated July 8, 2014, to the Atomic Safety and Licensing Board, the applicant submitted, as Enclosure 2, the apparent cause evaluation report entitled "Full Apparent Cause Evaluation Report - Shield Building Laminar Crack Propagation (Condition Report 2013-14097 dated 9/11/2013)" (hereafter referred to as FACE Report), for the 2013 identified crack propagation.
- RAIs B.2.43-5 (followup) and B.2.43-6 (followup) issued by letter dated September 29, 2014: RAI B.2.43-5 requested that the applicant discuss the technical rationale or criteria used to justify the selection of additional core holes, with identified crack propagation, for future inspections following discovery, and the number of subsequent inspections and/or time period for which they will continue to be inspected before possible removal from the sample. RAI B.2.43-6 requested that the applicant justify how the opportunistic inspection of rebar when exposed will adequately manage potential aging effects of corrosion on rebar located adjacent to laminar cracks, considering the plant-specific conditions of the shield building laminar cracking that is not passive, the presence of trapped water in the cracks, and potentially aggressive

environmental conditions; or provide information of modifications or enhancements that may be made to applicable AMPs to address the staff concern regarding implementation of opportunistic inspection of rebar to manage potential aging effects of corrosion.

- Responses to RAIs B.2.43-5 (followup) and B.2.43-6 (followup) provided by letter dated October 28, 2014: In response to RAI B.2.43-5, the applicant provided information that the selection of additional core holes for future examination of the laminar cracking will be based on the extent and direction of observed cracking, by monitoring the crack size, shape, and progression using 23 representative, strategically-selected core holes, 3 of which monitor the leading edge of identified propagation. The current extent of laminar cracking and its propagation is understood based on the examination of 80 core holes and impulse response testing conducted during 2011 through 2013. Of these, eight core holes were found in 2013 to have previously unidentified cracking. The applicant explained that 5 of these core holes, which were previously uncracked and into which the leading edge of an existing crack propagated, were included in the sample of 23, 3 of which were added to monitor the leading edge of propagation. A core hole may be removed from the sample only if it has been cracked circumferentially all around and can no longer bound cracking limits, in which case a new leading edge core hole may be installed. In the RAI B.2.43-6 response, the applicant explained that opportunistic inspections of rebar are adequate to manage aging effects of rebar corrosion because: (1) the environment within the concrete adjacent to the laminar crack is alkaline (as evidenced by chemical analysis of water in core holes indicating pH values greater than 10), (2) the mitigating nature of the applied exterior coating, and (3) the opportunistic inspection of the rebar in the construction opening made in the shield building in 2011, after over 30 years of operation, determined that the presence of cracking had not resulted in unacceptable rebar material loss or corrosion. The applicant also explained that since the grade elevation is more than 31 ft below the lowest elevation with laminar cracking, interaction between the potentially aggressive ground water and the laminar cracking condition is not credible.
- RAIs B.2.43-7 and B.2.43-8 issued by letter dated December 30, 2014: RAI B.2.43-7 requested that the applicant: (1) provide information of quantitative acceptance criteria for the shield building laminar cracking defined by bounding limits of cracking characterized in terms of crack width, crack planar limit, distribution, and/or any other appropriate parameters, against which the core hole inspection findings are compared and evaluated to determine (a) if the condition is bounded by and conforms to the design basis documentation referenced in the AMP, and (b) if corrective actions (e.g., re-evaluation of design basis documentation, repair) are needed to ensure that the structure and component intended functions are maintained consistent with all CLB design conditions during the period of extended operation, and (2) explain how the evaluation criteria hierarchy in Figure 5.1 of ACI 349.3R will be applied to the core hole inspection findings of laminar cracking to determine whether or not the condition is acceptable after evaluation. RAI B.2.43-8 requested that the applicant clarify and explain if core holes with worst-case observed laminar crack widths to-date are included in the representative sample of 23 core bore holes that will be monitored to determine if the condition is bounded by the design basis documentation, or to provide the bases for their exclusion.
- Responses to RAIs B.2.43-7 and B.2.43-8 provided by letter dated January 28, 2015: In response to RAI B.2.43-7, the applicant provided quantitative acceptance criteria in terms of crack width and planar limit against which core hole inspection findings will be evaluated to determine if corrective actions are necessary. The applicant also explained

how the evaluation criteria in Figure 5.1 of ACI 349.3R would be applied based on inspection findings of whether the cracks are passive and/or quantitative acceptance criteria are met. The response included LRA Amendment 54, which revised the "acceptance criteria" program element to include the quantitative acceptance criteria. In response to RAI B.2.43-8, the applicant clarified that currently 14 of the 23 core holes in the sample were cracked, and covered a range of crack widths including 3 core holes with the maximum observed crack widths of 0.01 inch to 0.013 inch.

The staff notes that the Shield Building Monitoring Program was revised in LRA Amendments 51 and 54, provided by letters dated July 3, 2014, and January 28, 2015, respectively, to account for the plant-specific operating experience of previously unidentified cracks and crack propagation discovered in 2013, which was addressed by RAIs B.2.43-4 through B.2.43-8 and their responses issued after the staff's SER dated September 3, 2013. By letter dated October 6, 2015, the Shield Building Monitoring Program was further revised in LRA Admendment No. 60 to incorporate the additional plant-specific operating experience of limited of limited crack propagation observed during inspections performed in 2015, and as a result of discussions held during the September 23, 2015, Advisory Committee on Reactor Safeguards Subcommittee meeting on the Davis-Besse LRA. These LRA amendments revised the "detection of aging effects," "monitoring and trending," "acceptance criteria," and "operating experience" program elements, and the USAR supplement of the Shield Building Monitoring Program.

Based on the evaluation of the program elements below, the staff finds the November 20, 2012 AMP, as amended by LRA Amendment 51, dated July 3, 2014, LRA Amendment 54, dated January 28, 2015, and LRA Amendment No. 60 dated October 6, 2015, acceptable. The applicant has adequately addressed the staff's concerns described in the RAIs to account for the 2013 operating experience related to the propagation of laminar cracking. The staff also finds that the applicant has sdequately addressed and incorporated the crack propagation observed in 2015 into the AMP by LRA Admendment No. 60.

Scope of the Program. There are no changes or updates to this section of the SER.

Preventive Actions. There are no changes or updates to this section of the SER.

Parameters Monitored or Inspected. There are no changes or updates to this section of the SER.

Detection of Aging Effects. LRA Section B.2.43, as amended by letter, dated July 3, 2014, in response to RAI B.2.43-4 (evaluated in this and the following sections), states that visual inspections will be performed on a representative sample of the shield building wall structural subcomponents by inspection of internal surfaces of core bores. The applicant stated that the representative sample size currently consists of a minimum of 23 (20 in the previous November 20, 2012, version of the AMP) core bore locations to include 8 of the 10 flute shoulders with a higher prevalence of event-driven laminar cracking. The locations also include four bores above the 780 ft elevation (within the upper 20 ft of the structure where cracking was also identified outside of the flute shoulders) and one at each main steam line penetration. The applicant stated that locations of the inspections will be chosen from the core bores that have been installed in the shield building wall, including new core bores installed as required to identify changes in the limits of cracking in areas with previously identified crack propagation. The applicant also stated that approximately 10 core bore inspection locations in the current sample are uncracked but strategically located adjacent to areas of known cracking; thereby providing the ability to monitor crack propagation. The applicant further stated that past

evidence of crack propagation will be additionally considered in choosing future inspection locations. The program element also includes provisions to supplement visual inspections with other established nondestructive examination (NDE) techniques and testing, as necessary. The applicant also stated that the inspections would occur annually prior to the period of extended operation (i.e., in 2015 and 2016), and in 2017 and 2018 after entering the period of extended operation. If no aging effects (defined as no discernable change in crack width or general appearance of crack or confirmation that no visible cracks have developed in previously uncracked core holes) are identified, then the frequency of visual inspections may be progressively changed to at least once every 2 years through 2026, and may be changed to at least once every 4 years thereafter if the 2-year frequency does not detect any degradation. Changes to the inspection schedule and parameters monitored will be evaluated if aging degradation is detected.

The staff reviewed the applicant's "detection of aging effects" program element against the criteria in SRP-LR Section A.1.2.3.4, which states that this element should address how the program would be capable of detecting or identifying the occurrence of age-related degradation prior to the loss of function. This element should also discuss "when" and "how" data will be collected for the program. This element should also justify the sample size of an inspection program based on sampling, and should justify the inspection frequency and method.

As described in detail in the "operating experience" program element, by letter dated April 15, 2014, the staff issued RAI B.2.43-4. This RAI in part requested that the applicant describe and justify modifications or enhancements, if any, that may be potentially made to the AMPs credited for the shield building for license renewal, to account for the plant-specific operating experience related to laminar cracking propagation identified in August/September 2013.

In its response dated July 3, 2014, to this part of RAI B.2.43-4, related to the August/September 2013 discovery of previously unidentified cracks, the applicant characterized the "newly" identified cracks as propagation of laminar cracking by the phenomenon of "ice-wedging". The applicant further stated that the minimum representative sample of monitoring core bores in the Shield Building Monitoring Program is increased from 20 to 23 as a result of this plant-specific operating experience. The applicant also stated that three monitoring bores will be used to aid in identifying changes in the limits of cracking in areas with previously identified crack propagation and that new core bores may be installed, if needed, during each inspection cycle in order to bound crack limits. The applicant further stated that the frequency of internal visual inspection for the 23 monitoring bores is changed to annual inspections for a minimum of 4 years (2015-2018). The applicant stated that, following acceptable results of the 1-year interval inspections, the interval will be changed to a 2-year interval in 2019, and a maximum 4-year interval after 2026. Further, the applicant stated that these inspection intervals will be evaluated for effectiveness and modifications to the Shield Building Monitoring Program will be determined in the Corrective Action Program should there be an identified significant change in cause, rate of crack growth, or a condition that is not bounded by the design basis documentation. The applicant revised the USAR Supplement in LRA Section A.1.43 to address the change in the core monitoring schedule. The applicant revised the "detection of aging effects," "monitoring and trending," "acceptance criteria," and "operating experience" program elements of the Shield Building Monitoring Program to incorporate changes due to the plant-specific operating experience of crack propagation. The staff notes that these changes have been reflected and evaluated in the staff evaluation of respective program elements in this and following sections. The staff evaluation of the part of the response to RAI B.2.43-4 that

relates to operating experience aspects is provided further below in the “operating experience” program element section.

The staff reviewed the applicant’s response dated July 3, 2014, to part of RAI B.2.43-4 regarding changes to the AMP resulting from its review of the 2013 plant-specific operating experience of laminar crack propagation and found portions of it acceptable. The staff finds the response with regard to the minimum representative sample size (minimum of 23) of core holes examined in future inspections and the strategy for distribution of core holes locations inspected acceptable because they cover areas of high prevalence of laminar cracking. In addition, consideration of past evidence of crack propagation in choosing inspection locations and provision for the addition of new core holes for inspection are incorporated in the program. The staff also finds the proposed inspection intervals acceptable because they are conservatively biased and increased progressively only if no aging effects are identified. The applicant also provided an adequate LRA update to reflect these considerations. However, the staff found that the basis or criteria used for selecting 3 out of the 8 core bore holes with discernable change in laminar cracking conditions in the sample size of 23 core holes for subsequent consecutive inspections of the shield building wall, and the number and/or time period of subsequent consecutive inspections in which they will be included in the representative sample for inspection after discovery, was not fully or clearly described. By letter dated September 29, 2014, the staff issued followup RAI B.2.43-5 (followup) requesting the applicant to provide: (1) additional discussion and detail of the technical rationale or criteria used to justify the selection of additional core holes with identified crack propagation for future inspections after discovery of a change in crack conditions, and (2) the minimum number and/or time period of subsequent consecutive inspections in which they will be inspected after discovery of a change in crack conditions before they may be removed from the representative sample.

In its response by letter dated October 28, 2014, to the first part of RAI B.2.43-5, the applicant stated that the selection of additional core bores for future examination of the laminar cracking identified within the shield building will be based on the extent and direction of propagation of laminar cracking in the structure. The applicant also stated that based on its current knowledge of the size and extent of the original pre-existing sub-surface crack, and the cause and rate of crack propagation due to ice-wedging, it is monitoring the crack size, shape and progression by the use of 23 strategically selected core bores, 3 of which were chosen to monitor the leading edge of crack propagation; these 23 bores are representative of the remaining cracked areas. The applicant stated that if crack planar propagation passes completely through one of the three leading edge core bores, then the need to add a new leading edge core bore adjacent to that bore will be evaluated to maintain at least three leading edge monitoring bores.

The applicant explained that it determined through causal analysis that the crack propagation, identified in 2013 inspections in the architectural flute shoulder areas, is a result of “ice wedging,” which requires a previously existing crack, the presence of water, and a freezing cycle. The applicant further explained that the extent and location of cracking for the entire shield building was mapped using impulse response testing and validated by over 80 core bores examined in 2011-12. During the 2013 monitoring of core bores for the laminar cracking condition, “new” cracking was identified in eight bores. The applicant stated that it conducted informational impulse response mapping at five locations (covering approximately 2,200 square feet) for overall perimeter identification and examined all 80 existing bore locations, as part of the extent of condition investigation. The applicant further stated that the 2013 discovery of changes in the condition of cracks in the structure provides a validation that the visual inspection method of bore monitoring is an effective means of identifying small changes in laminar cracking in the shield building. The features of the eight bores with observed changes

in laminar cracking were assessed and divided into two categories: Category 1, consisting of five bores where the leading edge of an existing crack propagated into previously uncracked concrete (i.e., perimeter expansion); and Category 2, consisting of three bores where changes (e.g., a crack offshoot developing in the bore) were identified in a previously cracked bore with no planar propagation.

The applicant explained that of the 5 Category 1 bores, 2 bores were already within the population of the originally proposed 20 bores selected for long-term monitoring under the Shield Building Monitoring Program. Therefore, the remaining 3 bores are added to the program for monitoring, to form the representative sample of 23 bores within the area of cracking, such that changes in planar limits of the crack are monitored and bounded. The applicant noted that two of the five Category 1 bores intersect the same leading edge (i.e., located approximately at the same elevation and horizontally offset from each other) and, therefore, the other three bores are used to monitor the leading edge of crack propagation. Further, the applicant clarified that Category 2 bores were not incorporated into the monitoring program since they do not provide information related to identifying the leading edge (i.e., perimeter) of the crack.

In its response dated October 28, 2014, to the second part of RAI B.2.43-5, the applicant stated that there is no minimum number or time period of subsequent consecutive inspections for which core bore holes with identified crack propagation will be inspected following discovery before they may be removed from the representative sample because it intends to perform inspections of the 23 monitoring bores throughout the period of extended operation. The applicant further explained that a bore hole added to the inspection scope for the purposes of monitoring laminar cracking limits may be removed from scope if it has been cracked 360 degrees around and can no longer bound cracking limits. However, prior to removal from the scope of inspection, the need to install a new leading edge core bore adjacent to the bore being removed will be evaluated in order to maintain the population of at least three leading edge core bores for monitoring crack propagation. The applicant further clarified that the progressively increasing inspection intervals proposed in the program will be evaluated for effectiveness by the Shield Building Monitoring Program. Should there be an identified change to the cause of the condition, significant change to the rate of crack growth, or a condition adverse to the bounding nature of the design basis documentation, the applicant stated that it would evaluate and determine modifications to the Shield Building Monitoring Program using its Corrective Action Program. The applicant explained that bore holes are also examined for changes in crack width. Therefore, even if a bore hole is cracked 360 degrees around and is no longer able to define planar limits, any previously identified changes to width will be entered into the Corrective Action Program, and that information will be used in the decision-making process to determine whether changes are required to the inspection schedule (e.g., increase inspection frequency) or parameters monitored (e.g., increase the number of core bores monitored). In a telephone conference call held on February 20, 2015, the applicant clarified that the “leading edge” of planar laminar cracks is where the inspection would observe a plane extend or expand compared to previous core bore inspection results. The applicant further clarified that leading edge monitoring is achieved through inspection of bores, and impulse response mapping if required. The applicant also clarified that a representative core bore provides information regarding crack width and/or the planar limits of cracking.

The staff noted that the applicant’s criteria for the selection of additional core bores for future examination of the laminar cracking identified within the shield building were based on the extent and direction of propagation of laminar cracking in the structure, which is determined based on expanded core hole inspections, and supplemental impulse response testing, if

required. The staff noted that the applicant selected additional core bores for future examinations to monitor crack propagation by reviewing the cracking features of all core bores with observed crack propagation in the last inspection cycle, and identifying those that uniquely define the leading edge (planar perimeter limit) of crack propagation, which resulted in the addition of three core bores to define the leading edge or planar limits of observed crack propagation. The staff also noted that according to the applicant's criteria, it intends to maintain at least three leading edge core bores in the representative sample. The staff finds the applicant's response to the first part of RAI B.2.43-5 acceptable because the applicant explained its criteria for selecting additional bores to monitor crack propagation based on a review of inspection findings from the last inspection cycle and identifying those core holes with identified crack propagation that uniquely define the leading-edge limits of crack propagation, which the staff determines to be a reasonable approach for monitoring the planar limits of crack propagation. The staff finds the applicant's response to the second part of RAI B.2.43-5 acceptable because it: (1) clarified that there is no specified number or time period of inspections criteria for core holes with identified crack propagation for subsequent examinations following discovery; and (2) explained that results of an engineering evaluation of the inspection findings, with regard to change in crack width, planar limits, and cause and rate of crack growth, in its Corrective Action Program will be used as the basis to remove and add core holes to the representative inspection sample or to make other changes to the monitoring program for future examinations, which the staff determines to be a reasonable approach to making changes to the AMP, as necessary, to ensure adequate management of crack propagation. The staff's concerns described in RAI B.2.43-5 are resolved.

The staff noted that the response dated July 3, 2014, to RAI B.2.43-4 did not identify changes to the Shield Building Monitoring Program with regard to monitoring the rebar for corrosion but documented operating experience of the presence of water within the pre-existing cracks that under freezing temperatures may cause the cracks to propagate. Further, the staff noted that the Shield Building Monitoring Program as submitted by letter dated November 20, 2012, proposed to monitor rebar for corrosion by visual inspection, only on an opportunistic basis, when exposed for some undefined reason. The presence of water and air trapped within the existing laminar cracks of the coated shield building wall increases the potential for corrosion of the adjacent rebar layers. Further, the groundwater chemistry at the Davis-Besse site is considered to be aggressive [i.e., chlorides = 2,870 ppm (max) and sulfates = 1,700 ppm (max)] which may also be indicative that the shield building is or has been exposed to potentially aggressive (i.e., high chloride content) air-outdoor environment. Given the above plant-specific conditions and operating experience of existing laminar cracking that may propagate, the presence of trapped water in the cracks, and potentially aggressive environmental conditions, the staff needed additional technical justification and basis regarding the AMP's implementation of opportunistic inspections to monitor aging effects in the rebar located near the laminar cracking. By letter dated September 29, 2014, the staff issued RAI B.2.43-6 (followup) requesting the applicant to explain, with sufficient technical detail and basis, how the opportunistic inspection of rebar, when exposed, will adequately manage the potential aging effects of corrosion for rebar layers located near laminar cracking; or to provide any modifications or enhancements that will be made to the Shield Building Monitoring Program to address the staff's concern regarding the implementation of opportunistic inspection of rebar when exposed to adequately manage potential aging effects of corrosion for rebar layers located near laminar cracking.

In its response to RAI B.2.43-6 by letter dated October 28, 2014, the applicant stated that the opportunistic inspection of rebar when exposed will adequately manage the potential aging

effects of corrosion for rebar layers located near laminar cracking, and no modifications or enhancements to the AMPs are necessary, for the reasons described below:

- FENOC visually inspected rebar in areas of laminar cracking at the construction opening during October/November, 2011, after over 30 years of operation, and determined that the presence of cracking itself has not resulted in unacceptable rebar material loss or corrosion.
- The presence of potentially aggressive environmental conditions (groundwater) is not a condition associated with the laminar cracking because impulse response mapping completed on exterior portions of the shield building during 2012 did not identify laminar cracking below elevation 615 ft, which is more than 31 ft above grade elevation at the site. Therefore, postulated scenarios of interaction between groundwater and the laminar cracking condition are not considered credible.
- As a corrective action related to the discovery of laminar cracking, an exterior coating was applied to the shield building during 2012, which is a means to limit the availability of oxygen and moisture required to sustain a corrosive environment (ACI Report 222R-85, "Corrosion of Metals in Concrete"). Therefore, despite the identification of water within the concrete, the introduction of additional oxygen and moisture into the shield building concrete has been limited such that the postulated corrosion rate of rebar is expected to remain minimal, with a decreasing trend.
- Chemical analysis of the water found in the core bores, conducted as part of the causal analysis of the observed laminar crack propagation, concluded that the water constituents were typical of water that was in contact with the concrete for a period of time, and exhibited high pH values averaging greater than 10. Therefore, the water itself with salt and high pH is not conducive to generating corrosion in the rebar.

The applicant summarized that it has elected to conduct opportunistic inspections of the rebar based on the supporting evidence that the alkali environment within the concrete is inhibiting corrosion and the mitigating nature of the coating. The applicant also explained that rebar corrosion would result in visual indications such as staining, cracking, or spalling of the exterior of the shield building structure or in core bores that are located near rebar, which are indications monitored for under the Structures Monitoring Program to adequately manage the aging effects of potential rebar corrosion.

The staff reviewed the applicant's response to RAI B.2.43-6 and noted that the protective exterior coating applied to the shield building provides a means to limit the ingress and availability of oxygen and moisture or other deleterious elements required to sustain a corrosive environment. The staff also noted that the high pH obtained from the chemical analysis of water found in the core holes indicate an alkaline environment of the concrete and water around the rebar adjacent to the laminar cracks that is not conducive to corrosion. The staff noted from the applicant's 2011-2012 root cause analysis for laminar cracking and the 2013-2014 apparent cause evaluation (FACE Report) for the laminar crack propagation that the majority of laminar cracking discovered in 2011-2012 and all of the 2013 observed crack propagation occurred in the flute shoulder regions of the shield building, which has a significantly larger cover depth to the laminar crack adjacent to the outer hoop reinforcement. Further, the FACE Report indicates that the laminar cracks are hairline tight with the maximum observed crack width being in the range of 0.010 to 0.013 inch, with the majority being smaller than 0.005 inch. Additionally, although the groundwater at the site is considered aggressive, the lowest elevation of observed laminar cracking is located over 31 ft above the grade elevation, which does not establish a nexus between the groundwater and laminar cracking. These factors and the fact that

opportunistic inspection of the rebar in 2011 in the area of the construction opening, after over 30 years of plant operation and after the initiation of laminar cracking during the 1978 blizzard, identified minimal to no rebar corrosion, which indicates that the presence of hairline laminar cracking, in itself, does not increase the risk for potential rebar corrosion. Additionally, visual indications of rebar corrosion, such as surface staining, cracking, spalls, and indications of corrosion products in the core holes are monitored under the Structures Monitoring Program. Therefore, the staff finds that the opportunistic inspection of rebar is an adequate method to manage potential rebar corrosion adjacent to laminar cracking in the shield building. The staff thus finds the applicant's response acceptable because the applicant provided justification to reasonably support the adequacy of opportunistic visual inspections to manage the aging effects of rebar corrosion. The staff's concern described in RAI B.2.43-6 is resolved.

The staff observed, in its RAI B.2.43-7 dated December 30, 2014, and described in the "acceptance criteria" program element evaluation section further below, that the laminar crack width is a critical parameter that could affect the bond strength and the capacity of the shield building's outside hoop rebar, adjacent to the laminar cracking, to perform its intended function. Although the "detection of aging effects" program element states that the representative sample of core holes are examined for discernable changes in general appearance and crack width of existing laminar cracks; and for indications of new cracking, it was not clear to the staff if the representative sample of 23 core bore holes to be inspected includes and tracks core bore holes with worst case observed crack widths. By letter dated December 30, 2014, the staff issued RAI B.2.43-8 requesting the applicant to clarify whether the core bore holes with the worst case observed laminar crack widths to date are included in the representative sample of 23 core bore holes that will be monitored during the period of extended operation to determine if the condition is bounded by the design basis documentation referenced in the AMP. The applicant was also requested to provide the number of core holes included in the sample and the basis for such number or to provide a basis for their exclusion.

In its response to RAI B.2.43-8, dated January 28, 2014, the applicant stated that the core hole with the worst case observed laminar crack width to date, which is 0.013 inch, is included in the representative sample of the 23 core bore holes that will be monitored in the period of extended operation. The applicant also stated that the current monitoring program includes inspection of 14 bore locations with identified laminar cracking with a range of crack width sizes, as indicated in SER Table 3.0.3.3.9-1 below.

Table 3.0.3.3.9-1 Crack Widths of Cracked Core Bores in Inspection Sample

Approximate Crack Width, Inch	Number of Core Bores in Sample
0.005 inch and lesser	8
0.006 inch to 0.009 inch	3
0.010 inch and greater	3

The applicant also stated that crack width was a parameter inherent to the reinforcement bond and splice evaluations completed at the University of Kansas and Purdue University, and is therefore treated as a limiting parameter and monitored by the AMP for a "discernable change" in general appearance and width. The applicant further stated that crack width is recorded on each bore inspection form retained under its records management program and is added to site drawings to facilitate trending. The applicant clarified that the selection of the bores with cracking in the sample was based on the prevalence of cracking as discussed in the responses

to RAIs B.2.43-2 and B.2.43-2a. As noted previously, the applicant also clarified in a telephone conference call held on February 20, 2015, that a representative core bore provides information regarding crack width and/or the planar limits of cracking.

The staff noted from the applicant's response that its representative sample of 23 core bores inspected includes 14 core holes with a range of laminar crack widths, including 3 with the maximum observed crack widths ranging from 0.010 inch to 0.013 inch. The basis for selection of these core holes was to cover the spectrum of locations with high prevalence of laminar cracking. The staff also noted that each representative bore in the inspection sample provides information regarding crack width (helps monitor maximum crack width) and/or the planar limits of cracking (i.e., helps monitor planar propagation). The staff finds the applicant's response acceptable because the representative inspection sample of 23 core holes consists of core holes that define maximum observed crack widths as well as planar propagation limits and, therefore, includes appropriate monitoring and trending of the limiting "crack width" parameter to effectively detect aging effects of potential propagation on the bond capacity of the rebar adjacent to the laminar cracking. The staff's concerns described in RAI B.2.43-8 are resolved.

In LRA Amendment No. 60 dated October 6, 2015, the applicant further revised the "detection of aging effects" program element to state that the representative inspection sample of 23 core holes was increased to 28, and that the current distribution of core bores included 9 (previously 8) of the 10 flute shoulders with a high prevalence of laminar cracking. The program element was also revised to state that visual inspections will be supplemented by impulse response (IR) mapping and/or other established NDE methods. The applicant stated that IR mapping will be performed on a total of eight 100 square foot grid areas of the exposed exterior of the shield building; inspections will be performed in four areas in 2016 and four areas in 2018. For the IR mapping to be conducted in the 2016 and 2018 inspections, two grids will be in areas known to have cracks and two grids in areas not currently known to contain laminar cracks, with all areas being away from existing core bores. These IR mapping inspections will be performed with the objective to identify changes in the limits of cracking outside the areas inspected by visual examination of core bores. The applicant stated that IR mapping will also be used to supplement visual inspections in a minimum 100 square foot area near leading edge core bores with identified crack propagation to provide an indication of the extent of cracking propagation for condition monitoring. The applicant also stated that IR mapping will be performed in accordance with vendor procedures and if crack growth is identified the condition will be entered into FENOC's Corrective Action Program.

The staff notes that the core bore inspection interval begins with inspections performed once every year and that the inspection interval may be progressively incremented, only if the cracks remains passive, to 2 years and a maximum of 4 years. The staff finds the interval of inspection acceptable because (1) it starts out conservatively at 1 year to include a complete winter cycle when propagation is likely, (2) it may be incremented only if aging effects are not identified, and (3) the maximum inspection interval of 4 years is more stringent than the GALL Report's recommended inspection frequency of 5 years for exterior concrete surfaces. The staff also notes that the program supplements visual inspections with IR mapping and/or other NDE methods and testing. As discussed in the "parameters monitored or inspected" program element, the staff finds the visual inspection method acceptable because visual inspection of core bores is a definitive method for detecting and characterizing changes in the laminar cracks. Further, the staff finds the use of IR mapping as a supplement to visual inspections of core bores acceptable because it is an effective NDE method to detect propagation or new indications of laminar cracking and to update the planar extent of cracking condition in the shield building. The staff further notes that the identification of laminar crack propagation in

August/September 2013 and 2015 by visual inspection of core holes indicate that the visual method of inspection of representative core holes is effective in detecting aging effects prior to loss of intended function.

Based on its review of the program as amended by responses to RAIs B.2.43-4 through B.2.43-8 in letters dated July 3, 2014, October 28, 2014, and January 28, 2015, and LRA Amendment No. 60 dated October 6, 2015, the staff confirmed that the "detection of aging effects" program element satisfies the criterion defined in SRP-LR Section A.1.2.3.4 and, therefore, the staff finds it acceptable.

Monitoring and Trending. LRA Section B.2.43, as amended by letter dated July 3, 2014, states that the Shield Building Monitoring Program will include a baseline inspection, performed prior to the period of extended operation and followed by periodic inspections of a representative and strategically located sample of a minimum of 23 core holes; includes consideration of past evidence of crack propagation in choosing inspection locations; and provides for the addition of new core holes for inspection. As previously discussed in the SER, inspection findings will be documented and evaluated by qualified engineering personnel such that the results can be trended. The applicant further stated that findings that do not meet the acceptance criteria will be evaluated and tracked using the Corrective Action Program.

The staff reviewed the applicant's "monitoring and trending" program element against the criteria in SRP-LR Section A.1.2.3.5, which states that this element should describe how data collected are evaluated.

The staff notes, based on the response to RAI B.2.43-8 evaluated in the "detection of aging effects" program element, that the data collected through inspections of the strategically located representative sample of 23 core holes includes appropriate recording and trending of the limiting "crack width" parameter required to evaluate aging effects on the bond strength of the rebar adjacent to the laminar cracking and will result in timely corrective or mitigative actions.

In LRA Amendment No. 60 dated October 6, 2015, the applicant revised the "monitoring and trending" program element to state that IR mapping will be completed on a minimum area of 100 square feet in the vicinity of a core bore with identified laminar crack propagation "to provide a relative indication of the extent of cracking propagation for condition monitoring." Based on the results obtained from the evaluation of new plant-specific operating experience observed during monitoring activities of the shield building performed in 2015, the applicant increased the minimum number of representative core bore holes to be inspected from 23 to 28, with a minimum of 14 of the core bore locations being currently uncracked. The applicant also increased the distribution of the inspections to include 9 (previously 8) out of 10 flute shoulder with high prevalence of laminar cracking. The 2015 plant-specific operating experience is documented and evaluated below in the "operating experience" program element subsection of this AMP.

Based on its review, the staff finds the monitoring and trending actions are acceptable because the inspection findings are being documented and evaluated by personnel qualified in accordance with industry standards, specifically ACI 349.3R, as recommended by the GALL Report. If inspection results do not meet the acceptance criteria, they will be evaluated and tracked, and the inspection frequency and sample size will be revised as necessary, and/or appropriately addressed in the Corrective Action Program.

The staff thus confirmed that the "monitoring and trending" program element satisfies the

criterion defined in SRP-LR Section A.1.2.3.5 and, therefore, the staff finds it acceptable.

Acceptance Criteria. By letter dated July 3, 2014, in response to the staff's RAI B.2.43-4, which addressed the 2013 plant-specific operating experience of laminar crack propagation, the applicant provided LRA Amendment 51. The staff's detailed discussion of RAI B.2.43-4 is documented in the "detection of aging effects," and "operating experience," program element evaluation of this AMP. In LRA Amendment 51 the applicant revised, in part, LRA Section B.2.43 to add that conditions to be evaluated following each inspection cycle for determination of acceptable results include conformance with the plant design and licensing basis, as well as comparison with previously determined crack propagation rates to identify potential changes in the driving force of the condition. Also, in LRA Amendment 54 dated January 28, 2015, in response to RAI B.2.43-7 discussed further below, the applicant revised, the "acceptance criteria" program element to reflect the above quantitative acceptance criteria, against which core hole inspection results will be evaluated to determine the need for any corrective actions. The staff reviewed the applicant's revised "acceptance criteria" program element against the criteria in SRP-LR Section A.1.2.3.6, which states that the acceptance criteria of the program and its basis should be described. The acceptance criteria, against which the need for corrective actions are evaluated, should ensure that the intended functions are maintained consistent with all CLB design conditions during the period of extended operation. The SRP-LR also states that the program should include a methodology for analyzing the results against applicable acceptance criteria. The acceptance criteria could be specific numeric values, or could consist of a discussion of the process for calculating specific numerical values of conditional acceptance criteria.

In its review of the "acceptance criteria" program element, as revised by LRA Amendment 51, the staff identified the following concerns related to implementation of the "acceptance criteria" program element of the AMP:

- The primary structural concern of the laminar cracking and its propagation with regard to the capability of the shield building to perform its intended functions is the potential loss of bond between the concrete and the rebar at the location of the laminar cracks and the ability for stress transfer to take place between the concrete and the rebar. This would be a function of the laminar crack width and length (or planar limit) along the rebar or rebar lap-splice. In this regard, it was not clear to the staff what quantitative (numerical) limits of laminar cracking characteristics [i.e., crack width, crack length (or planar limit), number of locations, and distribution, etc.,] are bounded by the design basis documentation referenced in the Shield Building Monitoring Program to determine the need for corrective actions (e.g., re-evaluation of design basis documentation, repair).
- The "acceptance criteria" program element in the Shield Building Monitoring Program specifies that the core bore inspection findings on the concrete laminar cracking will be compared and evaluated against two sets of pre-determined criteria to identify the need for corrective actions prior to loss of structure or component intended functions. These criteria include: (1) whether the laminar cracking is not passive (i.e., indications of new cracking, discernible change in previously identified cracks, or changes in previously determined crack propagation rates), and (2) whether the overall observed conditions are bounded by evaluations in the plant design and licensing basis documentation. With regard to the second case, the program does not appear to provide pre-determined quantitative acceptance criteria against which quantitative inspection findings can be compared and evaluated to determine the need for corrective actions (e.g., re-evaluation of design basis documentation, repair).

- The staff noted that Chapter 5 of ACI 349.3R does not provide quantitative acceptance criteria applicable to concrete laminar cracking. It was not clear to the staff how the evaluation criteria hierarchy in Figure 5.1 of ACI 349.3R will be applied to the core hole inspection findings of concrete laminar cracking, considering that only qualitative criteria related to whether laminar cracks are active or passive is available from Chapter 5 ACI 349.3R that would apply to laminar cracking.

Based on the concerns identified above related to implementing the "acceptance criteria" program element, by letter dated December 29, 2014, the staff issued RAI B.2.43-7 requesting that the applicant:

(1) Provide information on quantitative (numerical) acceptance criteria for the shield building laminar cracking defined by bounding limits of laminar cracking characterized in terms of crack width, crack length (or planar limit), distribution, and/or any other appropriate parameters, against which the core hole inspection findings are compared and evaluated to determine (a) if the condition is bounded by and conforms to the design basis documentation referenced in the AMP (e.g., FENOC calculation C-CSS-099.20-063), and (b) if corrective actions (e.g., re-evaluation of design basis documentation, repair) are needed to ensure that the structure and component intended functions are maintained consistent with CLB design conditions during the period of extended operation.

(2) Explain how the evaluation criteria hierarchy in Figure 5.1 of ACI 349.3R will be applied to the core hole inspection findings of laminar cracking from the Shield Building Monitoring Program to determine whether or not the condition is acceptable after evaluation.

In its response to the first part of RAI B.2.43-7 by letter dated January 28, 2015, the applicant stated that the quantitative acceptance criteria for core bore inspections will be (a) the maximum crack width of 0.013 inch and (b) the maximum circumferential laminar crack planar limits (in percent, rounded to the nearest whole number) as identified in SER Table 3.0.3.3.9-2 below by region (elevation) of the shield building structure.

Table 3.0.3.3.9-2 Shield Building Laminar Cracking Planar Limits

Region	Elevation (ft)	Planar Limit (percent [cracked])
1	801.0 – 812.75	0
2	774.5 – 801.0	70
3	643.0 – 774.5	20
4	565.0 – 643.0	0

The applicant clarified that the percent values provided are based on maximum currently observed cracked areas derived with respect to the entire surface area (i.e., vertical height multiplied by circumference) of the Region, and that Regions 1 and 4 are identified in the table as 0 percent cracked. The applicant also acknowledged that minor cracking exists in Region 4, which is negligible (i.e., less than 1 percent). In a telephone conference call on February 20, 2015, the applicant clarified that Region 1 was intended to be representative of the shield building dome above the spring line, which is monitored by the Structures Monitoring Program. The applicant also clarified in the call that there are no core holes in Region 1, and it

is only Regions 2 through 4 that are monitored by core hole inspections in the Shield Building Monitoring Program.

In its response to RAI B.2.43-7, the applicant also clarified that its design calculation reflects the results of the Purdue University and University of Kansas lap-splice test programs for the design basis capacity of the exterior hoop rebar that could potentially be affected by laminar cracking. The applicant stated that the test programs developed laminar cracks along the complete splice length and neglected the staggered location of lap splices in the actual shield building design, and that the results are bounding and conservative. Therefore, the splice distribution and crack length and width in relation to splice location are not considered an acceptance criteria parameter. The applicant also stated that the values of acceptance criteria above are bounded by the shield building design calculation. The applicant explained that if any of the quantitative acceptance criteria are not met based on core hole inspection findings, then the indications or conditions will be evaluated under the Corrective Action Program, which may result in corrective actions such as increased inspection frequency, re-evaluation of design basis documentation, or repair, as appropriate. As part of its RAI response, the applicant revised, in LRA Amendment 54, the “acceptance criteria” program element to reflect the above quantitative acceptance criteria, against which core hole inspection results will be evaluated to determine the need for corrective actions (if any).

In its response to the second part of RAI B.2.43-7, by letter dated January 28, 2015, the applicant stated that the 3-tiered evaluation criteria hierarchy illustrated in Figure 5.1 of Chapter 5 of ACI 349.3R was and will be applied to the shield building for the purposes of condition determination and functionality. The applicant stated that since the shield building laminar cracking condition is not passive at this time, Section 5.1, “Acceptance without further evaluation,” and Section 5.2, “Acceptance after review,” of ACI 349.3R are not applicable, and therefore, the structure was placed into the Figure 5.1 category of “Conduct Further Enhanced Inspections, Testing and Analyses,” based on the guidance of Section 5.3, “Conditions requiring further evaluation.” The applicant also stated that inspections and analysis of the structure were completed as described in Section 5.3 of ACI 349.3R, inclusive of the research lap-splice testing conducted at Purdue University and the University of Kansas, and of the revised design calculations. The applicant concluded that the “as-found condition is acceptable after evaluation” with regard to the structural adequacy of the shield building to perform its intended functions, as documented in its design basis calculation. The applicant stated that this places the structure in the “accept condition without further evaluation” conclusion of Figure 5.1 for the current condition. However, the applicant further explained that because of crack propagation identified in 2013, the condition is currently considered “not passive” and may change over time; therefore, the shield building is subject to ongoing monitoring by the Shield Building Monitoring Program during the period of extended operation. The applicant indicated that the option from Figure 5.1 being currently applied to address the “not passive” condition is to monitor core holes at an increased frequency, until no aging effects are occurring on the laminar cracks based on future inspections, as indicated in the “detection of aging effects” program element revised by letter dated July 3, 2014. The applicant also stated that inspection findings will continue to be evaluated under the Corrective Action Program when conditions adverse to quality are identified.

In a telephone conference call on February 20, 2015, the applicant clarified that if the crack size and/or planar limits change in future inspections, the new condition would be entered into its Corrective Action Program and would require re-evaluation against the structural evaluation hierarchy in Figure 5.1 of ACI Report 349.3R to ensure that the appropriate inspection frequency and scope are applied and an acceptable conclusion is reached for the condition

found at each inspection cycle. During the call the applicant also clarified its implementation process for the program's acceptance criteria. The applicant clarified that it would write a condition report in the FENOC Corrective Action Program based on any one or more of the following inspection findings:

- evidence of coating degradation that exceeds the criteria specified in the quantitative acceptance criteria for coatings in Chapter 5, Sections 5.1.4 and 5.2.4, of ACI Report 349.3R
- evidence of reinforcing bar corrosion or degradation
- any indication of new cracking
- a discernable change in previously identified cracks, such as a change in crack width or planar size
- a crack width greater than 0.013 inch
- maximum planar crack limits exceeding a value shown in SER Table 3.0.3.3.9-2 above

During the telephone conference call, the applicant also clarified that its evaluation of the condition through the Corrective Action Program would result in comparing the inspection results to design calculation limits to ensure the condition is bounded. For a change in crack planar limits or if a crack expands into a non-cracked bore, the Corrective Action Program review would trigger an evaluation to determine whether a new core bore needs to be installed to monitor future changes in the planar limits.

In its review of the first part of the RAI B.2.43-7 response, the staff noted that the laminar "crack width" is the limiting parameter with regard to the impact of the laminar cracking on the bond capacity of the outer hoop rebar and, thereby on the intended functions of the shield building. The staff also noted that the parameter defining the "planar limit" of cracking provides the means of tracking and trending the extent of laminar cracking in the shield building. Together, these parameters provide the means of tracking and evaluating potential crack propagation to determine if corrective actions are needed prior to loss of intended function. The applicant clarified, in a conference call on February 20, 2015, that if any of the quantitative limits (crack width, defined planar limit) for the crack parameters defined in the acceptance criteria are not met, or if there is any discernable change in the laminar cracks, the condition will be evaluated in the Corrective Action Program. The staff finds the response to the first part of RAI B.2.43-7 acceptable because the applicant defined quantitative acceptance criteria in terms of the critical crack width and planar limit parameters, against which core hole inspection findings are evaluated to ensure that corrective actions are taken, as appropriate, prior to the loss of intended functions of the shield building.

In its review of the second part of the RAI B.2.43-7 response, the staff noted that if the core hole inspection finding determines that there is a discernable change in the cracking or if the quantitative acceptance criteria are not met, the condition is considered as "not passive" and considered as a condition requiring further evaluation in accordance with Section 5.3 of ACI 349.3R. Further, even though the condition is found acceptable with regard to structural adequacy following further technical evaluation and/or testing, the condition is considered "not passive" and subject to monitoring at an increased frequency until it is established through several consecutive future inspections that the condition is passive. The staff finds the response to the second part of RAI B.2.43-7 acceptable because the applicant clearly explained its application of the evaluation criteria hierarchy in Figure 5.1 of ACI 349.3R to core hole

inspection findings, which the staff determined to be appropriate. The staff's concerns described in RAI B.2.43-7 are resolved.

By letter dated October 6, 2015, the applicant revised the "acceptance criteria" program element to state that concrete found to be cracked during IR mapping will be compared to the shield building baseline inspection and if crack growth is identified the condition will be entered into FENOC's Corrective Action Program.

As described above, the staff reviewed the applicant's updated acceptance criteria for findings from the core bore inspections, IR mapping, and the concrete surface inspections and finds them acceptable. The acceptance criterion for the core bore inspections is effectively no discernable change in laminar cracking, and the defined quantitative limits for crack width and planar limit are both met. Any indication of crack growth, or a new crack, will be evaluated and entered into the Corrective Action Program. The acceptance criteria for core bore inspection findings, against which the need for corrective actions are evaluated, will ensure that the shield building intended functions are maintained consistent with CLB conditions during the period of extended operation.

The staff thus confirmed that the "acceptance criteria" program element satisfies the criterion defined in SRP-LR Section A.1.2.3.6 and, therefore, the staff finds it acceptable.

Operating Experience. The staff reviewed the applicant's "operating experience" program element against the acceptance criteria in SRP-LR Section A.1.2.3.10, which states that consideration of future plant-specific and operating experience relating to AMPs should be discussed. Operating experience with similar existing programs should be discussed. The operating experience of AMPs that are existing programs, including past corrective actions resulting in program enhancements or additional programs, should be considered. Additionally, for new AMPs, an applicant should commit to a review of future plant-specific and industry operating experience for new programs to confirm their effectiveness.

The staff noted that the applicant previously reviewed industry operating experience and plant-specific experience with maintenance rule inspections of the shield building when developing the new program. In addition, the staff noted that the "operating experience" program element of the AMP states that future operating experience would be reviewed and incorporated into the program as necessary.

During Fall 2013 and early 2014, the staff became aware of the following plant-specific operating experience related to the shield building: (1) discovery of previously unidentified laminar cracks in several core bore holes during the baseline inspection conducted in August/September 2013, and (2) broken or cracked (damaged) rebar found in February 2014 near several mechanical splice locations during hydro-demolition activities for the creation of a temporary construction opening to support steam generator replacement. As indicated in the previous paragraph, the "operating experience" program element of the Shield Building Monitoring Programs states that future operating experience would be reviewed and incorporated into the program as necessary. The staff was concerned if this plant-specific operating experience had any impact on the AMPs credited for the shield building aging management for license renewal. Therefore, by letter dated April 15, 2014, the staff issued RAI B.2.43-4 requesting the applicant to describe and justify modifications or enhancements, if any, that may be potentially made to the AMPs, credited for the shield building for license renewal, to account for (1) plant-specific operating experience related to cracking identified in August/September 2013, and (2) the damaged rebar found in February 2014 in the temporary

construction opening area. The staff also requested that the applicant provide its basis if no changes to the AMPs were required.

The applicant provided its response to RAI B.2.43-4 by letter dated July 3, 2014. In its response to the second part of RAI B.2.43-4, related to the damaged rebar, the applicant stated that no revision to the AMPs will be made as a result of the rebar that were found broken or cracked in the shield building construction opening area in February 2014. The applicant stated that the rebar failure near the mechanical splice locations in the temporary construction opening area did not represent an aging management issue because it was caused by hydro-demolition from changes in rebar restraint, an increase in rebar dowel length, a decrease in temperature, and an increase in stress cycles from hydro-demolition trimming which collectively caused fatigue failure. The applicant also stated that all exposed rebar were examined and the damaged rebar were replaced prior to restoration of the construction opening.

The staff finds the applicant's response to the second part of RAI B.2.43-4, related to the February 2014 rebar damage, acceptable because the rebar failure resulted from the physical process of hydro-demolition for creation of the temporary construction opening, it was evaluated and addressed prior to restoration of the opening, and it did not represent an aging effect or aging management issue for license renewal. The staff's concern described in the second part of RAI B.2.43-4 is resolved.

In its response to the first part of RAI B.2.43-4, related to the August/September 2013 discovery of previously unidentified cracks, the applicant characterized the "newly" identified cracks as propagation of laminar cracking by the phenomenon of "ice-wedging" and stated that the minimum representative sample of monitored core bores in the Shield Building Monitoring Program was increased from 20 to 23 as a result of this plant-specific operating experience with 3 additional bores added to aid in identifying changes in the limits of cracking in areas with previously identified crack propagation. The applicant further stated that the frequency of internal visual inspection for the 23 monitoring bores is changed to annual inspections for a minimum of 4 years (2015-2018) and then progressively increased to a 4-year interval after 2026 if no aging effects are found. Further, the applicant stated that these inspection intervals will be evaluated for effectiveness and modifications to the Shield Building Monitoring Program will be determined in the Corrective Action Program, should there be an identified significant change in cause, rate of crack growth, or a condition that is not bounded by the design basis documentation. The applicant revised the USAR Supplement of the AMP in LRA Section A.1.43 to address the changes regarding selection of additional core hole locations in the representative inspection sample. The applicant also revised the description of the AMP in LRA Section B.2.43 to include an operating experience summary of the previously unidentified crack conditions found in 2013 and results of the evaluation of the crack progression. The applicant revised the "detection of aging effects," "monitoring and trending," "acceptance criteria," and "operating experience" program elements of the Shield Building Monitoring Program to incorporate changes due to the plant-specific operating experience of crack propagation. The staff notes that these changes have been reflected and evaluated in the staff evaluation sections above for the respective program elements of the Shield Building Monitoring Program.

The applicant supplemented the "operating experience" program element with the following description of the 2013 plant-specific operating experience:

> Inspections of 12 core bores were completed in 2013 under the "Design Guidelines for Maintenance Rule Evaluation of Structures" Procedure EN-DP-01511. During that cycle

of inspections, a crack was observed in one of the core bores. This finding, upon a review of records, was determined to be a pre-existing crack given that the extracted concrete core was cracked at the location identified. Given this finding, the inspection population was increased, eventually leading to inspection of all available core bores. This re-inspection identified a total of 7 core bores with similar conditions that were determined to be pre-existing. This re-inspection also identified eight conditions where the laminar cracking conditions were determined to have undergone a discernable change.

The cracking propagation was determined to be a result of ice-wedging (freezing water at a pre-existing crack leading edge). This condition requires water, freezing temperatures and pre-existing cracks. Because the [s]hield [b]uilding has been coated [Fall 2012] it contains a finite amount of water. It is not practical to remove the water in an accelerated manner given the cumulative magnitude of leading crack edges and transportability of water. It is also not practical to remove the existing cracks or prevent freezing temperatures. The rate of cracking propagation is estimated at 0.4 to 0.7 inches per freezing cycle based on laboratory simulation. By application of the evaluation criteria hierarchy of ACI 349.3R "Evaluation of Existing Nuclear Safety-Related Concrete Structures" Figure 5.1, the condition was acceptable through evaluation. The condition was not passive; however, it was bounded by design basis documentation. The Shield Building Monitoring Program was changed to ensure conformance with the design requirements and to maintain the USAR functions.

The shield building laminar cracking condition has been evaluated with respect to the design basis functions of the shield building. The condition is documented in FENOC Calculation C-CSS-099.20-063, as supported by Bechtel Report "Effect of Laminar Cracks on Splice Capacity of No. 11 Bars based on Testing Conducted at Purdue University and University of Kansas for Davis-Besse Shield Building," that the shield building meets all design requirements specified in the USAR and it will perform its USAR described design functions. This analysis bounds the identified changes in the laminar cracking condition from the conditions identified in 2011.

The staff evaluation of the applicant's response to the first part of RAI B.2.43-4, related to the "operating experience" program element, is provided here. The staff noted that the applicant's causal analysis characterized the 2013 plant-specific operating experience as laminar crack propagation determined to be the result of an "ice-wedging" mechanism (i.e., freezing and expansion of water at a pre-existing crack leading edge). The applicant determined that for this condition to occur requires pre-existing cracks, water in the laminar cracks, and freezing temperatures. The staff finds that the applicant's response to RAI B.2.43-4 with regard to the "operating experience" program element of the Shield Building Monitoring Program is acceptable because: (1) it provided an adequate review and characterization of the plant-specific operating experience of previously unidentified cracks discovered in 2013 and their apparent cause, (2) it confirmed the effectiveness of the AMP and identified areas for enhancements, and (3) it described modifications and enhancements to the applicable program elements to appropriately account for the plant-specific operating experience to ensure adequate aging management. The staff's concern with regard to the "operating experience" program element aspects of RAI B.2.43-4 is resolved. The staff evaluation of the applicant's response to the remaining aspects of the first part of RAI B.2.43-4, with regard to change in sample size and inspection frequency, are addressed in the "detection of aging effects" program element and are also acceptable. Therefore, the staff's concerns described in RAI B.2.43-4 are resolved.

By letter dated October 6, 2015, the applicant revised the "operating experience" program element to incorporate plant-specific operating experience resulting from the shield building monitoring activities performed in 2015. The applicant stated that it completed the inspection of 23 core bores in 2015 and observed cracking propagation in two of the locations where propagation was identified in 2013, as well as in one additional location. The applicant stated that the condition was evaluated with respect to design basis functions, and determined to be acceptable and bounded by the plant's design basis documentation (FENOC Calculations C-CSS-099.20-063 and C-CSS-099.20-069). The staff notes, as described in the "detection of aging effects" and "monitoring and trending" program elements above, that based on evaluation of the 2015 plant-specific operating experience, the applicant increased its core bore inspection sample size from 23 to 28.

Based on its review of LRA Section B.2.43 submitted by letter dated November 20, 2012, and as amended by letters dated July 3, 2014, January 28, 2015, and October 6, 2015, and the applicant's responses to RAIs B.2.43-4 through B.2.43-8, the staff finds that the applicant has appropriately evaluated plant-specific and industry operating experience related to the applicant's program, which demonstrates that it can adequately manage the detrimental effects of aging on SSCs within the scope of the program and that implementation of the program has resulted in the applicant taking appropriate corrective actions. The staff thus confirmed that the "operating experience" program element satisfies the criterion in SRP-LR Section A.1.2.3.10 and, therefore, the staff finds it acceptable.

USAR Supplement. LRA Section A.1.43, as amended by letters dated July 3, 2014, and October 6, 2015, provides the USAR supplement for the Shield Building Monitoring Program. The staff reviewed this USAR supplement description of the program against the recommended description for this type of program as described in SRP-LR Table 3.0-1.

The staff noted that the supplement description contained an appropriate level of detail, including a discussion of the parameters monitored or inspected, representative core bore sample size, method and frequency of inspections, inspection location distribution and consideration of past evidence of crack propagation in choosing inspection locations, acceptance criteria, inspector qualifications, and a reference to the appropriate industry guidance documents, specifically ACI 349.3R. The staff also notes that the applicant committed (Commitment No. 46) to implement the new Shield Building Monitoring Program prior to October 22, 2016 (i.e., 6 months prior to entering the period of extended operation) for managing aging of applicable components. The applicant also committed (Commitment No. 20) to use the acceptance criteria of ACI 349.3R for inspection of the coatings.

Based on its review, the staff finds that the information in the USAR supplement, as amended by letters dated July 3, 2014, and October 6, 2015, is an adequate summary description of the program, as required by 10 CFR 54.21(d).

Conclusion. On the basis of its technical review of the applicant's plant-specific Shield Building Monitoring Program, as submitted by letter dated November 20, 2012, and as amended by RAI response letters dated July 3, 2014, October 28, 2014, and January 28, 2015; and LRA Amendment No. 60 dated October 6, 2015; the staff concludes that the applicant has demonstrated that the effects of aging on the shield building laminar cracking will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP, as amended by letter dated July 3, 2014, and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.3.10 Service Level III Coatings and Linings Monitoring Program

Summary of Technical Information in the Application. After the LRA was submitted, based on reviews of recent industry operating experience and several LRAs, the staff identified an issue concerning loss of coating integrity of internal coatings of piping, piping components, heat exchangers, and tanks. By letter dated November 26, 2013, the staff issued RAI 3.0.3-3 requesting that the applicant address how it will manage loss of coating integrity for internal coatings on in-scope piping, heat exchangers, and tanks.

As amended by letter dated January 31, 2014, LRA Section B.2.44 describes the new Service Level III Coatings and Linings Monitoring Program as plant specific. The LRA states that the AMP addresses organic (e.g., elastomeric or polymeric) and inorganic (e.g., zinc-based) internal coatings and linings (e.g., rubber, cementitious) on in-scope piping, piping components, heat exchangers, and tanks to manage the effects of loss of coating integrity. The LRA also states that the AMP proposes to manage this aging effect through periodic visual inspections of the coatings.

Staff Evaluation. The staff reviewed program elements one through six of the applicant's program against the acceptance criteria for the corresponding elements as stated in SRP-LR Section A.1.2.3. The staff's review focused on how the applicant's program manages aging effects through the effective incorporation of these program elements. The staff's evaluation of each of these program elements follows. The staff's review of the "corrective actions," "confirmation process," and "administrative controls" program elements are documented in SER Section 3.0.4.

The staff recommended actions to manage loss of coating integrity for internal coatings of piping, piping components, heat exchangers, and tanks. The staff has determined that additional recommendations are appropriate to manage loss of coating integrity for internal coatings of piping, piping components, heat exchangers, and tanks. The staff has concluded that the following recommended actions provide one acceptable approach for managing the associated aging effects for components within the scope of license renewal. Throughout the remainder of this SER Section, the statement, "staff's recommended actions to manage loss of coating integrity," is in reference to this subsection of the SER. The staff concluded the following:

- Periodic visual inspections of coatings to detect blistering, cracking, flaking, peeling, delamination, rusting, spalling (for cementitious coatings), and physical damage should be conducted. For purposes of license renewal, physical damage would be limited to age-related mechanisms such as that occurring downstream of a throttled valve as a result of cavitation versus damage caused by inspection activities (e.g., chipping of the coating due to installation of scaffolding, removal and reinstallation of inspection ports). Inspections are conducted for each coating material and environment combination. The coating environment includes both the environment inside the component (e.g., raw water) and the metal to which the coating is attached.
- Baseline inspections should be conducted in the 10-year period prior to the period of extended operation. Subsequent inspections should be based on the results of these and follow-on inspections as follows:

  (a) If no peeling, delamination, blisters, or rusting are observed during inspections, and cracking, flaking, or spalling (in cementitious coatings) has been found acceptable, subsequent inspections should be conducted 6 years after the most recent

inspection. Peeling, delamination, blisters, or rusting can be indicative of loss of adhesion that could result in the coating becoming debris or not being able to perform a corrosion deterrence function. Cracking, flaking, or spalling, although indicators of some degree of coating degradation, are not significant enough to require more frequent inspections as long as the condition has been found acceptable by qualified personnel. For example, despite cracking being found, the base metal could still be isolated from the environment and the coating retain sufficient integrity so as not to become debris.

(b) If the prior inspection results do not meet "a." above, and a coatings specialist has determined that no remediation is required, subsequent inspections should be conducted 4 years after the most recent inspection. More frequent inspections are warranted to confirm the coatings specialist's evaluation. If two sequential subsequent inspections demonstrate no change in coating condition, subsequent inspections may be conducted at 6-year intervals.

(c) Given that coatings in redundant trains are exposed to the same environment, the inspection interval may be extended to 12 years as long as: (a) the identical coating material was installed with the same installation requirements in redundant trains (e.g., piping segments, tanks) with the same operating conditions and at least one of the trains is inspected every 6 years, and (b) the coating is not in a location subject to turbulence that could result in mechanical damage to the coating.

(d) Given that the coatings installed on the internal surfaces of diesel fuel oil storage tanks are generally exposed to a static environment, the inspection interval may be conducted in accordance with GALL Report AMP XI.M30, "Fuel Oil Chemistry," as long as the inspection results meet "a.," above.

- The extent of inspections should include all accessible tank and heat exchanger internal surfaces. The staff recognizes that, for piping, extensive amounts of coating could be installed. GALL Report AMPs such as XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components," are based on sampling a portion of the population. The staff has concluded that using a sampling-based extent of inspections is appropriate for coatings installed on the internal surfaces of piping. Where documentation exists that manufacturer recommendations and industry consensus documents (i.e., those recommended in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants" or earlier versions of those standards) were used during installation, the extent of piping inspections may be 25 1-foot axial length circumferential segments of piping or 20 percent of the total length of each coating material and environment combination. This extent of sampling is consistent with several GALL Report AMPs. However, where documentation does not exist that manufacturer recommendations and industry consensus documents were used during installation, the staff has concluded that a larger extent of inspection is appropriate, consisting of 73 1-foot axial length circumferential segments of piping or 50 percent of the total length of each coating material and environment combination. Regardless of the extent of inspections, the inspection surface includes the entire inside surface of the 1-foot sample. If geometric limitations impede movement of remote or robotic inspection tools, the number of inspection segments is increased in order to cover an equivalent length.

- The staff has concluded that, where loss of coating integrity cannot result in downstream effects such as reduction in flow, drop in pressure, or reduction in heat transfer for in-scope components, a representative sample of external wall thickness measurements can be used to confirm the acceptability of the corrosion rate of the base metal in lieu of visual inspections of the coating. The wall thickness measurements are an appropriate method to manage loss of coating integrity in this case because base metal corrosion is the only effect of loss of coating integrity.
- RG 1.54 provides the staff position for training and qualifying individuals involved in coating inspections and evaluating degraded conditions.
- A pre-inspection review of the previous two inspections should be conducted, including reviewing the results of inspections and any subsequent repair activities. A coatings specialist should prepare the post-inspection report to include: a list and location of all areas evidencing deterioration, a prioritization of the repair areas into areas that must be repaired before returning the system to service and areas where repair can be postponed to the next refueling outage, and where possible, photographic documentation indexed to inspection locations. When corrosion of the base material is the only issue related to coating degradation of the component and when external wall thickness measurements are used in lieu of internal visual inspections of the coating, the corrosion rate of the base metal should be trended. These recommendations are consistent with ASTM D7167-05, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant," which is referenced in RG 1.54.
- Based on the staff's review of industry documents (e.g., ASTM, EPRI), the staff has concluded that, with the exception of Service Level I qualification testing, there are no acceptance criteria in recognized industry consensus documents. Acceptance of degraded coatings is established by the coatings specialist. RG 1.54 states that, for Service Level I coatings: (a) peeling and delamination is not permitted, (b) cracking is not considered a failure unless it is accompanied by delamination or loss of adhesion, and (c) blisters are limited to intact blisters that are completely surrounded by sound coating bonded to the surface. The staff has established the following acceptance criteria for loss of coating integrity, based on the recommendations in RG 1.54.

  (a) Indications of peeling and delamination are not acceptable and the coating is repaired or replaced.

  (b) Blisters can be evaluated by a coatings specialist qualified in accordance with an ASTM International standard endorsed in RG 1.54, including staff guidance associated with the use of a particular standard. Blisters should be limited to a few intact small blisters that are completely surrounded by sound coating bonded to the substrate. If the blister is not repaired, physical testing (e.g., lightly tapping the coating, adhesion testing) is conducted to ensure that the blister is completely surrounded by sound coating bonded to the surface. Acceptance of a blister to remain in service should be based both on the potential effects of flow blockage and degradation of the base material beneath the blister.

  (c) If coatings are credited for corrosion prevention (e.g., corrosion allowance in design calculations is zero, the "preventive actions" program element credited the coating) and the base metal has been exposed or it is beneath a blister, the component's base material in the vicinity of the degraded coating is examined to determine if the minimum wall thickness is met and will be met until the next inspection.

(d) Indications such as cracking, flaking, and rusting are to be evaluated by a coatings specialist qualified in accordance with an ASTM International standard endorsed in RG 1.54, including staff guidance associated with the use of a particular standard.

(e) Minor cracking and spalling of cementitious coating is acceptable, provided there is no evidence that the coating is debonding from the base material.

(f) As applicable, wall thickness measurements meet design minimum wall requirements.

(g) Results of adhesion testing, when conducted, meet or exceed the degree of adhesion recommended in engineering documents specific to the coating and substrate.

- Coatings that do not meet the acceptance criteria should be repaired or replaced. Testing or examination is conducted to ensure that the extent of repaired or replaced coatings encompasses sound coating material. These recommendations are consistent with ASTM D7167-05, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant," which is referenced in RG 1.54.

Scope of Program. LRA Section B.2.44 states that the scope of the program includes internal coatings and linings for the fire water storage tank; portions of the underground fire water main piping and piping components; the diesel oil storage tank; main lubricating oil pump casings in the EDGs and station blackout diesel generator; piping in the circulating water system; and valves, strainers, and pumps in the service water system.

The staff reviewed the applicant's "scope of program" program element against the criteria in SRP-LR Section A.1.2.3.1, which states that the scope of the program should include the specific structures and components for which loss of coating integrity will be managed.

The staff finds the applicant's "scope of program" program element to be adequate because the scope of internally coated in-scope piping, heat exchangers, and tanks has been identified.

Based on its review of the application, as amended by letter dated January 31, 2014, the staff confirmed that the "scope of program" program element satisfies the criterion defined in SRP-LR Section A.1.2.3.1 and, therefore, the staff finds it acceptable.

Preventive Actions. LRA Section B.2.44 states that the Service Level III Coatings and Linings Monitoring Program is a condition monitoring program that does not include preventive actions.

The staff reviewed the applicant's "preventive actions" program element against the criteria in SRP-LR Section A.1.2.3.2, which states that some condition monitoring programs do not rely on preventive actions and thus, this information need not be presented.

The staff finds the applicant's "preventive actions" program element to be adequate because this program is a condition monitoring program that need not rely on preventive actions.

Based on its review of the application, as amended by letter dated January 31, 2014, the staff confirmed that the "preventive actions" program element satisfies the criterion defined in SRP-LR Section A.1.2.3.2 and, therefore, the staff finds it acceptable.

Parameters Monitored or Inspected. LRA Section B.2.44 states that visual inspections of the coatings and linings will be used to detect blistering, cracking, flaking, peeling, delamination, physical damage of coatings, and rusting of the underlying base metal.

The staff reviewed the applicant's "parameters monitored or inspected" program element against the criteria in SRP-LR Section A.1.2.3.3, which states that the program should identify the aging effects that the program manages and provide a link to the parameters that will be monitored and that the parameter monitored or inspected should be capable of detecting the presence and extent of aging effects.

The staff noted that the applicant did not address spalling in cementitious coatings; however, it is addressed in the "detection of aging effects" program element. The staff finds the applicant's "parameters monitored or inspected" program element to be adequate because the aging effects and parameters monitored or inspected are consistent with the staff's recommended actions to manage loss of coating integrity.

Based on its review of the application, as amended by letter dated January 31, 2014, the staff confirmed that the "parameters monitored or inspected" program element satisfies the criteria defined in SRP-LR Section A.1.2.3.3 and, therefore, the staff finds it acceptable.

Detection of Aging Effects. LRA Section B.2.44 states that baseline visual inspections will be conducted in the 10-year period prior to the period of extended operation, followed by periodic inspections. A coatings specialist will establish the periodicity of subsequent inspections, which, will be based on an evaluation of the effect of a coating failure on the in-scope component's intended function, potential problems identified during prior inspections, and known service life history. The periodicity of subsequent inspections is subject to a maximum interval between inspections as follows:

- six years if no peeling, delamination, blisters, or rusting are observed and any cracking and flaking have been found acceptable
- six years for cementitious coatings where no cracking or spalling is observed
- twelve years if (a) the identical coating material was installed with the same installation requirements in redundant trains (e.g., piping segments) with the same operating conditions and at least one of the trains is inspected every 6 years, and (b) the coating is not in a location subject to turbulence
- every other refueling outage if the inspection results do not meet the criteria in the first two bullets, above, but a coating specialist has determined that no remediation is required, noting that the subsequent inspections will include locations that resulted in subsequent inspections being conducted, as well as new locations, and that following two sequential subsequent inspections that demonstrate no change in coating condition, subsequent inspections may be conducted at 6-year intervals
- a refueling outage interval for the two intervals following a coating repair or replacement or for newly installed coatings, to establish a performance trend on the coatings, after which the inspection interval may be increased by the coatings program owner as described above
- every 5 years for the fire water storage tank inspection interval, as established in NFPA 25, Section 9.2.6

The program also includes both the extent of inspections (i.e., all accessible internal surfaces of in-scope internally coated tanks will be inspected and a representative sample of internally coated piping components consisting of at least 73 1-foot axial length circumferential segments of piping or 50 percent of the total length of each coating material and environment combination will be inspected) and the qualifications of individuals involved in coating inspection activities.

The program also states the following:

- Visual inspections are conducted in accordance with ASTM D7167-05, "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant."
- Coatings surveillance personnel will be knowledgeable on coatings, meeting the requirements of ASTM D4537-12; "Standard Guide for Establishing Procedures to Qualify and Certify Personnel Performing Coating and Lining Work Inspection in Nuclear Facilities."
- The coatings program owner will be a nuclear coatings specialist meeting the requirements of ASTM D7108-12, "Standard Guide for Establishing Qualifications for a Nuclear Coatings Specialist."
- Wall thickness measurements can be performed in lieu of internal inspections to confirm the acceptability of the corrosion rate of the base metal if: (a) the degradation of coatings cannot result in downstream effects such as reduction in flow, drop in pressure, or reduction in heat transfer for in-scope components and (b) corrosion of the base material is the only issue related to coating degradation of the component; external.

The staff reviewed the applicant's "detection of aging effects" program element against the criteria in SRP-LR Section A.1.2.3.4, which states that the program should describe: (a) how the program element will be capable of detecting the occurrence of age-related degradation prior to the loss of the current licensing-basis intended function(s) of in-scope components; (b) the when, where, and how data is collected; and (c) the basis of the sample size and location selection.

The staff finds the applicant's "detection of aging effects" program element to be adequate, in part because baseline inspections are conducted prior to the period of extended operation; the type of inspections, inspection intervals, extent of inspections, and sample size are consistent with the staff's recommended actions to manage loss of coating integrity; and the qualifications of individuals conducting and overseeing coating inspections are consistent with the standards listed in RG 1.54 "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants." In addition, conducting internal coating inspections of fire water storage tanks at 5-year intervals is consistent with LR-ISG-2012-02 AMP XI.M27. However, the RAI response did not address how inspection locations would be selected and the length of piping that will be examined if geometric limitations impede access to the entire internal circumference of any piping segment. By letter dated July 7, 2014, the staff issued RAI 3.0.3-3a requesting that the applicant state the criteria for selecting a representative sample of internally coated piping and piping components and state the length of piping that will be examined if geometric limitations impede access to the entire internal circumference of any piping segment.

In its response dated July 29, 2014, the applicant stated that the criteria for the selection of inspection locations would be based on several factors. Examples of these factors include the effect of coating failure, areas with aggressive environmental conditions, problem areas based on plant-specific operating experience, and areas where physical damage could occur. The

applicant also stated that the axial length of inspected piping would be increased if geometric limitations prevented inspection of the complete circumferential surface area of the sample. The applicant provided an example consistent with the staff's recommended actions to manage loss of coating integrity, above. The staff noted that the above changes were incorporated into LRA Section B.2.44.

The staff finds the applicant's response acceptable because the factors presented in the applicant's response are inclusive of areas where the potential for loss of coating integrity is highest and the extent of pipe inspections ensures that, despite potential access limitations, an appropriate amount of piping will be inspected, consistent with the staff's recommended actions to manage loss of coating integrity. The staff's concern described in RAI 3.0.3-3a is resolved.

Based on its review of the application, as amended by letter dated January 31, 2014, and review of the applicant's response to RAI 3.0.3-3a, the staff confirmed that the "detection of aging effects" program element satisfies the criteria defined in SRP-LR Section A.1.2.3.4 and, therefore, the staff finds it acceptable.

Monitoring and Trending. LRA Section B.2.44 states that, prior to performing inspections, a pre-inspection review of at least the previous two inspection reports will be conducted to identify areas where degraded coatings exist and any subsequent repair activities. Inspection results are reviewed by a qualified nuclear coatings specialist. Identified degradation that does not meet the acceptance criteria is documented and evaluated in accordance with the Corrective Action Program. The coatings program owner (or designee qualified as a nuclear coatings specialist) will evaluate inspection results and prepare the inspection report to include a list and location of all areas evidencing deterioration, a prioritization of the repair areas into areas that must be repaired before returning the system to service and areas where repair can be postponed to the next inspection, and where possible, photographic evidence indexed to inspection locations. When corrosion of the base material is the only issue related to coating degradation of the component and external wall thickness measurements are used in lieu of internal visual inspections of the coating, the corrosion rate of the base metal is trended.

The staff reviewed the applicant's "monitoring and trending" program element against the criteria in SRP-LR Section A.1.2.3.5, which states that monitoring and trending activities should be described and the results should be evaluated against the acceptance criteria to effect timely corrective or mitigative actions.

The staff noted that the applicant stated that previous inspection results would be reviewed and areas of coating degradation are to be documented and evaluated. In addition, when wall thickness measurements are used to monitor coated base material, the corrosion rate will be trended. The staff finds the applicant's "monitoring and trending" program element to be adequate because it is consistent with the staff's recommended actions to manage loss of coating integrity.

Based on its review of the application, as amended by letter dated January 31, 2014, the staff confirmed that the "monitoring and trending" program element satisfies the criteria defined in SRP-LR Section A.1.2.3.5 and, therefore, the staff finds it acceptable.

Acceptance Criteria. LRA Section B.2.44 states the definitions for potential degraded coating conditions (e.g., blistering - formation of bubbles in a coating film) and the corresponding acceptance criteria.

The staff reviewed the applicant's "acceptance criteria" program element against the criteria in SRP-LR Section A.1.2.3.6, which states that the acceptance criteria of the program and its basis should be described and the acceptance criteria should ensure that the component's intended function(s) are met.

The staff noted that the acceptance criteria are as follows: (a) acceptable coatings are free of peeling or delamination, (b) blistering will be evaluated by a nuclear coatings specialist to determine acceptability, (c) adhesion testing should be conducted to ensure that the blister is completely surrounded by sound coating bonded to the surface, (d) cracking, flaking, rusting, and physical damage will be evaluated by a nuclear coatings specialist to determine acceptability, (e) where access to the interior of components is permitted, adhesion testing should be conducted for coated areas that are determined to be suspect, deficient, or degraded as directed by the coatings program owner, (f) minor cracking and spalling of cementitious coatings is acceptable, provided there is no evidence that the coating is debonding from the base material, (g) wall thickness measurements meet design minimum wall requirements, and (h) adhesion testing results meet or exceed the degree of adhesion recommended in engineering documents specific to the coating and substrate. Coatings that do not meet the acceptance criteria will be documented in the Corrective Action Program. Corrective actions may include determining the cause of the coating failure, leaving the degraded coatings as-is until the next scheduled inspection, replacing, or repairing the coating. The staff finds the applicant's "acceptance criteria" program element to be adequate, in part, because it is consistent with the staff's recommended actions to manage loss of coating integrity. However, the staff noted that the applicant stated that adhesion testing should be conducted to ensure the blister is completely surrounded by sound coating bonded to the surface and where access to the interior of components is permitted. By letter dated July 7, 2014, the staff issued RAI 3.0.3-3b requesting that the applicant state what criteria will be used to determine when adhesion testing will be conducted.

In its response dated July 29, 2014, the applicant revised LRA Sections A.1.44 and B.2.44 to state that adhesion testing would be conducted in accordance with standards endorsed in RG 1.54, where physically possible, when the acceptance criteria for peeling, delamination, or blistering are met.

The staff finds the applicant's response acceptable because adhesion testing will be conducted to ensure that the extent of indications of peeling, delamination, and blistering are understood, consistent with the staff's recommended actions to manage loss of coating integrity, above. The staff recognizes that the applicant's statement, "where physically possible," is reasonable because adhesion testing will not be able to be performed in all configurations due to access restrictions perpendicular to the degraded coating. In these instances, the knowledge and experience of the nuclear coatings specialist is relied upon to conduct an acceptable evaluation of the degraded coatings. The staff's concern described in RAI 3.0.3-3b is resolved.

Based on its review of the application, as amended by letter dated January 31, 2014, and on its review of the applicant's response to RAI 3.0.3-3b, the staff confirmed that the "acceptance criteria" program element satisfies the criteria defined in SRP-LR Section A.1.2.3.6 and, therefore, the staff finds it acceptable.

The staff finds the applicant's response to RAI 3.0.3-3, RAI 3.0.3-3a, and RAI 3.0.3-3b associated with the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," and "acceptance criteria"

program elements acceptable as documented above. The staff's concern associated with these program elements in RAI 3.0.3-3 is resolved.

Operating Experience. LRA Section B.2.44 summarizes operating experience related to the Service Level III Coatings and Linings Monitoring Program. The applicant's review of plant-specific operating experience related to Service Level III coatings revealed that the epoxy lining for the four service water system strainers has degraded in the past and significant base metal pitting has occurred. This operating experience provided the basis for the existing strainer replacement schedule. No downstream affects from the degraded coating were identified.

The staff reviewed this information against the acceptance criteria in SRP-LR Section A.1.2.3.10, as amended by LR-ISG-2011-05, which states that currently available operating experience applicable to new programs and consideration of future plant-specific and industry operating experience relating to AMPs should be discussed. The staff's evaluation of the applicant's consideration of future operating experience is documented in SER Section 3.0.5.

During its review, the staff did not identify any operating experience that would indicate that the applicant should consider modifying its proposed program.

Based on its review of the application, as amended by letter dated January 31, 2014, the staff finds that the applicant has appropriately evaluated plant-specific and industry operating experience. The staff confirmed that the "operating experience" program element satisfies the criteria in SRP-LR Section A.1.2.3.10 and, therefore, the staff finds it acceptable.

USAR Supplement. LRA Section A.1.44, as revised by letter dated July 29, 2014, provides the USAR supplement for the Service Level III Coatings and Linings Monitoring Program.

The staff reviewed this USAR supplement description of the program and noted that it is consistent with the recommended description in SRP-LR Table 3.0-1 for plant-specific programs.

The staff also noted that the applicant committed to implementing the new Service Level III Coatings and Linings Monitoring Program prior to October 22, 2016, for managing the effects of aging for applicable components.

The staff finds that the information in the USAR supplement is an adequate summary description of the program.

Conclusion. On the basis of its technical review of the applicant's Service Level III Coatings and Linings Monitoring Program, the staff concludes that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the USAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.0.3.4 Aging Management Related to Recurring Internal Corrosion

3.0.3.4.1 Recurring Internal Corrosion

Summary of Technical Information in the Application. By letter dated February 19, 2014,the applicant provided changes to the Davis-Besse LRA to address the updated guidance in LR-ISG-2012-02, for loss of material due to recurring internal corrosion. Based on its reviews, the applicant identified recurring internal corrosion in the service water system at Davis-Besse and provided a new LRA Section 3.3.2.2, "Aging Management Review Results for Which Further Evaluation Is Recommended by NUREG-1801." The new LRA Section 3.3.2.2.16, "Loss of Material Due to Recurring Internal Corrosion," is associated with LRA Table 3.3.1, item 3.3.1-127, and addresses metallic piping components in auxiliary systems exposed to raw water for recurring loss of material. The LRA amendment states that ultrasonic thickness measurements identified piping segments in the service water system that were below procedural limits. The applicant evaluated the reduced wall thickness of the piping segments using the Corrective Action Program and replaced piping and associated components as required. The new LRA section also states that NDE methods, employing visual examinations and UT, are used to detect loss of material, prior to any loss of component intended function. When visual inspections detect surface irregularities that could indicate wall loss below nominal wall thickness, followup volumetric wall thickness examinations are performed. The applicant also stated that inspection procedures include acceptance criteria that are based on code requirements, design-basis calculations, or analyses of system performance (i.e., allowable wall thickness). The applicant concluded by stating that the Open-Cycle Cooling Water System Program uses the Corrective Action Program to document degradations and to evaluate corrosion rates, so that piping and related components are replaced prior to loss of function.

As part of the LRA amendment, the applicant also addressed AMR items 3.2.1-66 and 3.4.1-61, which are associated with the Further Evaluation items in the Engineered Safety Features Systems and the Steam and Power Conversion Systems, respectively. For both of these items, the applicant stated that Davis-Besse has no in-scope components subject to recurring internal corrosion in the related systems; therefore, these AMR items are not applicable.

Staff Evaluation. In its review of the amendment to the LRA, the staff noted that the applicant did not include all aspects of the acceptance criteria specified in LR-ISG-2012-02 for recurring internal corrosion. The applicant did not address the adequacy of augmented or lack of augmented inspections that will be performed and did not discuss the decision points, where increased inspections would be implemented. In addition, the applicant did not discuss how buried or underground components will be inspected, or how leaks in these components would be identified. Based on these concerns, by letter dated July 7, 2014, the staff issued RAI 2.3.3.26-2 requesting that the applicant provide a more comprehensive response that addresses all aspects of the Further Evaluation section for recurring internal corrosion included in LR-ISG-2012-02.

In its response dated July 29, 2014, the applicant addressed each of the five elements associated with the further evaluation for recurring internal corrosion that were delineated in LR-ISG-2012-02. In addition, the applicant revised LRA Section 3.3.2.2.16 in its entirety, to reflect the changes for each of the five elements. The applicant clarified that the effects of aging, including recurring internal corrosion, will be managed by the open-cycle cooling water system during the period of extended operation. The applicant stated that the program's examination methods, visual inspections, and augmented volumetric examinations have been demonstrated to be sufficient to detect recurring aging effects prior to any loss of component intended functions. With regard to the adequacy of inspections, the applicant stated that augmented volumetric inspections are performed where indications of wall loss below nominal pipe wall thickness are detected. In addition, corrosion rates are evaluated through the Corrective Action Program to replace components prior to loss of function. The applicant stated that wall thickness will be trended and the frequency of the inspections will be adjusted based on the observed trends. With regard to inspections of components not easily accessed and the identification of leaks in any buried components, the applicant stated that it will use UT and video cameras to provide information on the internal condition of the piping. Furthermore, system leaks will be identified through flow monitoring.

The staff finds the applicant's revised response acceptable because the examination methods proposed by the applicant will be sufficient to detect recurring internal corrosion. Specifically, the visual inspections followed by the augmented volumetric examinations will be adequate for detecting indications of wall loss below nominal pipe wall thicknesses, and subsequent inspection frequencies can be adjusted based on observed corrosion rates.

Based on the program identified, the staff also determines that the applicant's program meets SRP-LR Section 3.3.2.2.8 (as modified by LR-ISG-2012-02). For the item associated with LRA Section 3.3.2.2.16, the staff concludes that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). In addition, the staff evaluated the applicant's claim that AMR items 3.2.1-66 and 3.4.1-61 are not applicable and finds it acceptable, based on the associated statements in its response dated February 19, 2014.

3.0.3.5 Staff Evaluation of LRA Changes To Incorporate LR-ISG-2012-01, "Wall Thinning Due to Erosion Mechanisms"

Summary of Technical Information in the Application. By letter dated June 23, 2014, the applicant provided its evaluation of LR-ISG-2012-01, "Wall Thinning Due to Erosion Mechanisms." The applicant used keywords from the LR-ISG to search its corrective action database for previously identified issues related to erosion, flashing, or cavitation at Davis-Besse. The applicant's review identified issues of cavitation that had been addressed by either design or operational changes to eliminate the issue. In addition, the applicant noted that erosion in raw water systems is being managed by the Open-Cycle Cooling Water AMP. The applicant stated that plant operating experience shows that an additional monitoring program is not needed to manage erosion, flashing, or cavitation.

Staff Evaluation. The staff independently searched the applicant's operating experience database as part of its review of the Flow-Accelerated Corrosion Program during the onsite AMP Audit. During its reviews, the staff did not identify any concerns related to cavitation or erosion. The staff also noted that its evaluation of the applicant's Open-Cycle Cooling Water Program, as documented in SER Section 3.0.3.2.12, includes activities to manage loss of

material due to erosion. The staff finds the applicant's evaluation of LR-ISG-2012-01 acceptable and agrees with the applicant's determination that an additional monitoring program is not needed for managing wall thinning due to erosion mechanisms.

Conclusion. The staff has concluded that the applicant's evaluation of LR-ISG-2012-01 has not resulted in any changes to the staff's previously issued SER Section 3.0.3.2.12.

3.1 Aging Management of Reactor Vessel, Internals, Reactor Coolant System and Reactor Coolant Pressure Boundary, and Steam Generators

3.1.1 Summary of Technical Information in the Application

The staff does not have any changes or updates to this section of the SER.

Table 3.1-1 Revisions to SER Table 3.1-1

Component Group (GALL Report Item No.)	Aging Effect/ Mechanism	AMP in GALL Report	Further Evaluation in GALL Report	AMP in LRA, Supplements, or Amendments	Staff Evaluation
Steel with stainless steel or nickel-alloy cladding primary side components; SG upper and lower heads, tubesheets and tube-to-tube sheet welds (3.1.1-35)	Cracking due to SCC and PWSCC	ISI (IWB, IWC, and IWD) and Water Chemistry and for nickel alloy, comply with applicable NRC Orders and provide a commitment in the USAR supplement to implement applicable Bulletins and GLs and staff- accepted industry guidelines.	No, but applicant commitment needs to be confirmed	ISI, PWR Water Chemistry, Steam Generator Tube Integrity	Consistent with GALL Report (see SER Section 3.1.2.2.16(1))
Chrome plated steel, stainless steel, nickel-alloy SG anti-vibration bars exposed to secondary feedwater/steam (3.1.1-74)	Cracking due to SCC, loss of material due to crevice corrosion and fretting	Steam Generator Tube Integrity and Water Chemistry	No	Loss of material and cracking	Consistent with the GALL Report (See revisions to SER Section 3.1.2.1.1)

3.1.2 Staff Evaluation

3.1.2.1 AMR Results Consistent with the GALL Report

The staff does not have any changes or updates to this section of the SER.

3.1.2.1.1 AMR Results Identified as Not Applicable

The applicant submitted an annual update to the LRA by letter dated June 23, 2014, as amended by letter dated September 16, 2014. In its LRA update, the applicant revised LRA Table 3.1-1, AMR item 3.1.1-74, to address cracking due to SCC and loss of material due to crevice corrosion and fretting as applicable aging effects for SG secondary side components. These components are fabricated of chrome plated steel, stainless steel, and nickel alloy and are exposed to secondary water and steam. Specifically, the applicant updated LRA Table 3.1.2-4 to manage cracking and loss of material for SG tube support rods (also called tie rods) using the Steam Generator Tube Integrity Program and Water Chemistry Program.

In its review, the staff finds the applicant's update regarding AMR item 3.1.1-74 acceptable because the Steam Generator Tube Integrity Program includes visual inspections of SG secondary side components to confirm the integrity of these components and the Water Chemistry Program monitors and controls secondary-side water chemistry to minimize environmental effects on cracking and loss of material in these components, consistent with the GALL Report.

3.1.2.1.5 Aging Management of Reactor Vessel Internals (RVI) Vent Valve Assembly Components

LRA Table 3.1.1, as amended in a letter dated March 9, 2012 (ADAMS Accession No. ML12094A383), includes AMR item 3.1.1-11 which addresses CASS RVI vent valve upper and lower retaining rings exposed to a borated reactor coolant with neutron fluence environment. In this AMR item, the applicant states that the vent valve upper and lower retaining rings will be managed for reduction of fracture toughness due to thermal aging embrittlement. For this AMR item that cites generic note E, the LRA credits the PWR Reactor Vessel Internals Program to manage the applicable aging effect.

GALL Report AMP XI.M16A, PWR Vessel Internals, as updated in NRC LR-ISG-2011-04, recommends use of condition monitoring activities to manage reduction of fracture toughness in the vent valve upper and lower retaining rings of B&W-designed PWRs, including Davis-Besse.[2] The staff noted that, in general, this program applies either volumetric or visual inspection methods to manage aging effects of cracking, loss of material, and changes in dimension that may be occurring in the RVI components, and to manage loss of preload of bolts, keys, or other fasteners in bolted or fastened RVI assemblies. The staff also noted that this program uses the condition monitoring methods that are applied for detection cracking as an indirect basis for detecting and managing those changes that may be occurring in a component's material properties, including reduction of the fracture toughness property for a component's specific material of fabrication.

In the applicant's letter of April 21, 2015, the applicant identified a deviation with respect to the design of the vent valve assemblies at Davis-Besse when compared to the generic vent valve assembly design that was evaluated in the MRP-227-A and MRP-189, Revision 1 reports. However, in this letter, the applicant stated that the I&E methods in MRP-227-A for

[2] The condition monitoring bases for inspecting the upper and lower retaining rings in the vent valve assemblies are given in EPRI TR No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (MRP-227-A), December 2011. Henceforth, this report will be referred to as MRP-227-A or the MRP-227-A report. Refer to ADAMS Accession No. ML12017A193 for the transmittal letter from the EPRI-MRP that submitted the report to the NRC Document Control Desk and ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195, and ML12017A199 for the sections in the MRP-227-A report.

B&W-designed vent valve assembly components would not need to be further adjusted for aging management because the CLB had existing program surveillance testing and inspection requirements in the TS that would accomplish adequate aging management of those specific vent valve assembly components (including vent valve locking devices) not assessed in the MRP-189, Revision 1, report.

The staff noted that, for RVI vent valve assembly components in B&W-designed reactors like Davis-Besse, implementation of the MRP-227-A guidelines only applies visual VT-3 inspections to the upper and lower retaining rings in the vent valve assemblies and does not implement inspections of the valve bodies or any other miscellaneous parts of the vent valve assemblies. However, the staff also noted that the CLB includes TS 5.5.4, "Reactor Vessel Internals Vent Valve Program," which requires the applicant to implement the following surveillance inspection and testing requirements to the vent valve assemblies once every 24 months:

(a) Verify by visual inspection that the valve body and disc exhibits no abnormal degradation.

(b) Verify the valve is not stuck in an open position.

(c) Verify by manual actuation that the valve is fully open when a force of less than or equal to 400 lbs is applied vertically upward.

However, the staff noted that the applicant did not credit the TS 5.5.4 surveillance requirements as additional condition monitoring and performance monitoring activities for those vent valve bodies and parts that are not scheduled for inspection in accordance with MRP-227-A. Thus, in its review of AMR item 3.1.1-11, the staff did not find the applicant's proposal to manage aging using the PWR Reactor Vessel Internals Program to be acceptable because the AMR basis did not credit the TS 5.5.4 requirements as additional condition monitoring and performance monitoring activities for the vent valve assembly components. Therefore the deviation with respect to the assessment of vent valve assembly components in MRP-189, Revision 1, had yet to be resolved with respect to the AMR in AMR item 3.1.1-11.

The staff discussed this matter with the applicant in a teleconference on May 19, 2015. To address this issue, the applicant submitted an amended version of AMR item 3.1.1-11 for the vent valve assembly components in a letter dated June 5, 2015. In the amended version of the AMR item 3.1.1-11, the applicant amended the AMR item to add in a plant-specific AMR item note that credits the applicable requirements of TS 5.5.4 as being additional condition monitoring and performance monitoring activities that apply to the vent valve assembly components, including the vent valve bodies and miscellaneous vent valve parts. This is in addition to the use of the PWR Reactor Vessel Internals Program for implementing inspections of the upper and lower vent valve assembly retaining rings during the period of extended operation.

The staff reviewed the changes to AMR item 3.1.1-11. The staff noted that the addition of the TS 5.5.4 requirements to AMR item 3.1.1-11 is consistent with the CLB for Davis-Besse. Furthermore, in Section 3.0.3.3.6 of this SSER, the staff provides its supplemental evaluation of the applicant's PWR Reactor Vessel Internal Program, and response to A/LAI #2, which was issued on implementation of the MRP-227-A report. In this evaluation, the staff provides its basis for concluding that the PWR reactor Vessel Internals Program, when taken into account with the surveillance inspection and testing requirements in TS 5.5.4, provides for adequate aging management of those aging effects that apply to the vent valve assembly components at

Davis-Besse. Therefore, based on this review, the staff finds that the amended version of AMR item 3.1.1-1 is acceptable because: (a) the applicant has credited both the augmented criteria in MRP-227-A for inspecting vent valve upper and lower retaining rings (as implemented in accordance with the PWR Reactor Vessel Internals Program) and the surveillance inspection and testing requirements in TS 5.5.4 as the bases for managing aging in the vent valve assembly components, (b) the staff has determined the collective set of condition monitoring and performance monitoring criteria for vent valve assembly components in MRP-227-A and TS 5.5.4 provide for adequate aging of vent valve assembly components, and (c) the staff has determined that the applicant's bases for managing aging of vent valve assembly components is consistent with the CLB. The deviation from the assessment basis in MRP-189, Revision 1, for vent valve assembly components is resolved both with respect to the applicant's PWR Reactor Vessel Internals Program and the AMR bases in AMR item 3.1.1-11 of the LRA.

3.1.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation Is Recommended

3.1.2.2.16 Cracking Due to Stress Corrosion Cracking and Primary Water Stress Corrosion Cracking

Item1. In LRA Table 3.1.2-4, as amended by letter dated June 23, 2014, item 3.1.1-35 addresses nickel Alloy 690 primary side steam generator upper and lower heads, tubesheets, and tube-to-tubesheet welds exposed to borated reactor coolant ,which are being managed for cracking due to primary water stress corrosion cracking (PWSCC). For the AMR item that cites generic note E, the LRA credits the Inservice Inspection, PWR Water Chemistry, and Steam Generator Tube Integrity Programs to manage the aging effect. The GALL Report recommends GALL Report AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components, and GALL Report AMP XI.M2, "Water Chemistry," to ensure that these aging effects are adequately managed.

For the item associated with generic note E, GALL Report AMP XI.M1 recommends periodic visual, surface, and/or volumetric examination and GALL Report AMP XI.M2 recommends using water chemistry controls to manage aging. In its review of components associated with item 3.1.1-35 for which the applicant cited generic note E, the staff noted that the Inservice Inspection Program proposes to manage nickel Alloy 690 primary side steam generator upper and lower heads, tubesheets, and tube-to-tubesheet welds by performing periodic visual, surface, or volumetric examination and leakage testing. The PWR Water Chemistry Program proposes to manage aging effects by using water chemistry controls. In addition, the Steam Generator Tube Integrity Program proposes to manage aging through a combination of prevention, inspection, evaluation, removal of tubes from service (plugged) and leakage monitoring. The applicant indicated that preventive measures used per the Steam Generator Tube Integrity Program include primary-side and secondary-side water chemistry monitoring and control, and foreign material exclusion requirements. The staff notes that there has been no cracking due to PWSCC identified in tube-to-tubesheet welds fabricated of Alloy 690 material in the U.S. operating fleet; therefore, the staff finds that performing visual, surface, or volumetric examination is not required for managing cracking due to PWSCC of the tube-to-tubesheet welds.

The staff's evaluation of the applicant's Inservice Inspection, PWR Water Chemistry, and Steam Generator Tube Integrity Programs are documented in SER Sections 3.0.3.1.12, 3.0.3.1.15, and 3.0.3.1.18. In its review of components associated with item 3.1.1-35, the staff finds the applicant's proposal to manage aging using the Steam Generator Tube Integrity Program

acceptable because water chemistry control and NDE techniques may be used to control and detect cracking due to PWSCC in nickel Alloy 690 primary-side tube-to-tubesheet welds. Again, staff notes that cracking due to PWSCC has not been identified in tube-to-tubesheet welds fabricated of Alloy 690 material in the U.S. operating fleet.

The staff concludes that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.3 AMR Results Not Consistent with or Not Addressed in the GALL Report

3.1.2.3.5 Steam Generators—Primary Side: Tube—Aging Management Evaluation

In LRA Table 3.1.2-4, the applicant stated that the thermally-treated nickel Alloy 690 primary-side SG tubes exposed to treated water are being managed for reduction of heat transfer by the Steam Generator Tube Integrity Program. The AMR item cites generic note H.

The staff noted that this material and environment combination is identified in the GALL Report, item IV.D1.R-47, which addresses nickel alloy SG tubes exposed to secondary feedwater (treated water) or steam and recommends GALL Report AMP XI.M2, "Steam Generators," and AMP XI.M19, "Water Chemistry," to manage cracking; however, the applicant has identified this additional aging effect. The staff also noted that the applicant addressed the GALL Report identified aging effects for this component, material, and environment combination in AMR items in LRA Table 3.1.2-4.

The staff's evaluation of the applicant's Steam Generator Tube Integrity Program is documented in SER Section 3.0.3.1.18. The staff finds the applicant's proposal to manage aging using the Steam Generator Tube Integrity Program acceptable because performing periodic cleaning of the SG secondary side internals, including tubes and tubesheet, will remove accumulated deposits from the SG, thus ensuring that the ability of the tubes to transfer heat is not hindered.

On the basis of its review, the staff finds that the applicant has appropriately evaluated the AMR results of material, environment, AERM, and AMP combinations not addressed in the GALL Report. The staff finds that the applicant has demonstrated that the effects of aging for these components will be adequately managed, so that their intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.2 Aging Management of Engineered Safety Features

3.2.1 Summary of Technical Information in the Application

The staff does not have any changes or updates to this section of the SER.

3.2.2 Staff Evaluation

The staff does not have any changes or updates to this section of the SER.

3.2.2.1 AMR Results Consistent with the GALL Report

3.2.2.1.7 Loss of Material Due to Pitting and Crevice Corrosion

LRA Table 3.2.1, item 3.2.1-49, addresses stainless steel piping, piping components, piping elements, and tanks exposed to treated borated water, which will be managed for loss of material due to pitting and crevice corrosion by the One-Time Inspection Program. The staff's evaluation of item 3.2.1-49 is documented in SER Section 3.2.2.1.7. By letter dated February 19, 2014, the applicant amended LRA Table 3.2.1, item 3.2.1-49, and LRA Table 3.2.2-4 to state that loss of material due to pitting and crevice corrosion on the internal surfaces of the stainless steel BWST will be managed by the Aboveground Steel Tanks Inspection Program. The revised AMR item associated with the BWST cites generic note E.

The staff's evaluation of the applicant's revised Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1. The staff finds the applicant's proposal to manage the internal surfaces of the BWST using the Aboveground Steel Tanks Inspection Program acceptable because the program includes periodic examinations of the internal surfaces of in-scope tanks, which are capable of detecting the applicable aging effects.

The staff concludes that the applicant has demonstrated that the effects of aging for this component will be adequately managed so that the intended function(s) will remain consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.2.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation Is Recommended

3.2.2.2.3 Loss of Material Due to Pitting and Crevice Corrosion

Items 1-5. The staff does not have any changes or updates to this section of the SER.

Item 6. By letter dated February 19, 2014, the applicant amended LRA Section 3.2.2.2.3, item 6, to also address the BWST. As amended, the LRA states that loss of material due to pitting and crevice corrosion for the stainless steel BWST will be managed by the Aboveground Steel Tanks Inspection Program. The revised AMR item 3.2.1-08 associated with the BWST cites generic note E.

The staff's evaluation of the applicant's revised Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1. The staff finds the applicant's proposal to manage the internal surfaces of the BWST exposed to condensation using the Aboveground Steel Tanks Inspection Program acceptable because the program includes periodic examinations of the internal surfaces of in-scope tanks, which are capable of detecting the applicable aging effects.

Based on the program identified, the staff concludes that the applicant's program meets SRP-LR Section 3.2.2.2.3, item 6, criteria. For those items that apply to LRA Section 3.2.2.2.3.6, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.2.2.3 AMR Results Not Consistent with or Not Addressed in the GALL Report

3.2.2.3.4 Engineered Safety Features Systems—Decay Heat Removal and Low-Pressure Injection System—Aging Management Review Results—LRA Table 3.2.2-4

In LRA Table 3.2.2-4, revised by letter dated September 16, 2011, the applicant stated that the external surfaces of the BWST exposed to outdoor air will be managed for loss of material by the External Surfaces Monitoring Program. The AMR item cites generic note G. By letter dated February 19, 2014, the applicant amended LRA Table 3.2.2-4 to add plant-specific note 0215, for the LRA Table 3.2.2-4 item that cites the BWST. Plant-specific note 0215 states that, for outdoor insulated components, the External Surfaces Monitoring Program manages corrosion under insulation. By letter dated July 29, 2014, the applicant further revised LRA Table 3.2.2-4. The revised AMR item related to the BWST cites generic note G. The applicant also deleted plant-specific note 0215, and cited plant-specific note 0217. Plant-specific note 0217 states that, for the exterior surfaces of the BWST, corrosion under insulation and cracking will be managed by the Aboveground Steel Tanks Inspection Program.

The staff's evaluation of the applicant's revised Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1. Based on its review of the revisions associated with the applicant's Aboveground Steel Tanks Inspection Program, the staff finds the applicant's revised program acceptable because it will include baseline bare metal surface examinations under the insulation of the BWST that are capable of detecting loss of material and cracking, with subsequent periodic activities to either inspect the external surfaces of the insulation for damage that would allow moisture penetration or continue with bare metal inspections.

In LRA Table 3.2.2-4, the applicant stated that stainless steel bolting exposed to outdoor air will be managed for loss of material by the Bolting Integrity Program. The AMR items cite generic note F. The applicant also stated that the external surfaces of stainless piping exposed to outdoor air will be managed for loss of material by the External Surfaces Monitoring Program. The AMR items cite generic note G. By letter dated February 19, 2014, the applicant amended LRA Table 3.2.2-4, to add plant-specific note 0215 for these items. Plant-specific note 0215 states that, for outdoor insulated components, the External Surfaces Monitoring Program manages corrosion under insulation.

The staff's evaluation of the applicant's External Surfaces Monitoring Program is documented in SSER Section 3.0.3.2.5. Based on its review of the revisions associated with the applicant's External Surfaces Monitoring Program, the staff finds the applicant's revised program acceptable because it will include baseline bare metal surface examinations under the insulation that are capable of detecting loss of material with subsequent periodic activities to either inspect the external surfaces of the insulation for damage that would allow moisture penetration or continue with bare metal inspections. The staff's evaluation of the Bolting Integrity Program is documented in SER Section 3.0.3.2.2. In its review of the Bolting Integrity Program, the staff determined that the visual examinations performed under the program are capable of detecting loss of material.

By letter dated February 19, 2014, the applicant amended LRA Table 3.2.2-4 to include a new item to address loss of material for the BWST exposed to concrete (external) that will be managed by the Aboveground Steel Tanks Program. The new item cites generic note G and plant-specific note 0216. Plant-specific note 0216 states that, for tanks supported on earthen or concrete foundations, corrosion may occur at inaccessible locations, such as the tank bottom.

The staff's evaluation of the applicant's revised Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1. The staff finds the applicant's proposal to manage the bottom of the BWST using the Aboveground Steel Tanks Inspection Program acceptable because the program includes periodic examinations of the tank's bottom, which are capable of detecting the applicable aging effects.

On the basis of its review, the staff finds that the applicant has appropriately evaluated the AMR results of material, environment, AERM, and AMP combinations not addressed in the GALL Report. The staff finds that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that their intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.2.3 Conclusion

The staff does not have any changes or updates to this section of the SER.

3.3 Aging Management of Auxiliary Systems

Revisions to SER Table 3.3-1
Staff Evaluation for Auxiliary System Components in the GALL Report

Component Group (GALL Report Item No.)	Aging Effect/ Mechanism	AMP in GALL Report	Further Evaluation in GALL Report	AMP in LRA, Supplements, or Amendments	Staff Evaluation
Stainless steel high-pressure pump casing in PWR chemical and volume control system (3.3.1-9)	Cracking due to SCC and cyclic loading	Water Chemistry and a plant-specific verification program. The AMP is to be augmented by verifying the absence of cracking due to SCC and cyclic loading. A plant-specific AMP is to be evaluated.	Yes	Not Applicable	Not Applicable to Davis-Besse (see SER Section 3.3.2.2.4(3))

3.3.1 Summary of Technical Information in the Application

The staff does not have any changes or updates to this section of the SER.

3.3.2 Staff Evaluation

3.3.2.1 AMR Results Consistent with the GALL Report

3.3.2.1.6 Loss of Material Due to Pitting and Crevice Corrosion

LRA Table 3.3.1, item 3.3.1-25, addresses copper-alloy bolting, piping, and valve bodies exposed to condensation (external), which will be managed for loss of material due to corrosion by the Bolting Integrity (for bolting) and the External Surfaces Monitoring Programs (for piping). LRA Table 3.3.1, item 3.3.1-27, addresses stainless steel piping components exposed to condensation (external), which will be managed for loss of material due to corrosion by the External Surfaces Monitoring Programs. The LRA items cite generic note E for these components. By letter dated February 19, 2014, the applicant amended these items to also cite plant-specific note 0345. Plant-specific note 0345 states that for indoor insulated components exposed to condensation (because the in-scope component is operated below the dew point), corrosion under insulation is managed by the External Surfaces Monitoring Program.

The staff's evaluation of the applicant's revised External Surfaces Monitoring Program is documented in SSER Section 3.0.3.2.5. Based on its review of the revisions associated with the applicant's External Surfaces Monitoring Program, the staff finds the applicant's revised program acceptable because it will include baseline bare metal surface examinations under the insulation that are capable of detecting loss of material and cracking, with subsequent periodic activities to either inspect the external surfaces of the insulation for damage that would allow moisture penetration or continue with bare metal inspections. The staff's evaluation of the Bolting Integrity Program is documented in SER Section 3.0.3.2.2. In its review of the Bolting Integrity Program, the staff determined that the visual examinations performed under the program are capable of detecting loss of material.

The staff concludes that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.3.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation Is Recommended

3.3.2.2.4 Cracking Due to Stress Corrosion Cracking and Cyclic Loading

Item 3. LRA Section 3.3.2.2.4.3, associated with LRA Table 3.3.1, item 3.3.1-9, addresses cracking due to SCC and cyclic loading in stainless steel high-pressure pump casings exposed to treated water in the PWR chemical and volume control system. As documented in SER Section 3.3.2.2.4, item 3, the applicant stated that the makeup pumps in the makeup and purification system are susceptible to cracking due to cyclic loading only and that this aging effect would be managed by the PWR Water Chemistry and One-Time Inspection Programs.

However, by letter dated June 23, 2014, the applicant amended LRA Section 3.3.2.2.4.3 to state that cyclic loading is not identified as an aging effect and, as a result, this item is not applicable. The applicant also amended Commitment No. 13 associated with the One-Time Inspection Program to delete the reference to the inspection of the makeup pump casings. The applicant stated that the design of the pumps is such that their external casings, which have a pressure boundary function, are not subject to cyclic loading because they receive fluid discharged from the pumps' inner casings at a steady pressure, flow rate, and temperature. The applicant also

stated that the pump inner casings, which do not have a license renewal function, bear the mechanical loading associated with the 12 stages of the pumps. The applicant further stated that one of the two pumps operates continuously during normal operation, at which time it is exposed to a treated water environment with a relatively constant temperature of 120 ºF, and thus thermal cyclic loading is not a concern.

The staff evaluated the applicant's claim and finds it acceptable because the portion of the pump casing with a license renewal function (i.e., the external casing) is not subject to the pressure increases developed by the 12 stages of the pump that could promote cracking due to cyclic loading. In addition, because one of the makeup pumps operates continuously during normal operation, the external casing of the operating pump is exposed to a relatively constant internal water pressure and temperature. As a result, the design and usage of the pumps minimize the exposure of their external casings to cyclic loading from both mechanical and thermal sources.

3.3.2.3 AMR Results Not Consistent with or Not Addressed in the GALL Report

3.3.2.3.1 Auxiliary Systems—Auxiliary Building HVAC System—Aging Management Review Results—LRA Table 3.3.2.1

LRA Table 3.3.2-1, states that copper-alloy piping and copper alloy with greater than 15 percent zinc valve bodies exposed to outdoor air (external) will be managed for loss of material by the External Surfaces Monitoring Program. The AMR items cite generic note G. By letter dated February 19, 2014, the applicant amended LRA Table 3.3.2-1 to add plant-specific note 0344 for these items. Plant-specific note 0344 states that, for outdoor insulated components, the External Surfaces Monitoring Program manages corrosion under insulation.

The staff's evaluation of the applicant's External Surfaces Monitoring Program is documented in SSER Section 3.0.3.2.5. Based on its review of the revisions associated with the applicant's External Surfaces Monitoring Program, the staff finds the applicant's revised program acceptable because it will include baseline bare metal surface examinations under the insulation that are capable of detecting loss of material and cracking, with subsequent periodic activities to either inspect the external surfaces of the insulation for damage that would allow moisture penetration or continue with bare metal inspections.

By letter dated February 19, 2014, the applicant amended LRA Table 3.3.2-1 by adding steel bolting exposed to condensation, which will be managed for cracking, loss of material, and loss of preload by the Bolting Integrity Program. The AMR items cite generic notes B and H. The generic note H item also cites plant-specific note 0345. Plant-specific note 0345 states that, for indoor insulated components exposed to condensation (because the in-scope component is operated below the dew point), corrosion under insulation is managed by the External Surfaces Monitoring Program.

The staff's evaluation of the applicant's External Surfaces Monitoring Program is documented in SSER Section 3.0.3.2.5. Based on its review of the revisions associated with the applicant's External Surfaces Monitoring Program, the staff finds the applicant's revised program acceptable because it will include baseline bare metal surface examinations under the insulation that are capable of detecting loss of material and cracking, with subsequent periodic activities to either inspect the external surfaces of the insulation for damage that would allow moisture penetration or continue with bare metal inspections. The staff's evaluation of the Bolting Integrity Program is documented in SER Section 3.0.3.2.2. In its review of the Bolting Integrity

Program, the staff determined that the visual examinations performed under the program are capable of detecting cracking, loss of material, and loss of preload.

On the basis of its review, the staff concluded that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that their intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.3.2.3.6 Auxiliary Systems—Circulating Water System—Aging Management Review Results—LRA Table 3.3.2-6

Carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil.

As amended by letter dated January 31, 2014, LRA Tables 3.3.2-6, 3.3.2-12, 3.3.2-14, 3.3.2-15, 3.3.2-26, and 3.3.2-30 state that carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil will be managed for loss of coating integrity by the Service Level III Coatings and Linings Monitoring Program. The AMR items cite generic note H.

The staff's evaluation of the applicant's Service Level III Coatings and Linings Monitoring Program is documented in SER Section 3.0.3.3.10. The staff finds the applicant's proposal to manage loss of coating integrity using the Service Level III Coatings and Linings Monitoring Program acceptable because the program includes periodic visual inspections that are capable of detecting degraded coatings and followup physical testing, repair, and replacement activities when inspection results do not meet acceptance criteria.

On the basis of its review, the staff concludes that the applicant has demonstrated that the effects of aging for these items will be adequately managed so that their intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.3.2.3.12 Auxiliary Systems—Emergency Diesel Generators System—Aging Management Review Results—LRA Table 3.3.2-12

Carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil.

The staff's evaluation for carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil, which will be managed for loss of coating integrity by the Service Level III Coatings and Linings Monitoring Program and is associated with generic note H, is documented in SER Section 3.3.2.3.6.

3.3.2.3.14 Auxiliary Systems—Fire Protection System—Aging Management Review Results—LRA Table 3.3.2-14

Carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil.

The staff's evaluation for carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil, which will be managed for loss

of coating integrity by the Service Level III Coatings and Linings Monitoring Program and is associated with generic note H, is documented in SER Section 3.3.2.3.6.

3.3.2.3.15 Auxiliary Systems—Fuel Oil System—Aging Management Review Results—LRA Table 3.3.2-15

By letter dated February 19, 2014, the applicant amended LRA Table 3.3.2-15 to include a new item to address loss of material for steel tanks exposed to concrete (external). The new item cites generic note G and plant-specific note 0346. Plant-specific note 0346 states that, for storage tanks supported on earthen or concrete foundations, corrosion may occur at inaccessible locations.

The staff's evaluation of the applicant's revised Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1. The staff finds the applicant's proposal to manage the bottom of the tanks using the Aboveground Steel Tanks Inspection Program acceptable because the program includes periodic examinations of the tank's bottom, which are capable of detecting the applicable aging effects.

Carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil.

The staff's evaluation for carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil, which will be managed for loss of coating integrity by the Service Level III Coatings and Linings Monitoring Program and is associated with generic note H, is documented in SER Section 3.3.2.3.6.

3.3.2.3.26 Auxiliary Systems—Service Water System—Aging Management Review Results—LRA Table 3.3.2-26

Carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil.

The staff's evaluation for carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil, which will be managed for loss of coating integrity by the Service Level III Coatings and Linings Monitoring Program and is associated with generic note H, is documented in SER Section 3.3.2.3.6.

3.3.2.3.30 Auxiliary Systems—Station Blackout Diesel Generator System—Aging Management Review Results—LRA Table 3.3.2-30

Carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil.

The staff's evaluation for carbon steel (coated) and gray cast iron (coated) piping, piping components, and tanks exposed to raw water and lubricating oil, which will be managed for loss of coating integrity by the Service Level III Coatings and Linings Monitoring Program and is associated with generic note H, is documented in SER Section 3.3.2.3.6.

On the basis of its review, the staff finds that the applicant has appropriately evaluated the AMR results of material, environment, AERM, and AMP combinations not addressed in the GALL Report. The staff finds that the applicant has demonstrated that the effects of aging for

these components will be adequately managed so that their intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.3.3 Conclusion

The staff does not have any changes or updates to this section of the SER.

3.4 Aging Management of Steam and Power Conversion Systems

3.4.1 Summary of Technical Information in the Application

The staff does not have any changes or updates to this section of the SER.

3.4.2 Staff Evaluation

3.4.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation Is Recommended

3.4.2.2.2 Loss of Material Due to General, Pitting, and Crevice Corrosion

Item 1. LRA Section 3.4.2.2.2.1 is associated with LRA Table 3.4.1, item 3.4.1-06 and addresses steel and stainless steel tanks exposed to treated water (internal), which will be managed for loss of material by the One-Time Inspection and the PWR Water Chemistry Programs. LRA Table 3.4.2-2 also states that the interior steel surfaces of the condensate storage tank exposed to moist air (internal) will be managed for loss of material by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program. By letter dated February 19, 2014, the applicant revised LRA Table 3.4.2-2. As part of the revision, the applicant introduced a new item to address loss of material for steel tanks exposed to concrete (external). The new AMR item cites generic note G and plant-specific note 0416. The applicant also amended LRA Table 3.4.2-2, items associated with the condensate tank to remove the One-Time Inspection and PWR Water Chemistry Programs. The applicant stated that it will use the Aboveground Steel Tanks Inspection Program to manage for loss of material for the steel surfaces of the condensate tank exposed to treated water (internal) and moist air (internal). These AMR items cite generic notes E and G.

The staff's evaluation of the applicant's revised Aboveground Steel Tanks Inspection Program is documented in SSER Section 3.0.3.2.1. The staff finds the applicant's proposal to manage the internal surfaces of the tanks exposed to condensation using the Aboveground Steel Tanks Inspection Program acceptable because the program includes periodic examinations of the internal surfaces of in-scope tanks, which are capable of detecting the applicable aging effects. The staff also finds the applicant's proposal to manage the bottom of the BWST using the Aboveground Steel Tanks Inspection Program acceptable because the program includes periodic examinations of the tank's bottom, which are capable of detecting the applicable aging effects.

Based on the program identified, the staff concludes that the applicant's program meets SRP-LR Section 3.4.2.2.2, item 1 criteria. For those items that apply to LRA

Section 3.4.2.2.2.1, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.4.2.3 AMR Results Not Consistent with or Not Addressed in the GALL Report

The staff does not have any changes or updates to this section of the SER.

3.4.3 Conclusion

The staff does not have any changes or updates to this section of the SER.

3.5 Aging Management of Structures and Structural Components

The staff does not have any changes or updates to this section of the SER.

3.6 Aging Management of Electrical and Instrumentation and Controls

The staff does not have any changes or updates to this section of the SER.

SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

There are no changes or updates to this section of the SER.

4.1.1 Summary of Technical Information in the Application

The applicant provided additional information related to the time-limited aging analyses (TLAAs) after issuance of the SER. The additional information is added and evaluated in Section 4.1.2.1.

4.1.2 Staff Evaluation

There are no changes or updates to this section of the SER, other than the amendments to LRA Section 4.1.2.1 that were identified by the applicant and are evaluated in the following subsections.

4.1.2.1 Evaluation of the Applicant's Identification of Time-Limited Aging Analyses

Other than the changes identified for the evaluations in Sections 4.1.2.1.3 and 4.1.2.1.4 that follow, the staff evaluation of the applicant's bases for identifying TLAAs in the LRA are as documented in SER Section 4.1.2.1. The evaluations in Sections 4.1.2.1.3 and 4.1.2.1.4 that follow supersede those that were provided in SER Sections 4.1.2.1.3 and 4.1.2.1.4.

4.1.2.1.3 LRA Update of June 23, 2014—Changes to Metal Fatigue TLAAs for the Once-Through Steam Generators

On June 23, 2014, the applicant submitted an LRA update in order to comply with the LRA update requirements in 10 CFR 54.21(b). As part of this submittal, the applicant stated that the original once-through steam generators (OTSGs) in the Davis-Besse nuclear plant were replaced in the Cycle 18 (Spring 2014) refueling outage. Based on this plant modification, the applicant amended LRA Table 4.1-1 and LRA Section 4.3.2.2.6 to propose changes to the TLAAs on metal fatigue of the OTSGs and auxiliary feedwater nozzles of the facility. The staff's evaluations of the amendments of LRA Table 4.1-1 and LRA Section 4.3.2.2.6 are given in Section 4.3.2.2 of this SSER.

4.1.2.1.4 LRA Update of June 23, 2014—Changes to the Flaw Evaluation TLAA for the Once-Through Steam Generators

On June 23, 2014, the applicant submitted an LRA update in order to comply with the LRA update requirements in 10 CFR 54.21(b). As part of this submittal, the applicant stated that the original OTSGs in Davis-Besse were replaced in the Cycle 18 (Spring 2014) refueling outage. The applicant stated that the flaw evaluation TLAA previously given in LRA Section 4.7.5.2, "OTSG 1-2 Flaw Evaluations," is no longer applicable to the current licensing basis (CLB).

Therefore, the applicant deleted LRA Section 4.7.5.2 from the scope of the LRA, including the reference to LRA Section 4.7.5.2 in LRA Table 4.1-1. The staff's evaluation of the applicant's basis for deleting LRA Section 4.7.5.2 is contained in Section 4.7.5.2 of this SSER.

4.1.2.2 Evaluation of the Applicant's Identification of Those Exemptions in the CLB That Are Based on TLAAs

There are no changes or updates to this section of the SER. Therefore, the evaluation in SER Section 4.1.2.2 remains valid as previously written.

4.1.3 Conclusion

There are no changes or updates to this section of the SER. Therefore, the conclusion in SER Section 4.1.3 remains valid as previously written.

4.2 Reactor Vessel Neutron Embrittlement

4.2.7 Reduction in Fracture Toughness of Reactor Vessel Internals

4.2.7.1 Summary of Technical Information in the Application

LRA Section 4.2.7 describes the applicant's TLAA for the reduction in fracture toughness of the RVI. The applicant cited USAR Appendix 4A, which describes the detailed stress analysis of the internals under accident conditions for the current term of operation. According to the applicant, the analysis shows that the internals will not fail because the stresses are within established limits. The applicant stated that the effect of irradiation on the mechanical properties and deformation limits for the RVI was also evaluated for the current term of operation. The applicant also stated that the aforementioned analysis concluded that the RVI will have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation will not adversely affect deformation limits.

The applicant stated that the impact of the measurement uncertainty recapture (MUR) power uprate on the structural integrity of the RVI components was evaluated. The applicant concluded that the temperature changes due to the MUR power uprate are bounded by those used in the existing analyses. As part of the MUR power uprate, the applicant stated that, "[a]s appropriate, FENOC commits to incorporate recommendations from MRP [Materials Reliability Program] inspection guidelines into the RVI program at Davis-Besse Nuclear Power Station, Unit No. 1."

The applicant stated that this TLAA will be managed during the period of extended operation through the PWR RVI Program.

Based on the information above, the applicant concluded that the effects of neutron embrittlement on the reduction in fracture toughness for the RVI will be appropriately managed during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

By letter dated April 21, 2015, the applicant amended the LRA and submitted the inspection plan for the RVI components at Davis Besse in order to fulfill the conditions and criteria specified in LRA Commitment No. 15, which was included as a commitment in USAR

Supplement Table A-1, and stated that the applicant would submit the inspection plan for the RVI components to the NRC for review and approval by April 22, 2015. This RVIIP was included in the letter of April 21, 2015, and is documented in nonproprietary AREVA TR No. ANP-3920, Revision 1, "Reactor Vessel Internals Inspection Plan for Davis Besse Nuclear Plant Unit No. 1 – Licensing Report," dated March 2015.

In the RVIIP, the applicant responded to Applicant/Licensee Action Item (A/LAI) #8, Subitem 5, and confirmed that the CLB includes an evaluation of the potential drops in the ductile fracture toughness property of the components, as evaluated in B&W TR No. BAW-10008, Revision 1.

By letter dated May 20, 2015, the applicant amended USAR Supplement Table A-1 to include Commitment No. 54, which was linked to the USAR Supplement summary description for the PWR Reactor Vessel Internals Program (i.e., LRA Section A.1.32), and in which the applicant committed to submit an updated reduction of ductility fracture toughness to the NRC for review and approval by October 22, 2016.

4.2.7.2 Staff Evaluation

The evaluation in this section of the SSER supersedes the staff's previous evaluation of the reduction of ductility TLAA previously given in SER Section 4.2.7. The staff's evaluation of the applicant's basis for managing loss of fracture toughness in CASS RVI components is documented in SER Section 4.2.7, as supplemented by the evaluation in Section 3.0.3.3.6 of this SSER.

The staff reviewed LRA Section 4.2.7 on the reduction in fracture toughness for the RVI, as supplemented with information contained in the applicant's letter of April 21, 2015, to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of neutron embrittlement on the RVI will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA consistent with the review procedures in SRP-LR Section 4.7.3.1.3, which state that the applicant shall propose to manage the aging effects associated with the TLAA using an AMP in the same manner as described in the IPA in 10 CFR 54.21(a)(3). SRP-LR Section 4.7.3.1.3 also states that the applicable AMP is reviewed to verify that the effects of aging on the intended functions are adequately managed, consistent with the CLB for the period of extended operation.

Exposure of stainless steel RVI components to high-energy neutron radiation during the period of extended operation could result in a significant reduction in fracture toughness, depending on the material, irradiation temperature, and neutron fluence.

The staff determined that the reduction in fracture toughness of the stainless steel RVI is a TLAA that should be managed during the period of extended operation. The staff reviewed LRA Section 4.2.7 to determine if the applicant's TLAA of the reduction in fracture toughness for the RVI demonstrates that the effects of embrittlement on these components will be adequately managed during the period of extended operation, pursuant to 10 CFR 54.21(c)(1)(iii). The applicant appropriately referenced the Davis-Besse PWR Reactor Vessel Internals Program for managing the loss of fracture toughness for the stainless steel RVI components. The staff reviewed the Davis-Besse PWR Reactor Vessel Internals Program, as described in LRA Section B.2.32, and amended through LRA Amendment 15, and confirmed that it manages loss of fracture toughness due to neutron embrittlement of RVI components.

The staff noted that in the applicant's letter of April 21, 2015, the applicant submitted the RVIIP for the RVI components at Davis-Besse, which includes AREVA Nonproprietary TR Nos. ANP-3285, ANP-3359NP, and ANP-3920, Revision 1 (collectively given in ADAMS Accession Nos. ML15113B133 and ML15113B134). The staff noted that, in TR No. ANP-3920, Revision 1, the applicant responded to A/LAI #8, Subitem 5, and confirmed that the reduction of ductility fracture toughness analysis in TR No. BAW-10008, Revision 1, is a TLAA for the LRA. The staff noted that, in the letter of April 21, 2015, as supplemented by the letter of May 20, 2015, the applicant stated that the reduction of ductility analysis for the RVI components will be updated and projected to the end of the period of extended operation. The applicant also stated that the updated analysis will be submitted to the NRC for review and approval and that the commitment for the analysis provides an updated basis for accepting the reduction of ductility analysis in accordance with the requirement in 10 CFR 54.21(c)(1)(iii).

The staff verified that the letter of May 20, 2015, includes LRA Commitment No. 54, which was incorporated into USAR Supplement Table A-1 and committed to the following actions by the applicant:

> In response to MRP-227-A Applicant/Licensee Action Item 8, update and submit for NRC review and approval an evaluation for the period of extended operation regarding the effect of irradiation on the mechanical properties and deformation limits of the RV internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-10008, Part 1, Revision 1 supplemented by DB-1 [Davis-Besse] USAR Appendix 4A.

The staff noted that the applicant also committed to submitting the updated analysis to the NRC for review and approval by October 22, 2016 (i.e., at least 6 months prior to entering into the period of extended operation, which will begin on April 22, 2017) and linked these actions in Commitment No. 54 to the USAR Supplement (i.e., LRA Section A.1.32) for the PWR Reactor Vessel Internals Program. As explained in SER Section 3.0.3.3.6, the staff has determined that the applicant's PWR Reactor Vessel Internals Program provides an adequate means of using sampling-based inspections to detect flaw indications that may be indicative of a change in fracture toughness properties of the materials used to fabricate the components. This includes a reduction in fracture toughness that may be induced by a drop in the ductile behavior property of the materials or by neutron irradiation embrittlement or thermal embrittlement mechanisms. Therefore, the staff finds that the PWR Reactor Vessel Internals Program and the inclusion of Commitment No. 54 to the USAR Supplement provides an acceptable basis for approving the TLAA in accordance with 10 CFR 54.21(c)(1)(iii) because:

(a) The applicant will be projecting the reduction of ductility fracture toughness analysis in TR No. BAW-10008, Revision 1, to the end of the period of extended operation and will be submitting the analysis for NRC review and approval by at least 6 months prior to entering into the period of extended operation.

(b) The applicant will also be implementing a set of sample-based inspections of the RVI components in accordance with the Reactor Vessel Internals Program, which will provide adequate indications of any aging-related effects (including effects that may induce changes in applicable material properties of the RVI components) that may occur in the RVI components during the period of extended operation.

(c) The updated analysis, when coupled with the inspections performed in accordance with the Reactor Vessel Internals Program, provides an adequate basis for demonstrating

that loss of fracture toughness (including that induced by changes in ductility) will be adequately managed during the period of extended operations.

(d) This is an adequate basis for demonstrating that the implementation of the PWR Reactor Vessel Internals Program and Commitment No. 54 provides an acceptable basis for accepting the reduction of ductility TLAA in accordance with 10 CFR 54.21(c)(1)(iii) because: (a) implementation of the AMP and commitment will demonstrate that the impact of a reduction of fracture toughness (including ductile fracture toughness) on the intended functions of the RVI components will be adequately managed by inspection or analysis during the period of extended operation, and (b) this is consistent with the acceptance criteria in Section 4.7.2.1 of the SRP-LR.

The staff noted that the applicant's proposal to implement MRP-227-A as the basis for its plant-specific PWR Reactor Vessel Internals Program must address all of the plant-specific and vendor-specific action items associated with the plant-specific implementation of MRP-227-A, as specified in Section 4.2 of the staff's supplemental evaluation on MRP-227. The staff's evaluation of the PWR Reactor Vessel Internals Program is provided in SER Section 3.0.3.3.6, as supplemented by Section 3.0.3.3.6 of this SSER.

Based on the above evaluation, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the fracture toughness of the RVI will be adequately managed by the PWR Reactor Vessel Internals Program for the period of extended operation.

Additionally, the staff finds that the applicant's TLAA is consistent with SRP-LR Section 4.7.2.1 because the effects of aging on the intended function will be adequately managed for the period of extended operation.

USAR Supplement

LRA Section A.2.2.7 provides the USAR supplement summary description for the reduction in RVI fracture toughness TLAA evaluation. The staff noted that the USAR supplement summary description provided an adequate basis for accepting the TLAA, in accordance with 10 CFR 54.21(c)(1)(iii).

By letter dated May 20, 2015, the applicant amended USAR Supplement Table A-1 to include Commitment No. 54. The staff noted that this commitment was linked to USAR Supplement Section A.1.32 for the PWR Reactor Vessel Internals Program and that, in Commitment No. 54, the applicant committed to the following actions:

> In response to MRP-227-A Applicant/Licensee Action Item 8, update and submit for NRC review and approval an evaluation for the period of extended operation regarding the effect of irradiation on the mechanical properties and deformation limits of the RV internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-100081, Part 1, Revision 1 supplemented in DB-1 [Davis-Besse]USAR Appendix 4A.

The staff also noted that the applicant committed to submitting the updated analysis by October 22, 2016 (i.e., at least 6 months prior to entering into the period of extended operation). The staff's basis for accepting Commitment No. 54 and using the commitment to accept this TLAA, in accordance with 10 CFR 54.21(c)(1)(iii), is given in Section 4.2.7.2 of this SSER. Based on the information in the USAR supplement summary descriptions A.2.2.7 and A.1.32,

and the inclusion of LRA Commitment No. 54 in USAR Table A-1, the staff finds that the USAR supplement contains an adequate basis for accepting the reduction of ductility analysis for the RVI components, in accordance with the requirement in 10 CFR 54.21(c)(1)(iii).

Based on its review of the USAR supplement, the staff concludes that the information in the USAR supplement is an adequate summary description of the evaluation, as required by 10 CFR 54.21(d), and is consistent with SRP-LR Section 4.7.3.2.

4.2.7.3 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of reduction in fracture toughness on the integrity of RVI components will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.3 Metal Fatigue

4.3.2 Class 1 Fatigue

4.3.2.2 Class 1 Vessels, Pumps, and Major Components

4.3.2.2.1 Summary of Technical Information in the Application

On June 23, 2014, the applicant submitted an LRA update in order to comply with the LRA update requirements in 10 CFR 54.21(b). As part of this submittal, the applicant stated that the original OTSGs in Davis-Besse were replaced in the Cycle 18 (Spring 2014) refueling outage. Based on this plant modification, the applicant amended LRA Section 4.3.2.2.6 to propose changes to the TLAAs on metal fatigue of the OTSG primary and secondary side components, OTSG internal components, and auxiliary feedwater (AFW) headers, nozzles, and nozzle thermal sleeves of the facility.

In addition, by letter dated April 21, 2015, as amended in letters dated May 20, 2015, and June 5, 2015, the applicant amended the LRA to provide the RVIIP for Davis-Besse. As a result of these letters, the applicant amended its basis in LRA Section 4.3.2.2.6 for accepting the low-cycle fatigue TLAA for the RVI components, in accordance with 10 CFR 54.21(c)(1)(iii).

The following subsections reflect the changes to the applicant's low-cycle fatigue TLAAs for OTSG and RVI components. Otherwise, the summaries of the technical information for low-cycle fatigue TLAAs that apply to Class 1 vessels, pumps, and major equipment are as documented in SER Section 4.3.2.2.1.

Reactor Vessel Internals. LRA Section 4.3.2.2.2 describes the metal fatigue analysis for the RVI components, which include the plenum assembly and the core support assembly, consisting of the core support shield, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance specimen holder tubes. The applicant's metal fatigue TLAAs of RVIs are summarized below:

Low-Cycle Fatigue. LRA Section 4.3.2.2.2.1 states that the design of the RVIs meets the stress requirements of ASME Code Section III, but the design code did not require a fatigue analysis to be performed. The applicant stated that it performed fatigue analyses for the Alloy X-750 high-temperature annealed and aged condition heat treatment (HTH) bolts, which were designed to ASME Code Section III, to replace the majority of the vessel internals Alloy A-286 bolts. The applicant also stated that the CUFs for the Alloy X-750 HTH replacement bolts were based on the system design transients in LRA Table 4.3-1 and were found to be less than 1.0. The upper thermal shield bolts, flow distributor bolts, and guide block bolts have not been replaced. In the applicant's letters of May 20, 2015, and June 5, 2015, the applicant confirmed that the low-cycle fatigue analyses (i.e., CUF analyses) were performed for the UCB bolts, LCB bolts, and LTS bolts that were replaced in the plant design using bolts made from HTH X-750 materials. The applicant dispositioned the low-cycle fatigue TLAA of the RVIs, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation using a combination of the Fatigue Monitoring Program (LRA AMP B.2.18) and the PWR Reactor Vessel Internals Program (LRA AMP B.2.32).

Reactor Vessel Internals and Incore Instrument Nozzles Flow-Induced Vibration. LRA Section 4.3.2.2.2.2 discusses RVI metal fatigue and incore instrument nozzles subjected to FIV. The summary of technical information for the FIV fatigue TLAA is documented in SER Section 4.3.2.2.1. The applicant dispositioned the fatigue TLAA for the FIV of RVIs in accordance with 10 CFR 54.21(c)(1)(i), that the existing analysis remains valid for the period of extended operation.

Surveillance Capsule Holder Tubes Flow-Induced Vibration. LRA Section 4.3.2.2.2.3, as amended by letter dated June 17, 2011, discusses the high-cycle metal fatigue analysis (i.e., FIV cumulative usage factor [CUF analysis) of the surveillance capsule holder tubes. The summary of technical information for this TLAA is documented in SER Section 4.3.2.2.1. The applicant stated that it dispositioned the high-cycle metal fatigue TLAA for the surveillance capsule holder tubes in accordance with 10 CFR 54.21(c)(1)(ii) and that the analysis has been projected to the end of the period of extended operation.

Once-Through Steam Generators. LRA Section 4.3.2.2.6 states that the OTSG components exposed to RCS pressure are the hemispherical heads, the tubesheet, and the straight inconel tubes between the tubesheets. The applicant's metal fatigue TLAAs related to the OSTGs are separated into four parts, as summarized below.

On June 23, 2014, the applicant submitted an LRA update in order to comply with the LRA update requirements in 10 CFR 54.21(b). As part of this submittal, the applicant identified that the original OTSGs at Davis-Besse were replaced in the Cycle 18 (Spring 2014) refueling outage. Based on this plant modification, the applicant amended LRA Section 4.3.2.2.6 to propose changes to the TLAAs on metal fatigue of the OTSG primary and secondary side components, OTSG internal components, and AFW headers, nozzles, and nozzle thermal sleeves of the facility.

OTSGs Fatigue. LRA Section 4.3.2.2.6.1 states that the primary (tube) and secondary (shell) sides of the OTSGs were designed to the 1968 edition of ASME Code Section III, inclusive of 1968 summer addenda, and were analyzed for fatigue by the original equipment manufacturer. The CUFs for OTSG locations, which are less than 1.0, were based on the system design transients given in LRA Table 4.3-1. The applicant stated that the SG remote weld plugs have a limited design life of 33 heatup/cooldown cycles to maintain a fatigue usage of less than 1.0. The applicant's Fatigue Monitoring Program tracks the incurred cycles of these design

transients to ensure action is taken before reaching their design number of cycles for each transient. The applicant dispositioned the TLAA for the OTSGs in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of aging due to fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

By letter June 23, 2014, the applicant indicated that the OTSGs at Davis-Besse had been replaced and the new fatigue analyses had been performed for specific components in the replacement OTSGs, including those in the primary and secondary sides of the replacement OTSGs, associated with the replacement OTSG internals, and specific components in the AFW system (including AFW headers, nozzles, and nozzle thermal sleeves). The staff's evaluation of the new fatigue analyses for the replacement OTSG components is contained in Section 4.3.2.2.2 of this SSER.

OTSG Tube Sleeves Fatigue. LRA Section 4.3.2.2.6.2 describes the fatigue analysis for the tube sleeves that were used to repair leaking tubes of the OTSGs. In accordance with USAR Section 5.5.2.3, the applicant stated that the SG tubes may be plugged or repaired by mechanical (rolled) sleeving; however, Section III of the ASME Code does not provide design rules for mechanically roll-expanded attachments, and theoretical stress analyses are inadequate. The applicant stated that, in accordance with provisions of Appendix II, Section 1500, of ASME Code Section III, fatigue tests were performed to demonstrate the structural adequacy of the sleeves to withstand cyclic loadings based on the design transients.

The applicant indicated that the pressure cycling tests used 360 startup cycles to bound all B&W 177 fuel assembly plants. The applicant stated that, per USAR Table 5.1-8, its design basis is 240 startups, and it projected only 128 startups for 60 years of operation, as described in LRA Table 4.3-1. The applicant dispositioned the TLAA associated with fatigue testing of the OTSG tube sleeves, in accordance with 10 CFR 54.21(c)(1)(i) disposition, that the analysis will remain valid for the period of extended operation.

In the LRA update dated June 23, 2014, the applicant stated that the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage and that the new OTSGs no longer included any OTSG tube sleeves. The applicant stated that LRA Section 4.3.2.2.6.2 is being deleted as a result of the changes to the OTSG design. The staff's evaluation of the applicant's basis for deleting these metal fatigue analyses from the scope of the LRA is contained in Section 4.3.2.2.2 of this SSER.

OTSG Auxiliary Feedwater Modification. LRA Section 4.3.2.2.6.3 describes the fatigue analysis for the repair to the OTSGs AFW system. The modification was installed (in 1982) with an external header on each SG. The applicant stated that the AFW thermal sleeve stresses were also analyzed by B&W, and the analysis, performed in accordance with the requirements of the ASME Code for Class 1 components, provided a basis for demonstrating that the AFW thermal sleeve is capable of withstanding 40,000 cycles of AFW injection transients. The riser flange attachment to the SG shell was also analyzed per ASME Code requirements and was acceptable for a design life of 875 cycles of heatup/cooldown, bolt-up and unbolt, and AFW initiations. Transients 30A and 30B in LRA Table 4.3-1, which have 60-year projections of 387 and 442 cycles, respectively, are each less than the 875 design cycles for the riser flange attachment. The applicant stated that design transients are tracked for the number of occurrences under its Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. The applicant dispositioned the TLAAs of AFW repair, in accordance with 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the AFW modification will be managed for the period of extended operation by the Fatigue Monitoring Program.

In the LRA update dated June 23, 2014, the applicant stated that the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage and that the metal fatigue analyses for the repaired AFW nozzle design for the original OTSGs is not applicable to the CLB for the replacement OTSGs, which were installed during the Spring 2014 refueling outage. The applicant stated that LRA Section 4.3.2.2.6.3 is being deleted as a result of the OTSG replacement activities and the associated changes to the designs of the AFW systems. The staff's evaluation of the applicant's basis for deleting these metal fatigue analyses from the scope of the LRA is contained in Section 4.3.2.2.2 of this SSER.

OTSGs Tubes and Tube Stabilizers Flow-Induced Vibration. LRA Section 4.3.2.2.6.4 describes the fatigue analysis performed for FIV of the OTSG tubes and the tube stabilizers. The applicant stated that its latest analysis report showed the highest CUF for any existing tube configuration was 0.443 for an unrepaired tube next to the open lane, and the 60-year projected CUF value of 0.665 is acceptable. The applicant stated that the 60-year projected CUFs for the 3/8-inch tube-stabilizers, calculated using both high-cycle (FIV) and low-cycle (transients) fatigue, remains below the design limit of 1.0. The applicant dispositioned the fatigue TLAA associated with FIV of SG tubes and tube stabilizers, in accordance with 10 CFR 54.21(c)(1)(ii), that the TLAAs have been projected through the period of extended operation.

In the LRA update letter dated June 23, 2014, the applicant stated the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage and that the previous FIV analysis for the original OTSG tube and tube stabilizers did not apply to the replacement OTSGs that were installed during the Spring 2014 refueling outage. Therefore, by letter dated June 23, 2014, the applicant proposed to delete LRA Section 4.3.2.2.6.4 from the scope of the LRA. The staff's evaluation of the applicant's basis for deleting these FIV analyses from the scope of the LRA is contained in Section 4.3.2.2.2 of this SSER.

4.3.2.2.2 Staff Evaluation

The staff reviewed LRA Section 4.3.2.2, which consists of metal fatigue TLAAs for ASME Code Section III Class 1 vessels, pumps, and major components, to confirm, pursuant to the following, and dependent on the applicant's evaluation:

- 10 CFR 54.21(c)(1)(i), that the analyses remain valid during the period of extended operation
- 10 CFR 54.21(c)(1)(ii), that the analyses have been projected to the end of the period of extended operation
- 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation

In the LRA update letter dated June 23, 2014, the applicant stated that the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage.

In addition, by letter dated April 21, 2015, as amended in letters dated May 20, 2015, and June 5, 2015, the applicant amended the LRA to provide the RVIIP for Davis-Besse. As a result of these letters, the applicant amended the metal fatigue TLAAs for RVI components in LRA Section 4.3.2.2.6, and in particular, the applicant's basis for accepting the low-cycle fatigue TLAA for the RVI components, in accordance with 10 CFR 54.21(c)(1)(iii).

The following subsections reflect the staff's updated evaluations of the metal fatigue TLAAs for OTSG and RVI components. Otherwise, the staff's evaluations of the metal fatigue analyses for Class 1 vessels, pumps, and major equipment are as documented in SER Section 4.3.2.2.2.

Reactor Vessel Internals

Low-Cycle Fatigue. The staff reviewed LRA Section 4.3.2.2.2.1 on low-cycle fatigue of the RVI to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The staff reviewed the applicant's TLAA and the corresponding disposition, consistent with the review procedures in SRP-LR Section 4.3.3.1.1.3, which state that the reviewer should verify the appropriateness of the applicant's program for monitoring and tracking the number of critical thermal and pressure transients for the selected RCS components. The SRP-LR further states that the reviewer should verify that the applicant identified the appropriate program, as described and evaluated in the GALL Report. Furthermore, the reviewer should also ensure that the applicant's program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL Report.

The staff noted that, as discussed in LRA Section 4.3.2.2.2.1, the applicant has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts, and no fatigue analysis was performed for these bolts because it was not required during the original design. However, the staff noted that LRA Table 3.1.2-2, Row Nos. 42 and 110, for upper thermal shield bolts and flow distribution bolts, respectively, credit a TLAA to manage cumulative fatigue damage. It was not clear to the staff what TLAA was being referenced, since LRA Section 4.3.2.2.2.1 states that fatigue analyses were not performed for the RVIs. By letter dated May 2, 2011, the staff issued RAI 4.3-3 requesting that the applicant identify the fatigue TLAA that is being credited to manage cumulative fatigue damage of the components identified by the AMR items in LRA Table 3.1.2-2, Row Nos. 42 and 110.

In its response dated June 17, 2011, the applicant stated that it has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts; therefore, a correction is required to Row Nos. 42 and 110 of LRA Table 3.1.2-2. The staff noted that the applicant amended LRA Table 3.1.2-2 to remove the AMR items associated with stainless steel upper thermal shield bolts and flow distributor bolts exposed to borated reactor coolant that are being managed for cracking due to fatigue by a TLAA. Although these components do not have a fatigue TLAA associated with them, the staff noted that they will be managed by the applicant's PWR Reactor Vessel Internals Program for cracking during the period of extended operation. The staff finds the removal of these AMR items acceptable because a fatigue analysis was not performed for these components; therefore, they do not have a TLAA associated with them. The staff's concern described in RAI 4.3-3 is resolved.

The staff reviewed the CUF values provided by the applicant, in Table 3-1 of AREVA Document 51-9157140-001, in response to RAI 4.3-12 (letter dated June 17, 2011) and confirmed that the design CUF values for the replaced UCB, LCB, and LTS bolts are less than the design limit of 1.0. The staff noted that the applicant credited the cycle-counting activities of its Fatigue Monitoring Program as the basis for managing cumulative fatigue damage that may occur in the reactor vessel during the period of extended operation and will initiate corrective actions to ensure the design cycles and design limit of 1.0 will not be exceeded.

By letter dated June 05, 2015, the applicant amended LRA Section 4.3.2.2.1 to indicate that a combination of the Fatigue Monitoring Program (LRA AMP B.2.16) and the PWR Reactor Vessel Internals Program (LRA AMP B.2.32) will be used to demonstrate that the low-cycle fatigue analyses (i.e., low-cycle CUF analyses) for replaced UCB, LCB, and LTS bolts are acceptable, in accordance with 10 CFR 54.21(c)(1)(iii), and that the impacts of "cracking-fatigue" on the intended functions of these components will be adequately managed during the period of extended operation. The staff noted that the change in the basis for accepting the TLAA under 10 CFR 54.21(c)(1)(iii) was necessary because it was needed to resolve the applicant basis for responding to A/LAI #8, Subitem 5, on the methodology in EPRI MRP TR No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)" (ADAMS Accession Nos. ML12017A191 – ML12017A197 and ML12017A199), dated January 2012. This A/LAI was issued in the NRC's safety evaluation on the MRP-227-A report, dated December 16, 2011 (ADAMS Accession No. ML11308A770).

The staff noted that, under this amended basis, the PWR Reactor Vessel Internals Program would implement UT inspections of the UCB and LCB bolts because the components are designated as "primary" category components in the MRP-227-A report. Therefore, under this basis, the PWR Reactor Vessel Internals Program would be implementing direct volumetric inspections of the UCB and LCB bolts during the period of extended operation. The staff determined that this forms an acceptable basis for monitoring and managing any fatigue-induced cracking in the UCB and LCB bolts during the period of extended operation because it is consistent with the "monitoring and trending" element in GALL AMP XI.M16A, "PWR Vessel Internals." The staff also noted that, consistent with GALL AMP X.M1, the implementation of the Fatigue Monitoring Program provides an acceptable basis for managing fatigue-induced cracking in these components because it directly monitors for design transient occurrences at the plant and provides an acceptable basis for determining whether the CUF values of the components will remain within a CUF acceptance limit of 1.0. Therefore, the staff concludes that the combination of the Fatigue Monitoring Program and the PWR Reactor Vessel Internals Program provides an acceptable basis for accepting the CUF analyses for UCB and LCB bolts in accordance with 10 CFR 54.21(c)(1)(iii) because the applicant will perform both cycle counting and direct inspections of the components during the period of extended operation, which is consistent with the criteria in GALL Report AMP XI.M16A and in SRP-LR Section 4.3.2.1.1.3 and GALL Report AMP X.M1.

The basis is slightly different for the low-cycle CUF analyses that apply to the replaced LTS bolts, which are "expansion" category components in the MRP-227-A report. The staff noted that, under the MRP-227-A protocols for these "expansion" category bolts, the PWR Reactor Vessel Internals Program would only implement augmented UT inspections of the components if relevant indications of cracking were detected in the UCB or LCB bolts, which are the "primary" category components linked to potential inspections of LTS bolts. Thus, the staff concluded that the PWR Reactor Vessel Internals Program does not provide the main basis for accepting the CUF analyses for the LTS bolts, in accordance with 10 CFR 54.21(c)(1)(iii), because the program would not implement inspections of "expansion" components unless degradation was detected in the "primary" category components linked to the LTS bolts, as defined "expansion" category components for the AMP. Instead, the staff concluded that it would be acceptable to use the Fatigue Monitoring Program as the main basis for accepting the CUF analyses for these bolts, in accordance with 10 CFR 54.21(c)(1)(iii), because the applicant will use the Fatigue Monitoring Program to perform cycle-counting of the components during the period of extended operation, which is consistent with the criteria in SRP-LR Section 4.3.2.1.1.3 and in GALL Report AMP X.M1.

Therefore, consistent with the recommendation of GALL Report AMPs X.M1 and XI.M16A, the staff finds that the combined cycle-counting activities in the applicant's Fatigue Monitoring Program and inspection protocols for the PWR Reactor Vessel Internals Program are acceptable to manage fatigue-induced cracking (cumulative fatigue damage) in the replaced UCB, LCB, and LTS bolts during the period of extended operation and to accept the low-cycle fatigue analyses for these components in accordance with 10 CFR 54.21(c)(1)(iii). Additionally, it is consistent with SRP-LR Section 4.3.2.1.1.3 for the following reasons:

- The applicant's Fatigue Monitoring Program monitors and tracks the number of design basis transients that will occur through the period of extended operation.
- The applicant's Fatigue Monitoring Program includes action limits and corrective actions that will ensure that the CUF design limit of 1.0 will not be exceeded during the period of extended operation.
- The use of the applicant's Fatigue Monitoring Program is consistent with the recommendations of the GALL Report AMP X.M1 for managing cumulative fatigue damage.
- Additionally, implementing the applicant's PWR Reactor Vessel Internals Program will provide a direct means of inspecting for fatigue-induced cracking that may be occurring in the replaced UCB and LCB bolts during the period of extended operation, and for inspecting the LTS bolts if degradation exceeding the levels defined in Table 5-1 of MRP-227-A is detected in the UCB or LCB bolts.

The staff's evaluation of the applicant's Fatigue Monitoring Program is documented in SER Section 3.0.3.2.6. The staff's evaluation of the PWR Reactor Vessel Internals Program is documented in Section 3.0.3.3.6 of this SSER. A/LAI #8, Subitem 5, is resolved with respect to the basis for accepting the low-cycle fatigue analyses for the UCB, LCB, and LTS bolts, in accordance with 10 CFR 54.21(c)(1)(iii).

Reactor Vessel Internals and Incore Instrumentation Nozzles Flow-Induced Vibration. The staff reviewed LRA Subsection 4.3.2.2.2.2 on FIV of the RVI components and incore instrument nozzles to verify, pursuant to 10 CFR 54.21(c)(1)(i), that the FIV analysis for these components remains valid during the period of extended operation. The staff performed its review in accordance with the requirement in 10 CFR 54.21(c)(1)(i) and the acceptance criteria in SRP-LR Section 4.3.2.1.1.1.

The staff noted the applicant's submittal of the RVIIP (as documented in the letters of April 21, 2015, and supplemented by information in the letters of May 20, 2015, and June 5, 2015) did not result in any need to change the applicant's evaluation of the FIV analysis for the RVI components or incore instrumentation nozzles or the applicant's basis for accepting this TLAA, in accordance with 10 CFR 54.21(c)(1)(i). Therefore, the staff's prior evaluation of the FIV analysis for the RVI components or incore instrumentation nozzles remains valid for this SSER, as documented in SER Section 4.3.2.2.2. The evaluation in SER Section 4.3.2.2.2 provides the staff's basis for concluding that the FIV analysis for the RVI components or incore instrumentation nozzles is acceptable, in accordance with 10 CFR 54.21(c)(1)(i), and meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.1.

Based on this review, the staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the FIV analysis of the RVIs and incore instrument nozzles remains valid during the period of extended operation.

Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.1 because the endurance limit assumed in the original analysis would not be exceeded and the implicit CUF value of zero remains valid during the period of extended operation.

Surveillance Capsule Holder Tubes Flow-Induced Vibration. The staff reviewed LRA Subsection 4.3.2.2.2.3 on the high-cycle fatigue analysis (i.e., FIV-based CUF analysis) of the surveillance capsule holder tubes to verify, pursuant to 10 CFR 54.21(c)(1)(ii), that the analysis has been projected to the end of the period of extended operation. The staff performed its review in accordance with the requirement in 10 CFR 54.21(c)(1)(ii) and the acceptance criteria in SRP-LR Section 4.3.2.1.1.2.

The staff noted that the applicant's submittal of the RVIIP (as documented in the letters of April 21, 2015, and supplemented by information in the letters of May 20, 2015, and June 5, 2015) did not result in any need to change the applicant's evaluation of the high-cycle fatigue analysis for the reactor vessel surveillance capsule holder tubes or the applicant's basis for accepting this TLAA in accordance with 10 CFR 54.21(c)(1)(ii). Therefore, the staff's prior evaluation of the high-cycle fatigue analysis for the reactor vessel surveillance capsule holder tubes remains valid for this SSER, as documented in SER Section 4.3.2.2.2.

The evaluation in SER Section 4.3.2.2.2 provides the staff's basis for concluding that the high-cycle fatigue analysis for the reactor vessel surveillance capsule holder tubes is acceptable, in accordance with 10 CFR 54.21(c)(1)(ii), and meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.2.

Based on this review, staff finds the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the surveillance capsule holder tubes FIV analysis has been projected to the end of the period of extended operation. Additionally, it meets the acceptance criteria in SRP-LR Section 4.3.2.1.1.2 because the applicant demonstrated that the projected CUF values will be less than the ASME Code Section III, design limit of 1.0 through the period of extended operation with significant margin.

Once-Through Steam Generators Fatigue. The evaluation in this section of the SSER supersedes the staff's evaluation of the metal fatigue analyses for OTSG components in Section 4.3.2.2.2, Subsection, "*Once-Through Steam Generator Fatigue,"* of the staff's SER.

In the LRA update dated June 23, 2014, the applicant updated LRA Section 4.3.2.2.6, "Once-Through Steam Generators," and LRA Section 4.3.2.2.6.1, "OTSGs Fatigue," to indicate that the limiting components in the replacement OTSGs for Davis-Besse were assessed with updated fatigue analyses. The staff also noted that, in this LRA update, the applicant amended LRA AMR Table 3.1.2-4 to include updated AMR items for the following components in the replacement OTSG designs that were reanalyzed with updated metal fatigue analyses (i.e., that received updated CUF analyses):

- replacement OTSG primary boundary bolts
- replacement OTSG primary manway and inspection opening covers
- replacement OTSG primary side tubes
- replacement OTSG primary side tube plugs
- replacement OTSG primary side upper and lower heads
- replacement OTSG primary side inlet and outlet nozzles

- replacement OTSG primary side upper and lower tubesheets
- replacement OTSG primary side tube-to-tubesheet welds
- replacement OTSG secondary side AFW headers, risers, nozzles, and nozzle thermal sleeves
- replacement OTSG secondary side shrouds and shroud support rings and lugs
- replacement OTSG secondary side manways and handhole covers
- replacement OTSG secondary side main feedwater (MFW) header support plates and gussets
- replacement OTSG secondary side MFW headers and risers
- replacement OTSG secondary side MFW nozzles and MFW nozzle thermal sleeves
- replacement OTSG secondary side steam outlet nozzles, vent nozzles, drain nozzles, and level sensing nozzles
- replacement OTSG secondary side shells
- replacement OTSG secondary side tube support plates
- replacement OTSG secondary side tube support plate spacers
- replacement OTSG secondary side tube support rods (i.e., tie rods)
- replacement OTSG base support stools and base support platforms

The staff noted that, in these AMR items, the applicant stated that “cracking – fatigue” of the replacement OTSG components would be adequately managed for the period of extended operation using a metal fatigue TLAA.

The staff noted that, in its LRA amendment, the applicant did not amend LRA Section 4.3.2.2.6 and LRA Section 4.3.2.2.6.1 to specify the replacement OTSG, MFW, or AFW components that were analyzed in accordance with an updated metal fatigue analysis (i.e., in accordance with an updated ASME Section III CUF analysis) or to identify how the updated metal fatigue analyses for these components were being accepted, in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii).

By letter dated August 19, 2014, the staff issued RAI 4.3.2.2.6.1-1 (followup), requesting in Part 1 of the RAI that the applicant identify all replacement OTSG components that had been analyzed in accordance with an ASME Code Section III metal fatigue analysis (i.e., CUF analysis). In RAI 4.3.2.2.6.1-1, Part 2, for each replacement OTSG component that had been analyzed in accordance with an updated CUF analysis, the staff asked the applicant to perform a comparison of the CUF analysis for the component to the six criteria for defining a TLAA in 10 CFR 54.3(a) and to justify why the updated CUF analysis for the component would not need to be identified as a TLAA in accordance with the requirement in 10 CFR 54.21(c)(1).

In RAI 4.3.2.2.6.1-1, Part 3, for each replacement OTSG component that was analyzed in accordance with a metal fatigue analysis that conforms to the definition of a TLAA, as defined in 10 CFR 54.3(a), the staff asked the applicant to justify acceptance of the metal fatigue analysis in accordance with the requirements in 10 CFR 54.21(c)(1)(i), (ii), or (iii).

The applicant responded to RAI 4.3.2.2.6.1-1, Parts 1, 2, and 3 in a letter dated September 16, 2014. In its response to RAI 4.3.2.2.6.1-1, Part 1, the applicant stated that the

replacement OTSG components that have been analyzed with updated metal fatigue analyses have been identified in the applicant's update of LRA Table 3.1.2-4, as provided in the FENOC letter dated June 23, 2014, and that the fatigue analyses for these replacement OTSG components were performed in accordance with the 2001 Edition of the ASME Code Section III, inclusive of the 2003 Addenda. The applicant clarified that LRA Section 4.3.2.2.6.3, as previously revised in the letter dated June 23, 2014, is not needed as a separate section and remains as "Not used." The applicant stated that, unlike the original OTSGs, where, by modification, the AFW headers were relocated to the outside of the SG and the modification was evaluated separately, the AFW headers in the replacement OTSGs are also located on the outside of the OTSGs. The applicant stated that the updated fatigue analysis (CUF analysis) for the AFW headers is included with the scope of the evaluation that is given in the amended version of LRA Section 4.3.2.2.6.1, as discussed in the letter dated June 23, 2014.

In its response to RAI 4.3.2.2.6.1-1, Part 2, the applicant stated that the new CUF analyses for the subject replacement OTSG components, including the new CUF analyses for the AFW headers, nozzles and nozzle thermal sleeves, meet the six criteria for defining TLAAs in accordance with 10 CFR 54.3(a). In its response to RAI 4.3.2.2.6.1-1, Part 3, the applicant stated that the new CUF analyses for the replacement OTSG components were calculated using the applicable design transients for the components and that the CUF values for the components are all less than the acceptance criterion of 1.0 for ASME Code-calculated CUF values. The applicant also stated that the design transients used in the fatigue analyses for the subject replacement OTSG components are included in LRA Table 4.3-1, "60-year Projected Cycles." The applicant stated that, as provided in LRA Section 4.3.2.2.6.1, the number of occurrences of design transients is tracked and will continue to be tracked by the Fatigue Monitoring Program to ensure that: (a) action is taken before the cycle limits for the design transients are reached, and (b) the effects of "cracking – fatigue" on the intended functions of the replacement OTSGs (and their components) will be adequately managed by the Fatigue Monitoring Program during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

The staff noted that, in the LRA amendment, the applicant amended LRA Table 3.1.2-4 to identify the specific replacement OTSG components that had been within the scope of a metal fatigue TLAA (i.e., CUF analysis) and these components are within the scope of the metal fatigue TLAA assessment that is given in LRA Section 4.3.2.2.6.1. These are the specific replacement OTSG components that have been identified earlier in this SER section (refer to the components in the previous bulleted list of this SSER section).

The staff noted the applicant indicated that it will use the Fatigue Monitoring Program as the basis for managing the impact of "cracking – fatigue" on the intended functions of these replacement OTSGs during the period of extended operation and that the basis was consistent with the acceptance criterion in SRP-LR Section 4.3.2.1.1.3, which states that an applicant's Fatigue Monitoring Program may be used to accept a metal fatigue analysis in accordance with 10 CFR 54.21(c)(1)(iii) and to manage the impacts of "cracking – fatigue" on the intended functions of the components during the period of extended operation. Therefore, the staff determined that the applicant provided an acceptable basis for accepting the updated fatigue analyses for the replacement OTSG components in accordance with 10 CFR 54.21(c)(1)(iii) because the applicant's basis was consistent with the acceptance criteria guidance in SRP-LR Section 4.3.2.1.1.3.

Therefore, based on this review, the staff finds that the applicant has appropriately amended the LRA and provided an acceptable metal fatigue TLAA for the replacement OTSG components because of the following:

(a) The applicant has appropriately amended the LRA to include the specific replacement OTSG components that are within the scope of a CUF analysis.

(b) The applicant has shown that these CUF analyses are applicable TLAAs for the LRA.

(c) The applicant will use its Fatigue Monitoring Program to manage the impacts of "cracking – fatigue" on the intended functions of the replacement OTSGs and the replacement AFW header during the period of extended operation.

(d) This provides an acceptable basis for accepting these TLAAs, in accordance with the criterion in 10 CFR 54.21(c)(1)(iii), and for demonstrating that the effects of "cracking – fatigue" on the intended functions of the replacement OTSGs, and their components, will be adequately managed during the period of extended operation.

RAI 4.3.2.2.6.1-1, Parts 1, 2, and 3, are resolved.

Once-Through Steam Generator Tube Sleeves Fatigue. The evaluation in this section of the SSER supersedes the staff's evaluation of the metal fatigue analysis for OTSG tube sleeves in SER Section 4.3.2.2.2. The staff noted that, in the LRA, the applicant stated that the design of the original OTSGs included some sleeved OTSG tubes and that the design basis for the OTSG tube sleeves included a metal fatigue analysis that conformed to the definition of a TLAA in 10 CFR 54.3(a). The staff noted that the applicant dispositioned the metal fatigue analysis for the OTSG tube sleeves in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(i), which requires the applicant to demonstrate that the analysis will remain valid for the period of extended operation.

In the LRA update letter dated June 23, 2014, the staff noted that the applicant indicated the original OTSGs in the plant design were replaced during the Spring 2014 refueling outage and that the new OTSG did not include any tube sleeves. Therefore, the staff noted that the applicant stated that the previous metal fatigue analysis (as discussed in LRA Section 4.3.2.2.6.2) for tube sleeves in the original OTSGs did not apply to the design of replacement OTSGs.

The staff verified that, in the letter dated June 23, 2014, the applicant amended LRA Table 2.3.1-4, "Steam Generators Components Subject to Aging Management Review" to identify those components in the replacement OTSG designs that were required to be within the scope of the LRA. The staff noted that the revised version of LRA Table 2.3.1-4 did not identify any replacement OTSG tube sleeves that would need to be within the scope of the LRA, based on the updated design of the OTSGs. Based on the contents of the amended table for the replacement OTSG components, the staff noted that the previous metal fatigue analysis for OTSG tube sleeves in the LRA Section 4.3.2.2.6.2 does not apply to the updated CLB for the replacement OTSGs because the replacement OTSGs do not include any tube sleeves.

As a result, the staff determined that the previous metal fatigue analysis for the tube sleeves in the original OTSG design does not conform to Criterion 6 in 10 CFR 54.3(a) because the analysis is no longer contained or incorporated by reference in the updated CLB for the OTSGs. Therefore, based on this review, the staff finds that the applicant has provided an acceptable

basis for deleting the metal fatigue TLAA for the OTSG tube sleeves from the scope of LRA Table 4.1-1 and LRA Section 4.3.2.2.6.2 because the staff has verified the following:

(a) The analysis does not conform to Criterion 6 in 10 CFR 54.3(a).

(b) The analysis does not meet the definition of a TLAA, as defined in 10 CFR 54.3(a), for the updated CLB that applies to the replacement OTSGs.

Once-Through Steam Generators Auxiliary Feedwater Modification. The evaluation in this section of the SSER supersedes the staff's evaluation of the metal fatigue analyses for components in the modified AFW system design, as given in SER Section 4.3.2.2.2.

The staff noted that, in the LRA, the applicant stated that the original AFW headers internal to the SGs were found damaged during the 1982 refueling outage and that the ensuing repair modification installed an external AFW header on each of the original OTSGs. The applicant also indicated that the modification of the plant design included some rerouting of AFW piping and supports and a fatigue analysis for the repaired AFW nozzle design on the OTSGs. The staff noted that, in the original version of the LRA, the applicant had identified this metal fatigue analysis as a TLAA that was dispositioned in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii), which requires an applicant to demonstrate that the effects of aging on the intended functions of the components will be adequately managed during the period of extended operation.

The staff noted that, in the LRA update letter dated June 23, 2014, the applicant stated that the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage and that the metal fatigue analyses for the repaired AFW nozzle design on the original OTSGs is not applicable to the updated CLB for the replacement OTSGs. Instead, the staff verified that the applicant included its new CUF analyses for the external AFW headers, nozzles, and nozzle thermal sleeves in the replacement OTSGs within the scope of the metal fatigue TLAAs that are included in the updated version of LRA Section 4.3.2.2.6.1, as identified and discussed in the LRA update dated June 23, 2014.

As a result, the staff determined that the metal fatigue analysis for the previous, modified AFW design does not conform to Criterion 6 in 10 CFR 54.3(a) because the analysis is not contained or incorporated by reference in the updated CLB for the replacement OTSGs. Therefore, based on this review, the staff finds that the applicant has provided an acceptable basis for deleting the metal fatigue TLAA for the AFW modification from the scope of LRA Table 4.1-1 and LRA Section 4.3.2.2.6.3 because the staff has verified the following:

(a) The analysis does not conform to Criterion 6 in 10 CFR 54.3(a).

(b) The analysis does not meet the definition of a TLAA, as defined in 10 CFR 54.3(a), for the CLB that applies to the replacement OTSGs.

Instead, the staff has confirmed that the applicant included the updated metal fatigue analyses for the new external AFW headers and the new AFW nozzles and nozzle thermal sleeves in the scope of the June 23, 2014, LRA update (i.e., revision) of LRA Section 4.3.2.2.6.1. The staff evaluated the new fatigue analyses for these components in Section 4.3.2.2.2, Subsection "Once-Through Steam Generator Fatigue" of this SSER and has determined that the new analyses are acceptable because the applicant will use its Fatigue Monitoring Program to accept the metal fatigue analyses for these components, in accordance with the criterion in

10 CFR 54.21(c)(1)(iii), and to manage the impacts of "cracking – fatigue" on the intended functions of these components during the period of extended operation.

Once-Through Steam Generators Tubes and Tube Stabilizers Flow-Induced Vibration. The evaluation in this section of the SSER supersedes the staff's previous evaluation of the FIV analyses for tubes and tube stabilizers in the original OTSGs, as given in SER Section 4.3.2.2.2.

The staff noted that, in Section 4.3.2.2.6.4 of the LRA, the applicant stated that the design of the original OTSGs included tubes and tube stabilizers and the design basis for the tubes and tube stabilizers included a cyclical flow-vibration analysis which conformed to the definition of a TLAA in 10 CFR 54.3(a). The staff noted that, in the original version of the LRA, the applicant dispositioned the FIV analysis for the OTSGs tube and tube stabilizers, in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(ii), which requires the applicant to demonstrate that the analysis has been projected through to the end of the period of extended operation.

The staff noted that, in the LRA update letter dated June 23, 2014, the applicant stated that the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage and the previous FIV analysis for original OTSG tube and tube stabilizers did not apply to the replacement OTSGs. Therefore, in the letter dated June 23, 2014, the applicant proposed to delete LRA Section 4.3.2.2.6.4 from the scope of the LRA.

It was not evident to the staff why the tubes and or other components in the replacement OTSGs would not have been required to be analyzed with an FIV analysis, similar to how the tubes and tube stabilizers in the original OTSGs were analyzed for FIVs, or why such an FIV analysis would not need to be identified as a TLAA for the replacement OTSGs or specific subcomponents in the replacement OTSGs. Therefore, by letter dated August 19, 2014, the staff issued RAI 4.3.2.2.6.4-1 (LRA Update followup), requesting that the applicant provide additional clarification on whether the design code or codes for the replacement OTSGs, or specific components in the replacement OTSGs, required an FIV analysis. If it is determined that the design code for the replacement OTSGs did require an FIV analysis, the staff asked the applicant to justify why the applicable FIV analysis would not need to be identified as a TLAA, in accordance with the definition criteria for TLAAs in 10 CFR 54.3(a).

The applicant responded to RAI 4.3.2.2.6.4-1 (LRA Update followup) by letter dated September 16, 2014. In its response, the applicant stated that the design specification for the replacement OTSGs did require the applicant to perform a flow-induced and turbulence-induced vibration analysis for components in the replacement OTSG tube bundles. The applicant stated that the analyses were performed to show that fatigue failures, excess tube fretting and tube wear, or wear of other SG internals, will not occur during future plant operation (including operation of the replacement OTSGs). The applicant stated that the replacement OTSGs were installed in the Spring 2014 refueling outage and were qualified for 40 years of service from the time of installation. The applicant stated that replacement OTSGs will experience only about 23 years of operation by the end of the period of extended operation. The applicant stated that the inservice time of the replacement OTSGs will not exceed the time associated with a 40-year qualified life. Therefore, the applicant stated that the flow-induced and turbulence-induced vibration analysis for the replacement OTSG components does not meet the six criteria in 10 CFR 54.3(a) and does not a constitute a TLAA that would require evaluation, in accordance with the requirement in 10 CFR 54.21(c)(1).

Based on the applicant's response, the staff noted that a time-dependent flow-induced and turbulence-induced vibration analysis is part of the updated CLB for the internal components of the replacement OTSGs at Davis-Besse. The staff also noted that the time-dependent flow-induced and turbulence-induced vibration analysis for the internal replacement OTSG components would qualify the operation of the internal OTSG components beyond the time associated with the period of extended operation. Therefore, based on this review, the staff determined that the time-dependent flow-induced and turbulence-induced vibration analysis for the internal replacement OTSG components does not conform to Criterion 3 in 10 CFR 54.3(a) because the analysis does not involve time-dependent assumptions defined by the current operating term. Therefore, based on this review, the staff finds that the applicant has provided an acceptable basis for concluding that the LRA does not need to include an FIV TLAA for the internal components in the replacement OTSGs because the staff has verified the following:

(a) The analysis does not conform to Criterion 3 in 10 CFR 54.3(a).

(b) The analysis does not meet the definition of a TLAA, as defined in 10 CFR 54.3(a), for the CLB that applies to replacement OTSGs.

RAI 4.3.2.2.6.4-1 (LRA Update followup) is resolved.

4.3.2.2.3 USAR Supplement

Applicable subsections in LRA Section A.2.3 provide the USAR supplements that summarize the metal fatigue TLAAs for Class 1 vessels, pumps, and major equipment. The staff reviewed the applicable USAR supplements in LRA Sections A.2.3.1 (including Subsections A.2.3.1.1, A.2.3.1.3, and A.2.3.1.5) and A.2.3.2 (including Subsections A.2.3.2.1 – A.2.3.2.8), consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer verifies that the applicant provided information to be included in the USAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. This evaluation supplements the staff's previous evaluation, in SER Section 4.3.2.2.3, of the USAR supplement sections that apply to Class 1 vessels, pumps, and major equipment.

By letter dated June 23, 2014, the applicant amended USAR Supplement A.2.3.10.2, "Once Through Steam Generators," as follows in order to account for OTSG and AFW system design changes that resulted from the OTSG replacement activities in Spring 2014:

A.2.3.2.10 Once Through Steam Generator

The primary (tube) and secondary (shell) sides of the once through steam generators are designed to ASME Section III, ~~1968 Edition through Summer 1968 Addenda~~ 2001 Edition with 2003 Addenda. The steam generators were analyzed for fatigue by the original equipment manufacturer. The cumulative usage factors for the limiting primary and secondary side steam generator locations were calculated based on design transients, and are all less than 1.0. ~~*In addition, the steam generator remote weld plugs have a limited design life of 33 heatup/cooldown cycles to maintain a fatigue usage of less than 1.0.*~~ The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

The staff verified that the changes to USAR Supplement Section A.2.3.2.10 were appropriate for changes that were made to the plant design as a result of the OTSG replacement activities that were implemented in the Spring 2014 refueling outage and for the new fatigue analyses that were performed for specific primary side and secondary side components in the replacement OTSGs, including specific replacement OTSG internal components and portions of piping systems entering the replacement OTSGs, as identified in Section 4.3.2.2.2 of this SSER.

By letter dated June 23, 2014, the applicant amended LRA Appendix A to delete LRA Section A.2.3.1.5, "Steam Generator Remote Welded Plugs," from the scope of the USAR Supplement for the LRA. The applicant stated that the USAR supplement summary description for the OTSG remote welded plugs in LRA Section A.2.3.1.5 is being deleted from the scope of the USAR supplement as a result of the OTSG replacement design modification activities that were implemented in the Spring 2014 refueling outage.

The staff noted that, in the applicant's LRA update letter dated June 23, 2014, the applicant included its metal fatigue TLAAs (i.e., CUF analyses) for primary side tube plugs in the replacement OTSGs as a revision to the scope of LRA Section 4.3.2.2.6.1. Therefore, the staff reviewed the scope of USAR Supplement Section A.2.3.10.2, as updated in the letter dated June 23, 2014, to determine whether the USAR supplement for the OTSGs had provided a sufficient USAR supplement summary description for the primary side tube plugs that were included in the design of the applicant's replacement OTSGs.

The staff determined that the scope of USAR Supplement Section A.2.3.10.2 includes primary and secondary side OTSG component locations with a CUF analysis, which now includes the primary side plugs in the replacement OTSGs. Thus, the staff concludes that the applicant had provided an acceptable basis for deleting USAR Supplement Section A.2.3.1.5 from the scope of the LRA because the staff has verified that the USAR supplement summary description for the fatigue analyses of the primary side tube plugs in the replacement OTSGs is adequately summarized and addressed in the version of USAR Supplement Section A.2.3.10.2 that was included in the letter dated June 23, 2014.

By letter dated June 23, 2014, the applicant revised LRA Appendix A to delete USAR Supplement Section A.2.3.2.6, "Steam Generator Tube Sleeve Fatigue," from the scope of the USAR supplement for the LRA. The applicant stated that the USAR supplement summary description for the tube sleeves in LRA Section A.2.3.2.6 is being deleted from the scope of the USAR supplement as a result of the OTSG replacement design modification activities that were implemented in the Spring 2014. The applicant stated that the replacement OTSGs do not currently have any sleeves in the OTSG design. Therefore, the staff finds that the applicant has provided an acceptable basis for deleting LRA Section A.2.3.2.6, "Steam Generator Tube Sleeve Fatigue," from the scope of the USAR supplement for the LRA because tube sleeves are not currently included in the design of the replacement OTSGs.

By letter dated June 23, 2014, the applicant amended LRA Appendix A to delete LRA Section A.2.3.2.7, "Auxiliary Feedwater Header Modification," from the scope of the USAR supplement for the LRA. The applicant stated that the USAR supplement summary description in LRA Section A.2.3.2.7 is being deleted from the scope of the USAR supplement as a result of the OTSG replacement design modification activities that were implemented in the Spring 2014 refueling outage.

The staff noted that, in the applicant's LRA update letter dated June 23, 2014, the applicant deleted the metal fatigue TLAAs in LRA Section 4.3.2.2.6.3 from the scope of the LRA. Instead,

the staff verified that the applicant included the new metal fatigue analyses for the AFW nozzles and nozzle thermal sleeves that were included in the replacement OTSGs and identified these analyses as TLAAs in the amended version of LRA Section 4.3.2.2.6.1 that was included in the letter dated June 23, 2014.

The staff reviewed the scope of USAR Supplement Sections A.2.3.10.2, as updated in the letter dated June 23, 2014, in order to determine whether the USAR supplement for the OTSGs had provided a sufficient USAR supplement summary description for the fatigue analyses of the AFW nozzles and nozzle thermal sleeves that were installed during the Spring 2014 design modification. The staff determined that the scope of USAR Supplement Section A.2.3.10.2 includes primary and secondary side OTSG component locations for which a CUF analysis was performed, which now includes the AFW headers, nozzles, and nozzle thermal sleeves used in the design of the replacement OTSGs. Therefore, the staff finds that the applicant had provided an acceptable basis for deleting USAR Supplement Section A.2.3.2.7 because the staff has verified that the USAR supplement summary description of the fatigue analyses for the AFW headers, nozzles, and nozzle thermal sleeves is adequately summarized and addressed in the version of USAR Supplement Section A.2.3.10.2 that was included in the letter dated June 23, 2014.

By letter dated June 23, 2014, the applicant amended LRA Appendix A to delete LRA Section A.2.3.2.8, "Steam Generator Tubes and Tube Stabilizers Flow-Induced Vibration," from the scope of the USAR supplement for the LRA. The applicant stated that the USAR supplement summary description in LRA Section A.2.3.2.8 is being deleted from the scope of the USAR supplement as a result of the OTSG replacement design modification activities that were implemented in the Spring 2014 refueling outage. In its response to RAI 4.3.2.2.6.4-1 dated September 16, 2014, the applicant clarified that the updated FIV analyses for tubes and tube stabilizers in the replacement OTSGs do not qualify as TLAAs because they were not analyzed in terms of a time-limited assumption defined by the current operating period.

As previously discussed in Section 4.3.2.2.2 of the this SSER, the staff verified that the FIV analyses for the tubes and tube stabilizers in the replacement OTSGs were not analyzed in terms of a time-limited assumption defined by the current operating period. Specifically, the staff noted that the assessment of high-cycle vibrations in the FIV analyses of the replacement OTSG tube and tube stabilizer components goes well beyond the end of the proposed period of operation for the Davis-Besse facility. Therefore, the staff determined that the applicant had provided an acceptable basis for concluding that the updated flow-vibration analyses for these components do not constitute TLAAs because they do not conform to Criterion 3 in 10 CFR 54.3(a). Therefore, based on this analysis, the staff concluded that the applicant has provided an acceptable basis for deleting USAR Supplement Section A.2.3.2.8 from the scope of LRA Appendix A because the updated FIV analyses for these components do not qualify as TLAAs that need to be identified, in accordance with the requirement in 10 CFR 54.21(c)(1).

By letter dated June 5, 2015, the applicant amended the USAR Supplement Section A.2.3.2.1, "Reactor Vessel Internals Bolts," as follows:

> **A.2.3.2.1 Reactor Vessel Internals Bolts**
>
> Although the reactor vessel internals are designed to meet the stress requirements of ASME Section III, they are not code components. Consequently, a fatigue analysis of the reactor vessel internals was not required and not performed as part of the original design.

> FENOC has replace[d] the majority of the stainless steel, Alloy 286, bolts for the reactor vessel internals with Alloy X-750 HTH bolts at Davis Besse. The replacement bolts were designed to ASME Section III, and are provided with fatigue analyses. FENOC has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts at Davis Besse. Design cumulative usage factors for the reactor vessel internals bolts are based on design cycles.
>
> The effects of fatigue on the reactor vessel internals bolts will be managed by the Fatigue Monitoring Program and the PWR Reactor Vessel Internals Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

The staff has evaluated and accepted the applicant basis for using these AMPs to accept the fatigue analyses for replaced RVI bolts in the "Reactor Vessel Internals, Low-Cycle Fatigue," subsection of Section 4.3.2.2.2 of this SSER. Therefore, the staff finds the updated version of USAR Supplement Section A.2.3.2.1 to be acceptable because it is consistent with the applicant's updated basis to use both the Fatigue Monitoring Program and the PWR Reactor Vessel Internals Program as the bases for accepting these metal fatigue TLAAs, in accordance with 10 CFR 54.21(c)(1)(iii).

Based on its review of the USAR supplement, as amended by letters dated June 17, 2011, June 23, 2014, and June 5, 2015, the staff finds it meets the acceptance criteria in SRP-LR Section 4.3.2.3. Additionally, the staff determines that the applicant provided an adequate summary description of its actions to address the fatigue TLAAs of Class 1 vessels, pumps, and major components (including fatigue analyses for limiting locations in the primary sides and secondary sides of the replacement OTSGs and the fatigue analyses for replaced RVI bolts), as required by 10 CFR 54.21(d).

4.3.2.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that the FIV analyses for the RVIs and incore instrumentation nozzles remain valid during the period of extended operation. The staff also concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(ii), that the high-cycle fatigue analysis (i.e., vibrational-based CUF analysis) of the reactor vessel surveillance capsule holder tubes has been projected to the end of the period of extended operation. The staff also concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue of the RV, RVIs, control rod drive housings, RCP casings, pressurizer components, OTSG primary and secondary shell components, OTSG plugs, and OTSG AFW components will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains appropriate summary descriptions of the TLAAs, as required by 10 CFR 54.21(d).

4.3.2.3 Class 1 Piping and Valves

4.3.2.3.1 Summary of Technical Information in the Application

In the LRA update letter dated June 23, 2014, the applicant stated that the original OTSGs in the plant design were replaced in the Spring 2014 refueling outage. Several of the sections below have been changed by adding information to reflect this plant modification.

Class 1 Piping Fatigue

By letter dated June 23, 2014, the applicant amended LRA Section 4.3.2.3 and stated that it had replaced some of the hot leg piping in the main reactor coolant loops as part of the replacement OTSG activities that were implemented in the Spring 2014 refueling outage. The applicant stated that the new design code of record for the Class 1 piping sections is the 2001 Edition of ASME Code Section III, inclusive of the 2003 Addenda.

4.3.2.3.2 Staff Evaluation

Class 1 Piping Fatigue

The staff noted that, by letter dated June 23, 2014, the applicant amended LRA Section 4.3.2.3 and stated that it had replaced some of the hot leg piping in the main reactor coolant loops as part of the OTSG replacement activities that were implemented in the Spring 2014 refueling outage. The staff also noted that the applicant stated that the code of record for the new Class 1 piping sections is the 2001 Edition of ASME Code Section III, inclusive of the 2003 Addenda. The staff noted that this design code of record requires the applicant to perform a fatigue analysis (CUF analysis) for the new (replaced) piping segments in the RCS hot leg piping. The applicant conservatively identified the new CUF analysis as a TLAA for the LRA. Therefore, the staff determined the previous basis in 10 CFR 54.21(c)(1)(iii) for accepting the metal fatigue TLAA for Safety Class 1 piping is still valid and applicable to the evaluation of the CUF analyses that the applicant performed and applied to new hot leg piping segments because the applicant will use its Fatigue Monitoring Program to manage the impact of "cracking – fatigue" on the intended functions of the Safety Class 1 piping, including those for the hot leg portions of the piping.

4.3.2.3.3 USAR Supplement

Applicable subsections in LRA Section A.2.3 provide the USAR supplements that summarize the metal fatigue TLAAs for Class 1 piping and valves. The staff reviewed the applicable USAR supplements for Class 1 piping in LRA Sections A.2.3.1 (including Subsections A.2.3.1.1 – A.2.3.1.5) and A.2.3.2.11, consistent with the review procedures in SRP-LR Section 4.3.3.3, which state that the reviewer verifies that the applicant provided information to be included in the USAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. This evaluation supplements the staff's previous evaluation in SER Section 4.3.2.3.3 of the USAR supplement sections that apply to the metal fatigue analyses for Class 1 piping and valves.

By letter dated June 23, 2014, the applicant amended LRA Section A.2.3.2.11 to include the following USAR supplement summary statement for the Safety Class 1 piping at the plant:

A portion of the reactor coolant system hot leg piping was replaced in support of steam generator replacement in the spring of 2014. Applicable ASME Code of Construction for the replaced hot leg piping is Section III, 2001 Edition with 2003 Addenda.

The staff found the amendment to the USAR supplement summary description in LRA Section A.2.3.2.11 to be appropriate for the replacement piping in the hot leg portions of the main coolant loops because the amendment describes the modifications that were made to the Safety Class 1 piping and identifies the appropriate design code of record for the hot leg piping that was replaced during the Spring 2014 OTSG refueling outage.

4.3.2.3.4 Conclusion

There are no changes or updates to this section of the SER. Therefore, the previous conclusion for its evaluation of the metal fatigue TLAA for Class 1 piping remains valid as documented in SER Section 4.3.2.3.4. Based on this review, the staff concludes that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging related to fatigue of the Class 1 piping will be adequately managed for the period of extended operation. The staff also concludes that the USAR supplement contains appropriate summary descriptions of the TLAA, as required by 10 CFR 54.21(d).

4.7 Other Plant-Specific Time-Limited Aging Analyses

4.7.5 Inservice Inspection—Fracture Mechanics Analyses

4.7.5.2 Once-Through Steam Generator 1-2 Flaw Evaluations

4.7.5.2.1 Summary of Technical Information in the Application

LRA Section 4.7.5.2 describes the applicant's TLAA for the OTSG 1-2 flaw evaluations. In the LRA, the applicant concluded that the effects of fatigue on the OTSG 1-2 flaws will be appropriately managed during the period of extended operation by the Fatigue Monitoring Program (LRA Section B.2.16), in accordance with 10 CFR 54.21(c)(1)(iii).

By letter dated June 23, 2014, the applicant provided an LRA update and deleted Section 4.7.5.2, "Once-Through Steam Generator 1-2 Flaw Evaluations," from the LRA. The applicant stated that, based on the installation of the replacement SGs, LRA Section 4.7.5.2 as previously revised by FENOC in the letter dated June 3, 2011 (ADAMS Accession No. ML11159A132), is no longer applicable to the CLB for Davis-Besse. This section 4.7.5.2 replaces the previous section 4.7.5.2 of the SER.

4.7.5.2.2 Staff Evaluation

The staff initially reviewed LRA Section 4.7.5.2 on the OTSG 1-2 flaw evaluation to verify, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of fatigue on the OTSG 1-2 flaws will be adequately managed for the period of extended operation. However, the staff noted that, on June 23, 2014, the applicant provided an LRA update and deleted Section 4.7.5.2 from the LRA. The applicant stated that, based on the installation of the replacement SGs, LRA Section 4.7.5.2, as previously revised by FENOC in the letter dated June 3, 2011 (ADAMS Accession No. ML11159A132), is no longer applicable to the CLB for Davis-Besse.

The staff noted that during the Spring 2014 refueling outage, the applicant replaced the OTSGs at Davis-Besse. The staff noted that the replacement OTSGs do not contain the flaws that were detected in the previous OTSGs. Therefore, the staff determined that the previous flaw evaluation TLAA for the original OTSG design does not conform to Criterion 6 in 10 CFR 54.3(a) because the analysis is no longer contained or incorporated by reference in the updated CLB for the replacement OTSGs. Therefore, based on this review, the staff finds that the applicant has provided an acceptable basis for deleting the flaw evaluation TLAA for the OTSGs from the scope of LRA Table 4.1-1 and LRA Section 4.7.5.2 because the staff has verified the following:

(a) The analysis does not conform to Criterion 6 in 10 CFR 54.3(a).

(b) The analysis does not meet the definition of a TLAA, as defined in 10 CFR 54.3(a), for the updated CLB that applies to the replacement OTSGs.

4.7.5.2.3 USAR Supplement

As revised in LRA Amendment 8 by letter dated June 3, 2011, LRA Section A.2.6.2 provides the USAR supplement summary description for the TLAA of the OTSG 1-2 flaw evaluations. LRA Amendment 8 revised the disposition for the analysis of the OTSG 1-2 flaws in LRA Section A.2.6.2 from 10 CFR 54.21(c)(1)(iii) to 10 CFR 54.21(c)(1)(i), consistent with the revised disposition identified in LRA Amendment 8, Section 4.7.5.2. The staff initially reviewed the applicant's amended USAR supplement summary description for this TLAA and determined that it is consistent with the TLAA discussed in LRA Section 4.7.5.2, as amended. On June 23, 2014, the applicant provided an LRA update and deleted USAR Supplement A.2.6.2 from the LRA. The applicant stated that, as a result of OTSG replacement activities that were implemented in the Spring 2014 refueling outage, LRA Section A.2.6.2, "OTSG 1-2 Flaw Evaluations," previously revised by FENOC letter dated June 3, 2011 (ADAMS Accession No. ML11159A132), is no longer applicable to the CLB for the replacement OTSGs at the plant. The applicant deleted LRA Section A.2.6.2 from the scope of the LRA. The staff finds this change acceptable because the previous flaw evaluation for the original OTSGs is not applicable to the updated CLB for the replacement OTSGs at the plant.

Therefore, based on the OTSG replacements, the staff finds that the USAR supplement no longer needs to include a USAR supplement summary description for the flaw evaluation that was performed for the original OTSGs at the plant because the staff has verified that the flaw evaluation is not applicable to the updated CLB for the replacement OTSGs.

4.7.5.2.4 Conclusion

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration that the CLB no longer contains any referenced flaw evaluation for the OTSGs that conforms to the definition of a TLAA in 10 CFR 54.3(a).

SECTION 5

REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In accordance with Title 10 of the *Code of Federal Regulations* Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," the Advisory Committee on Reactor Safeguards (ACRS) reviewed the license renewal application (LRA) for Davis-Besse Nuclear Power Station (Davis-Besse). On November 4, 2015, FirstEnergy Nuclear Operating Company, and the staff of the United States (U.S.) Nuclear Regulatory Commission (NRC) (the staff) met with the full ACRS committee to discuss issues associated with the review of the LRA. On November 12, 2015, the full committee issued its report discussing the results of the review. Copies of the ACRS letter and the staff's response are provided on the following pages of this SER section.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 12, 2015

The Honorable Stephen G. Burns
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR DAVIS-BESSE NUCLEAR POWER STATION

Dear Chairman Burns:

During the 629th meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 4 -7, 2015, we completed our review of the license renewal application (LRA) for Davis-Besse Nuclear Power Station. Our review included the NRC staff final safety evaluation report (SER) issued in September 2013 and SER Supplement 1 issued in August 2015. Our Subcommittee on Plant License Renewal reviewed this matter during meetings on September 19, 2012 and September 23, 2015. During these reviews, we had the benefit of discussions with representatives of the NRC staff and First Energy Nuclear Operating Company (FENOC, or the applicant). We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATIONS

1. The established programs and commitments by FENOC to manage age-related degradation, including LRA Amendment 60 submitted in October 2015, provide reasonable assurance that the Davis-Besse Nuclear Power Station can be operated in accordance with the current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

2. An amendment to the current license to include the methodologies used to analyze the effects of concrete cracking in the shield building should be completed prior to commencement of the period of extended operation.

3. FENOC's application for renewal of the operating license for the Davis-Besse Nuclear Power Station should be approved.

BACKGROUND

Davis-Besse is located approximately 20 miles east of Toledo, Ohio. The NRC issued the construction permit on March 24, 1971 and the operating license on April 22, 1977. The unit is a pressurized water reactor design with a dry ambient containment which consists of a free-standing steel pressure vessel surrounded by a reinforced concrete shield building, with a 4.5-foot annulus between the pressure vessel and shield building. Babcock and Wilcox Corporation supplied the nuclear steam supply system, and Bechtel Corporation designed and constructed the balance of plant. The licensed power output of the unit is 2,817 megawatts thermal, with a gross electrical output of approximately 908 megawatts electric.

By letter dated August 27, 2010, FENOC submitted its LRA to the NRC for renewal of the Davis-Besse operating license for an additional 20 years, commencing with expiration of its current license on April 22, 2017.

In 2011, during construction of an access opening to replace the reactor pressure vessel head, FENOC identified laminar subsurface cracking of concrete along the outer mat of reinforcing steel in particular regions of the shield building parallel to the cylindrical wall. The extent of the cracking was carefully mapped, and characteristics such as crack width and crack depth from the outer wall surface were determined. In 2013 and 2015, follow-up inspections determined that limited growth of cracks mapped in 2011 had occurred in some areas. FENOC conducted analyses and testing to demonstrate that the shield building continues to perform its design function in accordance with the current licensing basis, with ample margin. The design function is to provide biological shielding, environmental protection for the steel containment pressure vessel, and for a controlled release of the annulus atmosphere under accident conditions.

In 2014, the Davis-Besse original steam generators were replaced, and they are currently in their first cycle of operation with no identified tube degradation mechanisms.

DISCUSSION

NRC Staff Review of LRA

In the final SER dated September 2013, and the Supplemental SER dated August 2015, the staff documented its review of the LRA and other information submitted by the applicant and obtained through staff audits and inspections at the plant site. The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) that are within the scope of license renewal, the integrated plant assessment process, the identification of plausible aging mechanisms associated with passive, long-lived components, the adequacy of the Aging Management Programs (AMPs), and the identification and assessment of Time-Limited Aging Analyses (TLAAs) requiring review.

FENOC's license renewal application demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 1) and documents and justifies deviations from the specified approaches in that report. FENOC will implement 44 AMPs for license renewal at Davis-Besse. The AMPs consist of 31 existing programs and 13 new programs.

For the 13 new AMPs, five of the programs are consistent with the GALL Report, two are consistent with enhancements, and six are plant-specific.

Of the 31 existing programs, nine are consistent with the GALL Report, eleven are consistent with enhancements, two are consistent with exceptions, and five are consistent with both enhancements and exceptions. Four of the existing programs are plant-specific.

The LRA includes seven exceptions to the GALL Report. The exceptions are in the areas of the Bolting Integrity Program, the Closed Cooling Water Chemistry Program, the Fire Protection Program, the Fuel Oil Chemistry Program, the Open-Cycle Cooling Water Program, the Water Control Structures Inspection Program, and the Reactor Head Closure Studs Program. We reviewed all of these exceptions and consider them to be acceptable.

The staff conducted license renewal audits and performed license renewal inspections at Davis-Besse. The audits verified the appropriateness of the scoping and screening methodology for AMPs, the appropriateness of the aging management review, and the acceptability of the TLAAs. The inspections verified that the license renewal requirements will be implemented appropriately. The inspections, and the reports of those inspections, as documented in the SER and SER Supplement 1, were thorough.

Based on the audits, the inspections, and the staff reviews related to this license renewal application, the staff concluded that the proposed activities will manage the effects of aging of the SSCs and that the intended functions of these SSCs will be maintained consistent with the current licensing basis for the period of extended operation as required by 10 CFR 54.21(a)(3). We concur with that conclusion, subject to the addition of the inspection enhancements included in LRA Amendment 60.

Shield Building AMP - Background

Among the AMPs which are specific to Davis-Besse is the Shield Building AMP. It is also unique to Davis-Besse due to the need for the AMP to address previously identified laminar subsurface cracking of the concrete. This cracking was discovered by FENOC in 2011, and a thorough inspection was then conducted of the entire cylindrical structure. A root cause analysis attributed the laminar cracks to the extreme environmental conditions of a blizzard in 1978. The cracks were initiated when moisture within the concrete froze and expanded in regions of high concrete stress. At the time, the building did not have an exterior coating to prevent moisture intrusion. In order to prevent additional moisture intrusion, a coating was subsequently applied to the shield building exterior. Also, FENOC established a shield building crack monitoring program.

In 2013 and 2015, monitoring program inspections determined that limited additional crack growth has occurred in some areas. A root cause analysis attributed this growth to moisture repeatedly freezing and expanding at the tips of the laminar cracks which had been initiated during the 1978 blizzard. This moisture had entered the concrete prior to application of the coating in 2012.

In accordance with American Concrete Institute Report ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," the propagation of the laminar cracks is a condition which is not passive and requires ongoing monitoring. The Shield Building AMP includes an extensive monitoring program for this purpose.

Cracking is of concern because the concrete must transfer rebar tension loads across the overlap splices which exist between lengths of horizontal rebar. This transfer occurs through shear loading between the concrete and the rebar at numerous splices throughout the cylindrical structure. In the presence of cracking, the effectiveness of the transfer of shear loads depends on the crack width not exceeding an acceptable size which has been established by testing. The applicant sponsored test programs at Purdue University and the University of Kansas that showed for crack geometries comparable to those in the shield building, the horizontal splices in the cracked regions are able to carry at least 90% of their original design capacity. Thus, measuring the width of the cracks, as well as mapping their extent, is an important element of the Shield Building AMP.

Shield Building AMP – Inspection Program

In LRA Amendment 60, dated October 6, 2015, FENOC provided a description of, and committed to, an updated shield building monitoring program, including enhancements to the primary method of monitoring the areal extent of laminar cracking. This method is a non-destructive technique known as impulse response mapping. It involves monitoring and recording the reflection of a manual impulse to the building exterior wall to determine if a crack exists within the concrete at the location of the impulse.

The monitoring program also includes visual inspections of the cracks themselves using core bores perpendicular to the building exterior wall. The core bores allow both inspection for the presence of cracking at the bore location and measurement of the width of the crack at that location. A narrow crack width is critical to the ability of the concrete to transfer in shear the rebar tension loads at a splice location where cracking exists. The AMP includes a limit on the width so that it does not exceed the size which has been demonstrated by testing to reliably transfer loads across a rebar splice.

Shield Building AMP – Margins Evaluation

In order to evaluate the effectiveness of the Shield Building AMP during the period of extended operation, we examined the applicant's calculations which demonstrate that the intended functions will be maintained under design basis loadings. Initially, FENOC performed structural integrity calculations pursuant to 10 CFR 50.59, which were reviewed by the NRC staff as part of the operability assessment of the shield building. It was subsequently determined that the analysis methodologies should also be submitted for approval as an amendment to the current license. In order for us to assess the margins for the purpose of evaluating the Shield Building AMP, the calculations were provided to the NRC docket for review in advance of submittal of the amendment to the current license.

In a letter dated October 6, 2015, FENOC provided additional information which clarified the margins with respect to the cracked area. Results of the calculations indicate that the cracked area can increase by a factor of at least 2.6 and the shield building will continue to perform its intended functions under design basis loads. Based on our review of this additional information, we conclude that adequate margin exists such that the AMP, with the enhanced inspections as provided in LRA Amendment 60, will assure against loss of intended functions due to crack area propagation.

With regard to crack width, the Shield Building AMP requires that any width greater than 0.013 inch would require further evaluation and testing in order to continue to credit load transfer at affected splices. We conclude that adequate margin exists for crack width such that the AMP, with the enhanced inspections as provided in LRA Amendment 60, will assure against loss of intended functions due to increases in crack width.

As noted, our review and the resulting conclusions concerning Shield Building AMP adequacy were based on the results of the analysis methodologies used by FENOC. To support these conclusions, an amendment to the current license to include the methodologies used to analyze the effects of concrete cracking in the shield building should be completed prior to commencement of the period of extended operation.

Our review also included consideration of two other potential cracking effects. The first potential effect we considered was whether existing cracks would propagate sufficiently to challenge the intended building functions during a design-basis seismic event. In those areas where the thickness of concrete between the crack layer and the exterior surface provides sufficient mass to produce large inertial forces under seismic cyclic loading, closely spaced rebar perpendicular to the exterior surface ties the concrete to the inner rebar mat of the building. Accordingly, in these areas, inertial forces due to seismic cyclic loading are transferred into the inner rebar mat, and crack propagation due to cyclic loading will not occur. In other areas, where the layer of concrete outside a laminar crack is thinner, inertial forces under seismic cyclic loading of the much smaller mass are not expected to result in significant crack propagation during the limited duration of the seismic event.

The second potential effect was that the laminar cracks could permit separation of pieces of concrete under design loading conditions. Sufficiently heavy sections of falling concrete could impact and damage adjacent safety-related SSCs. However, as noted above, the thicker concrete areas are tied by rebar to the building interior rebar mat and are not subject to this failure, even if cracked. The thinner concrete areas are not similarly tied to the inner rebar mat, but the mass of potential pieces which might become separated from the building due to cracking during a seismic event is estimated to be well below the SSC design requirements for protection against tornado missiles. Therefore, cracking in the thinner areas does not represent a threat to SSCs.

Containment Vessel

In its foundation area, the steel containment vessel is subject to wetting on its exterior due to the intrusion of groundwater past a protective membrane. It has also been exposed to water accumulation on its interior surface due to leakage from the refueling canal. That source of leakage was subsequently repaired. The applicant has performed inspections to verify that corrosion has not reduced the thickness of the vessel wall to less than the minimum required. These inspections will be repeated during the period of extended operation prior to the end of 2025. Also, there is an ongoing program providing for visual inspections and sampling of any water accumulation in the annular area at the base of the containment vessel. These measures, which are included in License Renewal Commitments 35 and 36, provide adequate assurance that corrosion will not threaten the containment function.

SUMMARY

There are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the Davis-Besse operating license. The established programs and commitments by FENOC provide reasonable assurance that Davis-Besse can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The FENOC application for renewal of the operating license for Davis-Besse should be approved.

Sincerely,

/RA/

John W. Stetkar
Chairman

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 1, 2015

John W. Stetkar, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

Dear Mr. Stetkar:

During the 629th meeting of the Advisory Committee on Reactor Safeguards (ACRS) held November 4-7, 2015, the Committee completed its review of the license renewal application (LRA) for Davis-Besse Nuclear Power Station, Unit 1. The Committee also reviewed the U.S. Nuclear Regulatory Commission (NRC) staff's associated final safety evaluation report for this application.

The NRC staff appreciates the Committee's objective and in-depth review of the Davis-Besse license renewal application and final safety evaluation report under the requirements of 10 CFR Part 54, "Requirements for Renewal of Operating License for Nuclear Power Plants." In the Committee's final report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15316A125), dated November 12, 2015, it concurred with the NRC staff's conclusion on the Davis-Besse, Unit 1 license renewal. Both the Committee and the NRC staff believe that the licensee, FirstEnergy Nuclear Operating Company (FENOC), has demonstrated that it will adequately manage the effects of aging and maintain intended functions consistent with the current licensing basis for the period of extended operation (as required by 10 CFR 54.21(a)(3)). The Committee further concurs with the NRC staff that there are no open license renewal issues related to 10 CFR 54.29(a)(1) or 10 CFR 54.29(a)(2) that would prevent the NRC from renewing the Davis-Besse operating license.

In accordance with the Atomic Energy Act of 1954, as amended, the Committee has statutory responsibilities to review and advise the Commission on the licensing and operation of production and utilization facilities and related safety issues. Under this statutory authority, the Committee conducted a thorough and in-depth review of the methodologies that FENOC used to analyze cracking effects that could impact the intended design function of the shield building under design basis loadings, including seismic events. Based on its independent review, the Committee concluded on the basis of the results of the analysis methodologies used by FENOC that adequate margin exists such that the shield building remains capable of performing its intended functions. The staff position is consistent with this conclusion (as documented in Inspection Reports (IR) 05000346/2012009 and 05000346/2014008). The Committee further indicated that the shield building monitoring aging management program, as described in LRA Amendment No. 60, is sufficient to prevent loss of intended functions due to potential growth of

the shield building cracking effects in accordance with the current licensing basis for the period of extended operation. In its final report, the Committee expressed the importance of the NRC staff's review and approval of the methodologies used to analyze the effects of concrete cracking in the shield building (cracking effects analysis methodologies) as an amendment to Davis-Besse's current license. The staff agrees with the importance of addressing this current licensing basis issue related to laminar cracking of the shield building under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and notes that the cracking effects analysis issue is currently being addressed by the staff under the NRC Enforcement Policy. Specifically, the staff issued a Severity Level IV non-cited violation of 10 CFR 50.59(c)(2) and an associated Green finding in IR 05000346/2014008 (ADAMS Accession No. ML15148A489) in May 2015 for FENOC's failure to request and obtain a license amendment under the regulations of 10 CFR 50.90. At this time, FENOC is expected to submit a license amendment to address this issue. However, the staff review of an amendment to the license addressing cracking effects analysis methodologies may not be completed prior to commencement of the period of extended operation. Since it is a current operating issue, review and approval of the cracking effects analysis methodologies will be conducted by staff as part of normal licensing activity under 10 CFR Part 50 and corrective action for the violation is to be reviewed by inspectors as part of the Reactor Oversight Program. Therefore, addressing these current operational activities does not preclude license renewal, and the staff is not conditioning the license renewal as suggested in item two of the Committee's final letter.

The NRC staff acknowledged the insights provided by the ACRS during the Subcommittee Meeting held on September 23, 2015, regarding recommended improvements to the Davis-Besse shield building monitoring program. On October 6, 2015, FENOC submitted revisions to the shield building monitoring program. The revisions, evaluated by the staff, include enhancements to the method of monitoring the areal extent of laminar cracking to require impulse response mapping, in addition to visual inspections of the cracks themselves using core bores perpendicular to the building exterior wall.

The NRC staff recognizes and appreciates the Committee's commitment to safety and appreciates its continued support of the license renewal process.

Sincerely,

/RA/

Victor M. McCree
Executive Director
for Operations

cc: Chairman Burns
Commissioner Svinicki
Commissioner Ostendorff
Commissioner Baran
SECY

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The NRC staff recognizes and appreciates the Committee's commitment to safety and appreciates its continued support of the license renewal process.

Sincerely,
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Victor M. McCree
Executive Director
for Operations

cc: Chairman Burns
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SECY

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Letter to J. Stetkar from V. McCree dated December X, 2015

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SECTION 6

CONCLUSION

The staff concludes that the additional information provided by FirstEnergy Nuclear Operating Company does not alter the conclusion proffered in the safety evaluation report issued in September 2013 and that the requirements of 10 CFR 54.29(a) have been met.

APPENDIX A

DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS

During the review of the Davis-Besse Nuclear Power Station (Davis-Besse) license renewal application by the staff of the U.S. Nuclear Regulatory Commission (NRC), FirstEnergy Nuclear Operating Company (FENOC) made commitments related to aging management programs (AMPs) to manage aging effects for structures and components.

The following table contains the final complete list of these commitments, along with the implementation schedules and sources for each commitment.

License Renewal Future Commitments

(Through LRA Amendment 59)

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
1.	Enhance the Aboveground Steel Tanks Inspection Program to: • Include a volumetric examination of tank bottoms to detect evidence of loss of material due to crevice, general, or pitting corrosion, or to confirm a lack thereof. Establish the examination technique, the inspection locations, and the acceptance criteria for the examination of the tank bottoms. Require that unacceptable inspection results be entered into the FENOC Corrective Action Program. The volumetric examination of the tank bottoms will be performed within 5 years after entering the period of extended operation. Additional opportunistic tank bottom inspections will be performed whenever the tanks are drained. • Include tank inspections conducted in accordance with Table 4a, “Tank Inspection Recommendations,” of License Renewal Interim Staff Guidance LR-ISG-2012-02, “Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation.” • Include an inspection of the borated water storage tank (BWST) exterior surface prior to the period of extended operation for loss of material and cracking. Sufficient insulation will be removed to determine the condition of the exterior surface of the tank. At a minimum, either 25 1-square-foot sections or 20 percent of the surface area of insulation will be removed to permit inspection of the exterior surface of the tank. The sample inspection points will be distributed in such a way that inspections will be performed near the tank bottom, at points where structural supports, pipe, or instrument nozzles penetrate the insulation and where water could collect, such as on top of stiffening rings. In addition, inspection	A.1.2 B.2.2 Responses to NRC RAIs B.2.2-1 from NRC Letter dated April 20, 2011; RAI A.1-1 from NRC Letter dated March 26, 2013; NRC LR-ISG-2012-02, and NRC RAI 3.0.3.4.3-02 from NRC Letter dated July 7, 2014	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-13-160, L-14-085, and L-14-244

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	locations will be based on the likelihood of corrosion under insulation occurring. As an alternative to removing the insulation, subsequent inspections may consist of an examination of the exterior surface of the insulation for indications of damage to the protective outer layer of the insulation when the results of the initial inspection meet the following criteria: 1. no loss of material due to general, pitting, or crevice corrosion, beyond that which could have been present during initial construction, is observed, and 2. no evidence of stress corrosion cracking (SCC) is observed. The subsequent inspections will be performed during each 10-year period of the period of extended operation. If these subsequent inspections reveal damage to the exterior surface of the insulation, or there is evidence of water intrusion through the insulation, periodic inspections under the insulation will continue as conducted for the initial inspection and will be performed during each 10-year period of the period of extended operation.			
2.	Implement the Boral® Monitoring Program as described in LRA Section B.2.5.	A.1.5 B.2.5 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160
3.	Enhance the Buried Piping and Tanks Inspection Program to: • Add (1) bolting for buried Fire Protection System piping and (2) the emergency diesel fuel oil storage tanks (DB-T153-1, DB-T153-2) to the scope of the program. • Conduct annual ground potential surveys of the cathodic protection system. Monitor cathodic protection voltage	A.1.7 B.2.7 Responses to NRC RAIs B.2.7-1 from NRC Letter	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-13-160, L-13-304

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	and current monthly to determine the effectiveness of cathodic protection systems and, thereby, the effectiveness of corrosion mitigation. Trend voltage, current, and ground potential readings and evaluate for adverse changes. • Require that the activity of the jockey fire pump or equivalent parameter be monitored on at least a monthly interval. Conduct a flow test by the end of the next refueling outage when unexplained changes in jockey pump activity are observed. • Require that the directed buried pipe inspection locations be selected based on risk. • Require that the minimum number of buried in-scope piping inspections during the 30-40, 40-50, and 50-60-year operating period is one steel piping segment. Perform the directed buried steel pipe inspections each 10-year interval, based upon Table 4a, "Inspections of Buried Pipe," in the XI.M41 aging management program described in LR-ISG-2011-03. Each inspection will have a minimum of 10 feet of piping inspected. • Require that, IF the cathodic protection system for the emergency diesel generator (EDG) fuel oil storage tanks (DB-T153-1 and DB-T153-2) meets the availability criteria of Table 4c, "Inspections of Buried Tanks for all Inspection Periods" (i.e., footnotes 3.i, 3.ii and 3.iii) of LR-ISG-2011-03, Appendix A, "Revised GALL Report AMP XI.M41," THEN no Table 4c inspections of tanks DB-T153-1 and DB-T153-2 are required. Otherwise, perform inspections of tanks DB-T153-1 and DB-T153-2, in accordance with Table 4c of LR-ISG-2011-03. • Require that ultrasonic testing (UT) thickness measurements of the manways and vents for EDG fuel oil storage tanks T153-1 and T153-2 will be performed prior to entering the period of extended operation and every 10 years during the period of extended operation	dated April 20, 2011; A.1-1 from NRC Letter dated March 26, 2013; and 2011-03-1 and 2011-03-2 from NRC Letter dated February 11, 2014		and L-14-114

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	to ensure that the metal thickness in those areas remains satisfactory. • Require that underground piping in the decay heat removal and low pressure injection system located in the borated water piping trench will be visually inspected during the 30-40, 40-50, and 50-60-year operating periods to confirm the absence of aging effects. • Require that, if adverse indications are detected, the inspection sample sizes, within the affected piping categories, are initially doubled and if adverse conditions are discovered in the expanded sample, the size of the follow-on inspections is determined by establishing the extent of condition and extent of cause, consistent with the FENOC Corrective Action Program. Scheduling of additional examinations is based on the severity of the degradation identified and commensurate with the consequences of a leak or loss of function, but in all cases, the expanded sample inspection should be completed within the 10-year interval in which the original adverse indication was identified. Further inspections are conducted in locations with similar materials and environment, or the piping is replaced on a schedule based upon either the station's need to return the system to service for non-Technical Specification-related systems or the allowed outage time for Technical Specification-related systems. • Require that an inspection of buried fire protection system bolting will be performed, when the bolting becomes accessible during opportunistic or focused inspections. • Require that the inspections of buried piping be conducted using visual (VT-3 or equivalent) inspection methods. Excavation shall be a minimum of 10 linear feet of piping, with all surfaces of the pipe exposed. • Include the following acceptance criteria in the program procedure:			

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	○ The cathodic protection survey acceptance criteria for protected piping and tanks, with the exception of the manways and vents at the top of the mound over EDG fuel oil storage tanks T153-1 and T153-2, are the -850 mV relative to a copper/copper sulfate reference electrode (CSE), instant off and limiting critical potential not more negative than 1200 mV. For the manways and vents at the top of the mound over tanks T153-1 and T153-2, the acceptance criterion is the 100 mV minimum polarization testing criteria listed in National Association of Corrosion Engineers (NACE) SP0169 2007; ○ 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 Coating Inspector Program Level 2 or 3 inspector qualification or an individual has attended the Electric Power Research Institute (EPRI) Comprehensive Coatings Course and completed the EPRI Buried Pipe Condition Assessment and Repair Training Computer-Based Training Course. Where damage to the coating has been evaluated as significant and the damage was caused by nonconforming backfill, an extent-of-condition evaluation should be conducted to ensure that the as-left condition of backfill in the vicinity of observed damage will not lead to further degradation; ○ If 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; and, ○ Changes in jockey pump activity or equivalent parameter that cannot be attributed to causes other than leakage from buried piping are not occurring.			

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
4.	Implement the Collection, Drainage, and Treatment Components Inspection Program as described in LRA Section B.2.9.	A.1.9 B.2.9 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160
5.	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection as described in LRA Section B.2.11. Enhance the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection to: Include high-voltage connections to confirm the absence of aging effects for metallic electrical connections.	A.1.11 B.2.11 Responses to NRC RAIs 3.6-3 from NRC Letter dated April 5, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-134 and L-13-160
6.	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.12.	A.1.12 B.2.12 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
7.	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program as described in LRA Section B.2.13.	A.1.13 B.2.13 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160
8.	Enhance the External Surfaces Monitoring Program to: • Add to the scope of the program systems that credit the program for license renewal but do not have Maintenance Rule intended functions. • Perform opportunistic inspections of surfaces that are inaccessible or not readily visible during normal plant operations or refueling outages, such as surfaces that are insulated. Surfaces that are accessible will be inspected at a frequency not to exceed one refueling cycle. • Perform, in conjunction with the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program, inspection and surveillance of elastomers and polymers exposed to air-indoor uncontrolled or air-outdoor environments, but not replaced on a set frequency or interval (i.e., are long-lived), for evidence of cracking and change in material properties (hardening and loss of strength) and loss of material due to wear. Specify acceptance criteria of no unacceptable visual indications of cracks or discoloration that would lead to loss of function prior to the next inspection, and of no hardening as evidenced by a loss of suppleness during manipulation. • Perform inspection of the control room emergency ventilation system air-cooled condensing unit cooling coil tubes and fins and the station blackout diesel generator radiator tubes and fins for visible evidence of	A.1.15 B.2.15 Responses to NRC RAIs 3.3.2.2.5-1 and B.2.2-2 from NRC Letter dated April 20, 2011; RAI 3.3.2-2 from NRC Letter dated May 2, 2011; RAI 3.3.2.2.5-2 from NRC Letter dated July 12, 2011; Supplemental RAI OIN-352 from NRC Region III IP-71002 Inspection; RAI A.1-1 from NRC Letter dated March 26, 2013; and NRC LR-ISG-2012-02	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-11-166, L-11-238, L-13-160, and L-14-085

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	external surface conditions that could result in a reduction in heat transfer. Specify acceptance criteria of no unacceptable visual indications of fouling (build up of dirt or other foreign material) that would lead to loss of function prior to the next scheduled inspection. • Manage cracking of copper alloys with greater than 15-percent zinc and stainless steel components exposed to an outdoor air environment through plant system inspections and walkdowns for evidence of leakage. Specify acceptance criteria of no unacceptable visual indications of cracks that would lead to loss of function prior to the next scheduled inspection. • Include inspection parameters and acceptance criteria for polymers, elastomers and metallic components as applicable in system inspection and walkdown documentation. Retain system inspection and walkdown documentation in plant records. • Inspect or remove portions of insulation from outdoor insulated components, and indoor insulated components exposed to condensation (because the in-scope component is operated below the dew point), to determine whether the exterior surface of the component is degrading or has the potential to degrade. Inspect a minimum of 20 percent of the in-scope piping length, or 20 percent of the surface area for components whose configuration does not conform to a 1-foot axial length determination (e.g., valve, accumulator), after the insulation is removed. Alternatively, any combination of a minimum of 25 1-foot axial length sections and components for each material type is inspected. Inspection locations should focus on the bounding or lead components most susceptible to aging because of time in service, severity of operating conditions (e.g., amount of time that condensate would be present on the external surfaces of the component), and lowest design margin. The inspections will be conducted during			

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	each 10-year period of the period of extended operation. The following are alternatives to removing insulation: a. Subsequent inspections may consist of an examination of the exterior surface of the insulation with sufficient acuity to detect indications of damage to the jacketing or protective outer layer of the insulation when the results of the initial inspection meet the following criteria: i. No loss of material due to general, pitting, or crevice corrosion, beyond that which could have been present during initial construction is observed, and ii. No evidence of SCC is observed. If the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or jacketing, or there is evidence of water intrusion through the insulation (e.g., water seepage through insulation seams/joints), periodic inspections under the insulation should continue as conducted for the initial inspection. b. Removal of tightly adhering insulation that is impermeable to moisture is not required unless there is evidence of damage to the moisture barrier. If the moisture barrier is intact, the likelihood of corrosion under insulation (CUI) is low for tightly adhering insulation. Tightly adhering insulation is considered to be a separate population from the remainder of insulation installed on in-scope components. The entire population of in-scope piping that has tightly adhering insulation is visually inspected for damage to the moisture barrier with the same frequency as for other types of insulation inspections. These inspections are not credited towards the inspection quantities for other types of insulation.			

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
9.	Enhance the Fatigue Monitoring Program to: • Provide for updates of the fatigue usage calculations on an as-needed basis if an allowable cycle limit is approached. When the number of accrued cycles is within 75% of the allowable cycle limit for any transient, a condition report will be generated. For any transient whose cycles are projected to exceed the allowable cycle limit by the end of the next plant operating cycle (Davis-Besse operating cycles are normally 2 years in duration), the program will require an update of the fatigue usage calculation for the affected component(s). • Establish an acceptance criterion for maintaining the cumulative fatigue usage below the Code design limit of 1.0 through the period of extended operation, including environmental effects where applicable.	A.1.16 B.2.16 Responses to NRC RAIs B.2.16-3, B.2.16-4 and B.2.16-5 from NRC Letter dated April 20, 2011 and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-166 and L-13-160
10.	Enhance the Fire Water Program to: • Include inspections and testing conducted in accordance with Appendix D, Table 4a, "Fire Water System Inspection and Testing Recommendations," of LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion Under Insulation." • Include augmented testing and inspections beyond those of Table 4a for portions of water-based fire protection system components that are (a) normally dry but periodically subjected to flow (e.g., dry-pipe or pre-action sprinkler system components) and (b) cannot be drained or allow water to collect: 1. In each 5-year interval, beginning 5 years prior to the period of extended operation, a flow test or flush sufficient to detect potential flow blockage will be conducted, or a visual inspection of 100 percent of the internal surface of piping segments will be conducted. 2. In each 5-year interval of the period of extended operation, 20 percent of the length of piping segments	A.1.18 B.2.18 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013; NRC LR-ISG-2012-02; and NRC RAI B.2.18-2 from NRC Letter dated July 7, 2014	Prior to October 22, 2016	LRA and FENOC Letters L-13-160, L-14-085, and L-14-244

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	that cannot be drained or piping segments that allow water to collect will be subject to volumetric wall thickness inspections. Measurement points are obtained to the extent that each potential degraded condition can be identified (e.g., general corrosion, microbiologically-influenced corrosion (MIC)). The 20 percent of piping that is inspected in each 5-year interval will be in different locations than previously inspected piping. If the results of a 100-percent internal visual inspection are acceptable, and the segment is not subsequently wetted, no further augmented tests or inspections are necessary. • Perform representative sprinkler head sampling (laboratory field service testing) or replacement prior to 50 years inservice (installed), and at 10-year intervals thereafter, in accordance with the 2011 Edition of National Fire Protection Association (NFPA) 25, or until there are no untested sprinkler heads that will see 50 years of service through the end of the period of extended operation. • Include a requirement that, when visual inspections are used to detect loss of material, the inspection technique is capable of detecting surface irregularities that could indicate wall loss to below nominal pipe wall thickness due to corrosion and corrosion product deposition. Where such irregularities are detected, followup volumetric wall thickness examinations are performed. • Include a requirement that, if the presence of sufficient foreign organic or inorganic material to obstruct pipe or sprinklers is detected during pipe inspections, the material is removed and its source is determined and corrected.			

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
11.	Implement the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.21. Enhance the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program to: • Include inaccessible underground lower service voltage cables (400VAC to 2kV). • Not use 'significant voltage' (defined as being subjected to system voltage for more than 25 percent of the time) as a criterion for inclusion into the program. • Include inspection of electrical manholes that contain power cables within the scope of the program. • Inspect electrical manholes at least once per year. The frequency of inspections for accumulated water will be established and adjusted based on plant-specific inspection results. Also, manhole inspections will be performed in response to event-driven occurrences (e.g., heavy rain or flooding). • Include a requirement in preventive maintenance (PM) activities PM 4297, PM 4294, PM 8025, and PM 4296 to generate a condition report in cases where in-scope inaccessible non-environmental qualification (EQ) power cable manhole inspection identifies submerged cables. Although the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program, preventive maintenance activities exist for inspection of water accumulation in the manholes associated with the in-scope inaccessible non-EQ power cables. • Perform cable testing on a frequency of at least every 6 years. Testing will be evaluated for more frequent performance based on test results and operating experience.	A.1.21 B.2.21 Responses to NRC RAIs B.2.21-1 and B.2.21-3 from NRC Letter dated April 5, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-134 and L-13-160

APPENDIX A: DAVIS-BESSE NUCLEAR POWER STATION LICENSE RENEWAL COMMITMENTS				
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12.	Enhance the Masonry Wall Inspection Program to: • Include and list the structures within the scope of license renewal that credit the program for aging management. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Specify that, for each masonry wall, the extent of observed masonry cracking or degradation of steel edge supports or bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking or steel degradation is sufficient to invalidate the evaluation basis. An option is to develop a new evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation). • Specify that, for the masonry walls within the scope of license renewal, inspections will be conducted at least once 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.	A.1.27 B.2.27 Responses to NRC RAIs B.2.39-5 from NRC Letter dated April 5, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153 and L-13-160

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13.	Implement the One-Time Inspection as described in LRA Section B.2.30.	A.1.30 B.2.30 Responses to NRC RAIs 3.3.2.2.4.3-1 from NRC Letter dated May 2, 2011; Supplemental Question – Makeup Pump Casing Inspections, A.1-1 from NRC Letter dated March 26, 2013; and 2014 Annual Update	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-11-166, L-11-218 L-11-237, L-11-252, L-13-160, and L-14-206
14.	Implement the PWR Reactor Vessel Internals Program as described in LRA Section B.2.32.	A.1.32 B.2.32 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160
15.	In association with the PWR Reactor Vessel Internals Program, a plant-specific inspection plan for ensuring the implementation of Materials Reliability Program (MRP)-227 guidelines, as amended by the safety evaluation for MRP-227, and Davis-Besse's responses to the plant-specific action items, as identified in Section 4.2 of the safety evaluation for MRP-227, will be submitted for NRC review and approval. * NOTE: The inspection plan will be submitted no later than 2 years after issuance of the renewed operating license or 2 years prior to the beginning of the period of extended operation (April 22, 2015), whichever is earlier.	A.1.32 B.2.32 Response to NRC RAI B.2.32-1 from NRC Letter dated July 11, 2011	Complete	LRA and FENOC Letter L-11-252, and L-15-214

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16.	Enhance the Reactor Head Closure Studs Program as follows: • Select an alternative stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant for the reactor head closure stud assemblies will be included in the program. • Preclude the future use of replacement closure stud bolting fabricated from material with actual measured yield strength greater than or equal to 150 kilo-pounds per square inch (ksi), except for use of the existing spare reactor head closure stud bolting.	A.1.34 B.2.34 Responses to NRC RAIs B.2.34-1 from NRC Letter dated June 20, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-218 and L-13-160
17.	Enhance the Reactor Vessel Surveillance Program as follows: • The Capsule Insertion and Withdrawal Schedule for Davis-Besse will be revised to schedule testing of the TE1-C capsule.	A.1.35 B.2.35 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160
18.	Implement the Selective Leaching Inspection as described in LRA Section B.2.36.	A.1.36 B.2.36 Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-13-160

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19.	Implement the Small Bore Class 1 Piping Inspection as described in LRA Section B.2.37.	A.1.37 B.2.37 Responses to NRC RAIs B.2.37-2 from NRC Letter dated April 20, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Completed within the six year period prior to October 22, 2016	LRA and FENOC Letters L-11-153 and L-13-160
20.	Enhance the Structures Monitoring Program to: • Include and list the structures within the scope of license renewal that credit the program for aging management. • Include aging effect terminology (e.g., loss of material, cracking, change in material properties, and loss of form). • List American Concrete Institute (ACI) 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," and American National Standards Institute/American Society of Civil Engineers (ANSI/ASCE) 11-90 as references and indicate that they provide guidance for the selection of parameters monitored or inspected. • Clarify that a "structural component" for inspection includes each of the component types identified within the scope of license renewal as requiring aging management. • Require the responsible engineer to review site raw water pH, chlorides, and sulfates test results prior to the inspection to take into account the raw water chemistry for any unusual trends during the period of extended operation. Raw water chemistry data shall be collected at least once every 5 years. Data collection dates shall	A.1.39 B.2.39 Responses to NRC RAIs B.2.39-4, B.2.39-5, B.2.39-6 and B.2.39-7 from NRC Letter dated April 5, 2011; RAIs B.2.39-11 and 3.5.2.3.12-4 from NRC Letter dated July 21, 2011; Supplemental RAI B.2.39-11 from telecon held with the NRC on September 13, 2011; Supplemental RAI OIN-380 from Region III IP-71002 Inspection; RAI B.2.4-1a from NRC Letter dated November 14, 2012; RAI B.2.43-3a from NRC Letter dated January 4, 2013;	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-11-237, L-11-292, L-11-317, L-12-455, L-13-037, and L-13-160

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	be staggered from year to year (summer-winter-summer) to account for seasonal variation. • Perform an inspection for loss of material for carbon steel structural components subject to aggressive groundwater. Require the use of the FENOC Corrective Action Program for identified concrete or steel degradation. • Specify that, upon notification that a below-grade structural wall or other in-scope concrete or metal structural component will become accessible through excavation, a followup action is initiated to the responsible engineer to inspect the exposed surfaces for age-related degradation. Such inspections will include concrete examination using acceptance criteria from GALL Report AMP XI.S6, Program Element 6. Degradation found that exceeds the acceptance criteria will be trended and processed through the FENOC Corrective Action Program. • List ACI 349.3R, ANSI/ASCE 11-90, and EPRI Report 1007933 as references and indicate that they provide guidance for detecting aging effects. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Add sufficient acceptance criteria and critical parameters to trigger an increased level of inspection and initiation of corrective action. Indicate that ACI 349.3R provides acceptable guidelines that will be considered in developing acceptance criteria for concrete structural elements, steel liners, joints, and waterproofing membranes. The acceptance criteria for visual inspection of coatings on in-scope concrete structures will be in accordance with ACI 349.3R. Plant-specific quantitative degradation limits, similar to the three-tier hierarchy acceptance criteria from Chapter 5	and RAI A.1-1 from NRC Letter dated March 26, 2013		

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	of ACI 349.3R, will be developed and added to the inspection procedure. The Structures Monitoring Program procedure will also be enhanced to reflect the "Periodic Evaluation" criteria defined in Chapter 3.3 of ACI 349.3R. The Structures Monitoring Program procedure will include the "prioritization process" to develop a representative sample of areas to inspect in accordance with ACI 349.3R. • Require that personnel performing the structural inspections meet qualifications that are commensurate with ACI 349.3R, Chapter 7, "Qualifications of Evaluation Team." • The program procedure will be enhanced by specifying that, for the structures within the scope of license renewal, inspections will be conducted at least once every 5 years. • Conduct a baseline inspection of the structures within the scope of license renewal prior to entering the period of extended operation. • Require optical aids, scaling technologies, mechanical lifts, ladders or scaffolding for tall structures or difficult-to-reach areas of structures, to allow visual inspections that meet the guidelines of Chapter 5 of ACI 349.3R. Select the areas to be inspected in accordance with the guidelines of Chapter 5 of ACI 349.3R to reflect the "Periodic Evaluation" criteria defined in Chapter 3.3 of ACI 349.3R. Include the "prioritization process" in the selection methodology to develop a representative sample of areas to inspect in accordance with ACI 349.3R. • Monitor elastomeric vibration isolators and structural sealants for cracking, loss of material, and hardening. • Supplement visual inspection of elastomeric vibration isolation elements by feel to detect hardening if the vibration isolation function is suspect.			

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	• Identify that: ○ loose bolts and nuts and cracked high-strength bolts are not acceptable unless accepted by engineering evaluation ○ structural sealants that are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing ○ elastomeric vibration isolation elements that are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function. • Require that high-strength (i.e., American Society for Testing and Materials (ASTM) A540 Grade B23) structural bolting materials with an actual measured yield strength greater than or equal to 150 ksi and greater than 1 inch in nominal diameter are monitored for SCC. Perform periodic visual inspections of susceptible ASTM A540 bolting to identify locations where ASTM A540 bolting may be exposed to a potentially corrosive environment for SCC. Complete the initial visual inspections prior to entering the period of extended operation, and perform recurring inspections at an interval not to exceed 5 years. Perform volumetric examination (i.e., ultrasonic testing (UT)) on a sampling basis of bolting exposed to a corrosive environment, as determined by engineering evaluation, to a depth of at least 12 inches. • Require that personnel performing UT examinations of structural bolting have a current ASME Code Section XI, Appendix VIII, Supplement 8 endorsement. • Revise the applicable structural bolting specifications to prevent future use of ASTM A540 bolting with measured yield strength equal to or exceeding 150 ksi.			

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21.	Enhance the Water Control Structures Inspection to: • Include the Service Water Discharge Structure that is within the scope of license renewal. • Include parameters monitored and inspected for water control structures, including the Service Water Discharge Structure, in accordance with applicable inspection elements listed in Section C.2 of Regulatory Guide 1.127 Revision 1. Descriptions of concrete conditions will conform to the appendix to the publication ACI 201. The use of photographs for comparison of previous and present conditions will be included as part of the inspection program. • Specify that water control structure periodic inspections are to be performed at least once every 5 years. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Add sufficient acceptance criteria and critical parameters to trigger an increased level of inspection and initiation of corrective action. Indicate that ACI 349.3R provides acceptable guidelines that will be considered in developing acceptance criteria for water control structures. Plant-specific quantitative degradation limits, similar to the three-tier hierarchy acceptance criteria from Chapter 5 of ACI 349.3R, will be developed and added to the inspection procedure. The Structures Monitoring Program procedure will also be enhanced to reflect the "Periodic Evaluation" criteria defined in Chapter 3.3 of ACI 349.3R. The Structures Monitoring Program procedure will include the "prioritization process" to develop a representative sample of areas to inspect in accordance with ACI 349.3R.	A.1.40 B.2.40 Responses to NRC RAIs B.2.39-6 from NRC Letter dated April 5, 2011; Supplemental RAI OIN-379 from Region III IP-71002 Inspection; and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-11-292 and L-13-160

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	• Conduct a baseline inspection of the structures within the scope of license renewal prior to entering the period of extended operation. • Require that loose bolts and nuts, cracked high-strength bolts, and degradation of piles and sheeting (sheet pilings) are accepted by engineering evaluation or subject to corrective actions. Engineering evaluation will be documented and based on codes, specifications, and standards such as American Institute of Steel Construction specifications, Structural Engineering Institute (SEI)/ASCE) 11, and codes, specifications, or standards referenced in the Davis-Besse current licensing basis.			
22.	Enclose or otherwise protect the safety-related station ventilation radiation monitors located in the Turbine Building such that leakage and spray from surrounding piping systems does not adversely affect the intended function of the radiation monitors.	Response to NRC RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	FENOC Letter L-13-160
23.	In association with the time-limited aging analysis for effects of environmentally assisted fatigue of the high-pressure injection (HPI) nozzle safe end including the associated Alloy 82/182 weld (weld that connects the safe end to the nozzle), replace the HPI nozzle safe end, including the associated Alloy 82/182 weld, for all four HPI nozzles prior to the period of extended operation. Apply the Fatigue Monitoring Program to evaluate the environmental effects and manage cumulative fatigue damage for the replacement HPI nozzle safe ends and associated welds.	A.2.3.4.2 A.2.7.4 Responses to NRC RAIs 4.7.4 1 from NRC Letter dated April 15, 2011; RAI 4.3-18 from NRC Letter dated June 17, 2011; RAI 4.7.4-1 from NRC Letter dated October 11, 2011;	Prior to October 22, 2016	LRA and FENOC Letters L-11-107, L-11-203, L-11-334, and L-13-160

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		RAI A.1-1 from NRC Letter dated March 26, 2013;		
24.	Apply the elements of corrective actions, confirmation process, and administrative controls in the Quality Assurance Program Manual to the credited aging management programs and activities for safety-related and nonsafety-related structures and components determined to require aging management for the period of extended operation.	A.1 Response to NRC RAI 3.0 from NRC letter dated May 2, 2011, and RAI A.1-1 from NRC letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-166 and L-13-160
25.	Not used.			
26.	Obtain and evaluate for degradation a concrete core bore from two representative inaccessible concrete components of an in-scope structure subjected to aggressive groundwater prior to entering the period of extended operation. Based on the results of the initial core bore sample, evaluate the need for collection and evaluation of representative concrete core bore samples at additional locations that may be identified during the period of extended operation as having aggressive groundwater infiltration. Select additional core bore sample locations based on the duration of observed aggressive groundwater infiltration. Document identified concrete or steel degradation in the FENOC Corrective Action Program.	Responses to NRC RAI B.2.39-3 from NRC Letter dated April 5, 2011; RAI B.2.39-11 from NRC Letter dated July 21, 2011; and Supplemental RAI B.2.39-11 from telecon held with the NRC on September 13, 2011	COMPLETE	FENOC Letters L-11-153, L-11-237, L-11-292, and L-15-120

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27.	DBNPS Surveillance Test Procedure DB-PF-03009, Revision 06, "Containment Vessel and Shielding Building Visual Inspection," Subsection 2.1.2, shall be enhanced to state, "Personnel who perform general visual examinations of the exterior surface of the Containment Vessel and the interior and exterior surfaces of the shield building shall meet the requirements for a general visual examiner in accordance with Nuclear Operating Procedure NOP-CC-5708, 'Written Practice for the Qualification and Certification of Nondestructive Examination Personnel.' These individuals shall be knowledgeable of the types of conditions which may be expected to be identified during the examinations."	Responses to NRC RAIs B.2.1-1 from NRC Letter dated April 5, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	FENOC Letters L-11-134 and L-13-160
28.	Enhance the Fuel Oil Chemistry Program to: • Require that internal surfaces of EDG fuel oil storage tanks and day tanks, diesel oil storage tank, diesel fire pump day tank, and station blackout diesel generator day tank are periodically drained (at least once every 10 years) for cleaning and are visually inspected to detect potential degradation. If degradation is identified in a diesel fuel tank by visual inspections, a volumetric inspection is performed. • Require that biological activity be monitored and trended at least quarterly.	A.1.20 B.2.20 Responses to NRC RAIs B.2.20-1 and B.2.20-2 from NRC Letter dated April 5, 2011; Supplemental RAI OIN-368 from NRC Region III IP-71002 Inspection; and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-134, L-11-238, and L-13-160

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29.	Enhance the Cranes and Hoists Inspection Program to: • Include visual inspections for loose bolts and missing or loose nuts in crane, monorail, and hoist inspection procedures at the same frequency as inspections of rails and structural components.	A.1.10 B.2.10 Responses to NRC RAI B.2.10-2 from NRC Letter dated April 20, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153 and L-13-160
30.	Enhance the Leak Chase Monitoring Program to: • Include acceptance criteria such that measurement of leakage from any monitoring line exceeding 15 ml/min will be documented in the Corrective Action Program for evaluation and potential corrective actions. Evaluation will include consideration of more frequent monitoring. • Analyze collected leak chase drainage for pH monthly and for iron every 6 months. The initial acceptance criteria will be 7.0 to 8.0 for pH. The results for iron will be monitored and trended to ensure that there is no indication of corrosion of the reinforcing bars in the walls or floor of the pool and pits. An acceptance criterion for the iron analyses will be developed after 3 years of measurements. Analyses that exceed the limits will be documented in the Corrective Action Program. • Perform the leak chase inspection and cleaning recurring PM activity every 18 months. • Inspect once per year for leakage migrating through the accessible outside walls and floor (from the ceiling side) of the pool and pits. Document the inspection results and retain in plant records. Indication of leakage through the walls will be documented in the Corrective Action Program.	A.1.25 B.2.25 Responses to NRC RAI B.2.25-5 from NRC Letter dated April 5, 2011; RAIs B.2.25-7 and B.2.39-10 from NRC Letter dated July 21, 2011; and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-11-238, and L-13-160

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31.	Incorporate reference to and the preventative actions of the Research Council for Structural Connections "Specification for Structural Joints Using ASTM A325 or A490 Bolts" into the Davis-Besse specifications and implementing procedures that address Davis-Besse structural bolting within the scope of license renewal.	Responses to NRC RAIs B.2.39-8 from NRC Letter dated April 5, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	FENOC Letters L-11-153 and L-13-160
32.	Enhance the Closed Cooling Water Chemistry program to: • Document the results of periodic inspections of opportunity, performed when components are opened for maintenance, repair, or surveillance. • Ensure that a representative sample of piping and components will be inspected on a 10-year interval, with the first inspection taking place prior to entering the period of extended operation. • Ensure that component cooling water radiochemistry is sampled on a weekly interval to verify the integrity of the letdown coolers and seal return coolers.	A.1.8 B.2.8 Responses to NRC RAI B.2.8-1 from NRC Letter dated April 20, 2011; Supplemental RAI 2.3.3.18-4 from telecon held with the NRC on November 9, 2011; and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153, L-11-354, and L-13-160
33.	<u>Refueling Canal Leakage Mitigation</u> <u>Phase 1</u> Perform the following actions to reduce or mitigate the refueling canal leaks inside containment: 1. Select and implement a leak detection method to locate the leakage area. 2. Evaluate temporary and permanent repair methods to stop or significantly reduce the leakage, and implement a repair plan.	Responses to NRC RAIs B.2.39-9 from NRC Letter dated July 27, 2011; RAI A.1-1 from NRC Letter dated March 26, 2013; and 2014 Annual Update	Phase 1: Action 1 COMPLETE Action 2 COMPLETE	FENOC Letters L-11-252, L-13-160, and L-14-206

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	Phase 2 Perform the following actions to evaluate the impact of refueling canal leaks on concrete and reinforcing steel structures. Discontinue core bores, testing, and reinforcing steel inspections when indications of refueling canal leakage are no longer present: 1. Perform a core bore in the south wall of the east-west section of the core flood pipe tunnel. a. Assess borated water degradation of the concrete by testing the core bore sample for compressive strength and by petrographic examination, and evaluate the results. a. Conduct a visual examination of the concrete and reinforcing steel to identify aging effects (e.g., concrete degradation or steel corrosion). Enter identified aging effects into the FENOC Corrective Action Program and evaluate in accordance with the requirements of the current licensing basis Maintenance Rule Program. 2. If leakage from the refueling canal has not been eliminated or resumes by the beginning of the period of extended operation, then evaluate the concrete structures in a manner similar to the way that they were evaluated under Phase 2, Action 1. However, use acceptance criteria from ACI Report 349.3R for the evaluation. 3. If leakage from the refueling canal has not been eliminated or resumes during the period of extended operation, then evaluate the concrete structures again in a manner similar to the way that they were evaluated under Phase 2, Action 2. Perform evaluations every 10 years until the end of the period of extended operation.		Phase 2: Action 1 COMPLETE Action 2 prior to December 31, 2023 Action 3 – Ongoing	

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34.	Enhance the Bolting Integrity Program to: • Select an alternative stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant will be included in the program.	A.1.4 B.2.4 Responses to NRC RAIs B.2.4-3 from NRC Letter dated April 20, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153 and L-13-160
35.	Perform the following actions for each of two examinations (Phase 1 and Phase 2) of the Containment Vessel in the sand pocket region: • Perform nondestructive examination of the Containment Vessel from the outer surface at five areas of previously-identified groundwater in-leakage. ○ Examine the vessel at a minimum of three vertical grid locations at 12 inches nominal horizontal spacing at each area. Examine the Containment Vessel at a minimum of three elevations: a. approximately 3 inches below the existing grout-to-vessel interface in the sand pocket region b. at the existing grout-to-vessel interface level in the sand pocket region c. approximately 3 inches above the existing grout-to-vessel interface in the sand pocket region • Compare the UT thickness readings to minimum ASME Code vessel thickness requirements and to the results obtained during previous UT examinations of the Containment Vessel. Determine the need for maintenance or repair of the Containment Vessel based on the results and evaluation of the examinations.	Response to NRC RAI B.2.22-5 from NRC Letter dated July 21, 2011, and 2014 Annual Update	Phase 1 COMPLETE and Phase 2 prior to December 31, 2025	FENOC Letter L-11-252 and L-14-206

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	• Document the results of each of the two examinations in the work order system. Document and evaluate adverse conditions in accordance with the FENOC Corrective Action Program for an evaluation of potential degradation of the steel Containment Vessel thickness over the longer term.			
36.	Perform the following actions related to the Containment Vessel sand pocket region each refueling outage: • Perform visual inspection of 100 percent of the accessible areas of the wetted outer surface of the Containment Vessel in the sand pocket region. • Perform visual inspection of accessible dry areas of the outer surface of the Containment Vessel in the sand pocket region and the areas above the grout-to-steel interface up to Elevation 566 feet + 3 inches, - 1 inch. • Perform visual inspection for deterioration (e.g., missing or damaged grout) of accessible grout and the containment exterior moisture barrier in the sand pocket area. • Perform opportunistic visual inspections of inaccessible areas of the Containment Vessel in the sand pocket region when such areas are made accessible. • Perform opportunistic visual inspections for deterioration (e.g., missing or damaged grout) of inaccessible grout in the sand pocket region when such areas are made accessible. Inaccessible grout is the grout below the normally exposed surface of the grout in the sand pocket area. • Address issues of pitting or MIC, and degraded grout, moisture barrier or sealant identified during the inspections using the FENOC Corrective Action Program. • Sample the water in the sand pocket region when sufficient volumes are available. The number of sampled	Responses to NRC RAI B.2.22-5 from NRC Letter dated July 21, 2011, and Supplemental RAI B.2.22-5 from telecons held with the NRC on October 5 and November 14, 2011	Ongoing	FENOC Letters L-11-252 and L-11-354

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	water volumes will be determined by the number of water volumes observed and the size of those water volumes. Analyze the sample(s) for pH, chlorides, iron, and sulfates. Treat or wash (or a combination thereof) the sand pocket area to reduce measured chloride concentrations to less than 250 parts per million (ppm) if the concentration of chlorides in a sample exceeds 250 ppm. Note: Water samples may be taken at different times during each outage. Engineering judgment may be used to determine the priority of the chemical analyses to be performed if sufficient water is not available in a given sample for all analyses.			
37.	Perform and evaluate core bores of the emergency core cooling system Pump Room No. 1 wall and the Room 109 ceiling. • The core bores will be deep enough to expose reinforcing bar in the wall and ceiling. The core samples from the core bores will be examined for signs of corrosion or chemical effects of boric acid on the concrete or reinforcing bars. The examination will include a petrographic examination. The reinforcing steel that will be exposed for a visual inspection will have corrosion products collected for testing. Degradation identified from the samples will be entered into the FENOC Corrective Action Program. The core bores will be performed in areas where leakage has been observed in the past. • The first set of core bores will be performed prior to the end of 2014 (Phase 1). • The second set of core bores will be performed prior to the end of 2020 (Phase 2). • Further core bores will be conducted, if warranted, based on the evaluation of the results of the inspection and testing of the core bores or if spent fuel pool leakage through the wall or ceiling recurs after the	Responses to NRC RAI B.2.39-2 from NRC Letter dated April 5, 2011, and RAI B.2.39-10 from NRC Letter dated July 21, 2011	Phase 1 COMPLETE and Phase 2 prior to December 31, 2020	FENOC Letters L-11-153, L-11-238, and L-15-120

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	second set of core bores is performed. If spent fuel pool leakage through another wall or ceiling is identified, then core bores will be performed in a manner similar to that stated for the emergency core cooling system Pump Room No. 1 wall and the Room 109 ceiling.			
38.	Evaluate the concrete cracking observed on the underside of the spent fuel pool for necessary repairs. Note: A core bore of the Room 109 ceiling will be performed by the end of 2014 (see license renewal Commitment No. 37). Degradation identified from the samples will be entered into the FENOC Corrective Action Program. The condition of the concrete and the reinforcing steel will be evaluated at that time to assist in determining what repairs, if any, need to be made to the underside of the spent fuel pool concrete. The criterion for determining the need to repair the cracking will be the continued capability of the structures to perform their intended functions during the period of extended operation.	Responses to NRC RAI B.2.39-2 from NRC Letter dated April 5, 2011; RAI B.2.39-10 from NRC Letter dated July 21, 2011; and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	FENOC Letters L-11-153, L-11-238, and L-13-160
39.	Address the potential for borated water degradation of the steel containment vessel through the following actions: • Access the inside surface of the embedded steel containment at a vertical height no greater than 10 inches above bottom dead center. A core bore will be completed by the end of 2014 (Phase 1). If necessary, a second core bore will be completed by the end of 2020 (Phase 2). If there is evidence of the presence of borated water in contact with the steel containment vessel, conduct nondestructive testing to determine what effect, if any, the borated water has had on the steel containment vessel. Based on the results of the nondestructive testing, perform a study to determine the effect through the period of extended operation of any identified loss of thickness in the steel containment due to exposure to borated water.	Responses to NRC RAI B.2.22-2 from NRC Letter dated April 5, 2011; RAI B.2.22-6 from NRC Letter dated July 27, 2011; Supplemental RAI B.2.22-6 from NRC telecon held on May 9, 2013; and 2014 Annual Update	Phase 1 COMPLETE Phase 2 prior to December 31, 2020	FENOC Letters L-11-153, L-11-237, L-13-180, and L-14-206

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
40.	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Program as described in LRA Section B.2.41.	A.1.41 B.2.41 Responses to NRC RAIs 3.3.2.2.5-1 and 3.3.2.71-2 from NRC Letter dated April 20, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-153 and L-13-160
41.	Establish a PM task to periodically replace the flexible connections exposed to fuel oil in the Fuel Oil System.	Responses to NRC RAI 3.3.2.3.12-2 from NRC Letter dated May 2, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	FENOC Letters L-11-166 and L-13-160
42.	Enhance the Fatigue Monitoring Program to: • Evaluate additional plant-specific component locations in the reactor coolant pressure boundary that may be more limiting than those considered in NUREG/CR-6260. This evaluation will include identification of the most limiting fatigue location exposed to reactor coolant for each material type (i.e., CS, LAS, SS, and NBA) and that each bounding material/location will be evaluated for the effects of the reactor coolant environment on fatigue usage. Nickel-based alloy items will be evaluated using NUREG/CR-6909. Submit the evaluation to the NRC 1 year prior to the period of extended operation.	A.1.16 B.2.16 Response to NRC RAI B.2.16-2 from NRC Letter dated April 20, 2011	Prior to April 22, 2016	LRA and FENOC Letter L-11-166

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
43.	Ensure that the current station operating experience review process includes future reviews of plant-specific and industry operating experience to confirm the effectiveness of the License Renewal aging management programs, to determine the need for programs to be enhanced, or indicate a need to develop new aging management programs.	Responses to NRC RAI B.1.4-1 from NRC Letter dated May 19, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	COMPLETE	FENOC Letters L-11-188, L-13-160, and L-13-257
44.	Cathodically protect the EDG fuel oil storage tanks (DB-T153-1 and DB-T153-2) and the in-scope fuel oil and Service Water buried piping in accordance with NACE SP0169-2007 or NACE RP0285-2002.	Responses to NRC RAIs B.2.7-1 from NRC Letter dated April 20, 2011, as modified per telecon with the NRC held on June 7, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	COMPLETE	FENOC Letters L-11-203, L-11-218, L-13-160, and L-14-114
45.	Implement the Nuclear Safety-Related Coatings Program as described in LRA Section B.2.42.	A.1.42 B.2.42 Responses to NRC RAIs XI.S8-1 from NRC Letter dated April 5, 2011, and A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-203, L-11-218, and L-13-160

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
46.	Implement the Shield Building Monitoring Program as described in LRA Section B.2.43.	A.1.43 B.2.43 Responses to NRC RAIs B.2.16-2 from NRC Letter dated December 27, 2012, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-12-028 and L-13-160
47.	Enhance the Inservice Inspection (ISI) Program - IWE to: • Include surface examinations to monitor for cracking of containment 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. The inspection sample size will include 10 percent of the containment penetration population that is subject to cyclic loading but has no current licensing basis fatigue analysis. Penetrations included in the inspection sample will be scheduled for examination in each 10-year ISI interval that occurs during the period of extended operation. Should fatigue analyses be performed in the future for the subject containment penetrations, the surface examinations will no longer be required.	A.1.22 B.2.22 Responses to NRC RAI B.2.22-7 from NRC Letter dated July 21, 2011; Supplemental RAI B.2.22-7 from NRC telecons held on September 13 and 16, 2011; and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-238, L-11-292 and L-13-160

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
48.	Complete an investigation and needed repairs or modification of the degraded portion of the safety-related intake canal embankment.	Responses to NRC RAI B.2.40-2 from NRC Letter dated July 21, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Complete	FENOC Letters L-11-238, L-13-160, and L-15-214
49.	Enhance the Nickel-Alloy Management Program to: • Provide for inspection of dissimilar metal butt welds in accordance with the requirements of ASME Code Case N-770-1, "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, Section XI, Division 1," as modified by 10 CFR 50.55a(g)(6)(ii)(F).	A.1.28 B.2.28 Responses to NRC RAI B.2.28-1 from NRC Letter dated July 27, 2011, and RAI A.1-1 from NRC Letter dated March 26, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-11-238 and L-13-160
50.	Enhance the ISI Program – IWF to: • Include monitoring of ASTM A490 high-strength bolting (i.e., actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1 inch nominal diameter for cracking using volumetric examination. The volumetric examinations will be performed in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section V, Article 5, Appendix IV, 2007 Edition through 2008 Addenda. The representative sample size will be equal to 20 percent (rounded up to the nearest whole number) of the entire IWF population of ASTM A490 high-strength bolts in sizes greater than 1 inch nominal diameter, with a maximum sample size of 25 bolts. The selection of the representative sample will consider	A.1.23 B.2.23 Supplemental response to NRC RAI B.2.4-1b from NRC Letter dated February 14, 2013, and from telephone conference calls held on April 11, April 24, May 2, and May 28, 2013	Prior to October 22, 2016	LRA and FENOC Letters L-13-181 and L-13-199

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
	susceptibility to SCC (e.g., actual measured yield strength) and as low as reasonably achievable (ALARA) radiation dose reduction principles. The frequency of examination will be once each 10-year ISI interval, beginning with the fourth interval that started September 21, 2012. • Include monitoring of ASTM A540 high-strength bolting (i.e., actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1 inch in nominal diameter for cracking. Periodic visual inspections of susceptible ASTM A540 bolting will be conducted prior to the period of extended operation and at an interval not to exceed 5 years to identify locations where the A540 bolting may be exposed to a potentially corrosive environment for SCC. If the visual inspections identify one or more bolts in a potentially corrosive environment, then an engineering evaluation will be performed to determine whether the bolting material had been subjected to a corrosive environment for SCC. The bolts determined to have been subjected to a corrosive environment for SCC comprise the population subject to sampling for volumetric examinations. The representative sample size is equal to 20 percent (rounded up to the nearest whole number) of the bolts in the sample population, with a maximum sample size of 25 bolts. The volumetric examinations are performed in accordance with the requirements of ASME Code Section V, Article 5, Appendix IV. Volumetric examinations will be performed no later than the subsequent refueling outage following visual identification of bolting subject to a corrosive environment. Deferral of volumetric examinations to the subsequent refueling outage is not permitted if the visual inspection indicates evidence of contaminant penetration through the coatings. The frequency of examination is once each 10-year ISI interval, beginning with the 4th interval that started September 21, 2012. For ASTM A540 high-strength bolts that are not exposed to a corrosive environment, the volumetric			

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	examinations are waived based on plant-specific operating experience associated with the volumetric examination of the Davis-Besse reactor head closure studs (60 each) constructed of high-strength ASTM A540 material, where the studs are examined once each ISI interval, and after three intervals, no unacceptable indications have been noted. • As an alternative to the visual examinations and the subsequent volumetric examinations of ASTM A540 bolts subjected to a corrosive environment, the ISI Program – IWF provides an option to perform periodic volumetric examinations as follows. The program includes monitoring of ASTM A540 high-strength bolting (i.e., actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1 inch nominal diameter for cracking using volumetric examination. The volumetric examinations are performed in accordance with the requirements of ASME Code Section V, Article 5, Appendix IV. The representative sample size is equal to 20 percent (rounded up to the nearest whole number) of the entire IWF population of ASTM A540 high-strength bolts in sizes greater than 1 inch nominal diameter, with a maximum sample size of 25 bolts. The selection of the representative sample considers susceptibility to SCC (e.g., actual measured yield strength) and ALARA radiation dose reduction principles. The frequency of examination is once each 10-year ISI interval, beginning with the 4th interval that started September 21, 2012.			
51.	Implement the Service Level III Coatings and Linings Monitoring Program.	A.1.44 B.2.44 Response to NRC RAI 3.0.3-3 from NRC Letter dated November 26, 2013	Prior to October 22, 2016	LRA and FENOC Letter L-14-061

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
52.	In response to MRP-227-A Applicant/Licensee Action Item 6, submit for NRC review and approval an evaluation justifying the acceptability of inaccessible and non-inspectable component items (core barrel cylinder including vertical and circumferential seam welds, former plates, external baffle-to-baffle bolts and their locking devices, core barrel-to-former bolts and their locking devices, and internal baffle-to-baffle bolts) for continued operation through the period of extended operation and, if necessary, provide a plan for replacement of the components.	A.1.32 B.2.32	Within 1 year of the detection of degradation exceeding the acceptance criteria of the linked MRP-227-A primary component items leading to expansion	LRA and FENOC Letters L-15-139 and L-15-166
53.	In response to MRP-227-A Applicant/Licensee Action Item 7, develop and submit for NRC review and approval a plant-specific analysis to demonstrate that the Incore Monitoring Instrumentation (IMI) guide tube assembly spiders, Control Rod Guide Tube (CRGT) spacer castings, and additional RV Internals component items that may be fabricated from CASS, martensitic stainless steel, or martensitic precipitation-hardened stainless steel materials (e.g., Core Support Shield (CSS) vent valve top and bottom retaining rings) will maintain their functionality during the period of extended operation. The analysis will consider the possible loss of fracture toughness in these component items due to thermal embrittlement and/or irradiation embrittlement and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The Davis-Besse analysis will be consistent with the licensing basis and the need to maintain the functionality of the component items being evaluated under all licensing basis conditions of operation.	A.1.32 B.2.32	One year prior to the MRP-227-A inspection of the applicable component items	LRA and FENOC Letters L-15-139 and L-15-166

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Item Number	**Commitment**	**Updated Safety Analysis Report (USAR) Supplement Section No/ Comments**	**Implementation Schedule**	**Source**
54.	In response to MRP-227-A Applicant/Licensee Action Item 8, update and submit for NRC review and approval an evaluation for the period of extended operation regarding the effect of irradiation on the mechanical properties and deformation limits of the RV internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-10008, Part 1, Revision 1, supplemented by Davis-Besse USAR Appendix 4A.	A.1.32 B.2.32	Prior to October 22, 2016	LRA and FENOC Letters L-15-139 and L-15-166
55.	Perform the following actions to improve and maintain the fidelity of the data in the Flow-Accelerated Corrosion Program: • Perform a review of the CHECWORKS SFA model to determine which inputs are critical to the determination of fitness for service and which inputs are noncritical. This action will document the listing of all input fields within the software, and whether their accuracy affects the output of the model. • Perform a validation of the data inputs into CHECWORKS SFA. This task will include the validation of any input which would have consequence, as used by the CHECWORKS SFA software in the determination of fitness for service of piping and components for the Flow-Accelerated Corrosion Program. Data contained within the CHECWORKS SFA model that does not affect fitness for service will be annotated during this validation as being noncritical to the function of the software, while still attempting to validate it. • Document the results of the validation of the CHECWORKS SFA database. This action will create a document (e.g., Reference Material, Program Manual) that will serve as a listing of inputs into the CHECWORKS SFA database and be maintained as a quality record. • Revise the CHECWORKS SFA model to correct the restriction orifices' size/dimension for the orifice and flow elements identified in the Steam Line Failure Root Cause Evaluation.	A.1.19 B.2.19	Prior to October 22, 2016	LRA and FENOC Letter L-15-192

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	• Establish a list of components for the site that meet the bulleted items within Section 4.4.4 of NSAC-202L, Revision 4. Compile the inspection history of the relevant components. Perform an evaluation for any components without inspection data and add components requiring inspection to 19RFO scope. These locations are to specifically include: ○ locations downstream of orifices, flow elements, venturis, thermowells, angle valves, flow control valves, or level control valves ○ locations or lines known to contain backing rings or counterbore ○ field-fabricated tees and laterals ○ nozzles ○ complex geometric locations such as components located within two diameters of each other (e.g., an elbow welded to a tee) ○ components downstream of replaced components (upstream if expander), and components that have been replaced in the past if not upgraded to resistant material ○ components (including straight pipe) immediately downstream of flow-accelerated corrosion-resistant components (e.g., containing chromium greater than 0.10%) ○ locations immediately downstream of turning vanes ○ expansion joints • Revise the Flow-Accelerated Corrosion Program procedure as follows:			

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	○ Add requirements to the procedure that would involve review and selection of examination scope based on recommendations from NSAC-202L, Revision 4, Section 4.4.4. This action requires documentation of the basis for selection or exclusion of the scope for the given outage. Documentation would be in the form of discussion in the Outage Technical Report (pre-outage) and Outage Summary Report (post-outage). ○ Add a step that would require review, approval, and documentation of updates to the CHECWORKS SFA database. The scope of these changes would exclude data collected and evaluated during outages but would be inclusive of all others (e.g., plant uprates, plant modifications, engineering change packages). Documentation for this step would be through an Engineering Evaluation Request.			

APPENDIX B

CHRONOLOGY

This appendix contains a chronological listing of the routine correspondence between the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) and FirstEnergy Nuclear Operating Company (FENOC or the applicant) and other correspondence regarding the staff's reviews of the Davis-Besse Nuclear Power Station (Davis-Besse), Docket Number 50-346, license renewal application (LRA).

Table B-1. Chronology

Date	Subject
September 20, 2013	Letter from Lieb R. A., FENOC: License Renewal Application Amendment No. 46 - Annual Update (TAC No. ME4640). (ADAMS Accession No. ML13269A027)
October 18, 2013	Letter from Lieb R. A., FENOC: Review of the Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML13294A463)
November 26, 2013	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML13294A500)
December 6, 2013	Summary of Telephone Conference Calls Held on November 19 and 22, 2013, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Draft Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application (TAC No. ME4640). (ADAMS Accession No. ML13330B026)
January 31, 2014	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 47. (ADAMS Accession No. ML14035A164)
February 11, 2014	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML14029A201)
February 19, 2014	Letter from Lieb R. A., FENOC: Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) Amendment No. 48. (ADAMS Accession No. ML14055A067)
March 4, 2014	Summary of Telephone Conference Call Held on February 5, 2014, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, Concerning Draft Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application (TAC No. ME4640). (ADAMS Accession No. ML14056A152)
March 11, 2014	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 49. (ADAMS Accession No. ML14072A008)
April 15, 2014	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML14097A454)

Date	Subject
April 22, 2014	"Summary of Telephone Conference Held on March 27, 2014, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company Concerning Commitment No. 13 of the Safety Evaluation Report Pertaining to the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application." (TAC No. ME4640). (ADAMS Accession No. ML14099A339)
June 23, 2014	Letter from Lieb R. A., FENOC: License Renewal Application Amendment No. 50 - Annual Update (TAC Nos. ME4640 and ME4613). (ADAMS Accession No. ML14175B381)
July 3, 2014	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 51. (ADAMS Accession No. ML14184B184)
July 7, 2014	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML14178A962)
July 29, 2014	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 52. (ADAMS Accession No. ML14211A023)
August 19, 2014	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML14218A145)
September 16, 2014	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 53. (ADAMS Accession No. ML14259A067)
September 29, 2014	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML14258A285)
October 28, 2014	Letter from Summers T. J., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640). (ADAMS Accession No. ML14301A369)
December 30, 2014	Letter to Lieb R. A., FENOC: Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station (TAC No. ME4640). (ADAMS Accession No. ML14353A425)
January 28, 2015	Letter from Lieb R. A., FENOC: Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 54. (ADAMS Accession No. ML15028A540)
April 8, 2015	Letter from Lieb R. A., FENOC: Notification of Completion of License Renewal Commitments Related to the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 55. (ADAMS Accession No. ML15098A443)
April 20, 2015	Summary of Telephone Conference Call Held on February 20, 2015, Between the U.S. Nuclear Regulatory Commission and FirstEnergy Nuclear Operating Company, To Clarify the Responses to Requests for Additional Information Pertaining to the Davis-Besse Nuclear Power Station, Unit 1, License Renewal Application (TAC No. ME4640). (ADAMS Accession No. ML15056A456)
April 21, 2015	Letter from Lieb R. A., FENOC: Davis-Besse Nuclear Power Station Unit No. 1, License Renewal Reactor Vessel Internals Inspection Plan (TAC No. ME4640). (ADAMS Accession No. ML15113B132)

Date	Subject
April 21, 2015	Letter from Lieb R. A., FENOC: Davis-Besse Nuclear Power Station Unit No. 1, License Renewal Reactor Vessel Internals Inspection Plan, non-proprietary AREVA Report No. ANP-3920, Revision 1, "Reactor Vessel Internals Inspection Plan for Davis Besse Nuclear Plant Unit No. 1 – Licensing Report (TAC No. ME4640). (ADAMS Accession No. ML15113B133)
April 21, 2015	Letter from Lieb R. A., FENOC: Davis-Besse Nuclear Power Station Unit No. 1, License Renewal Reactor Vessel Internals Inspection Plan, Non-proprietary AREVA NP Licensing Report No. ANP-3285, Revision 0, "Confirmation of Stress Relief for the DB-1 [Davis-Besse] Core Support Structure Upper Flange Weld (TAC No. ME4640). (ADAMS Accession No. ML15113B134)
May 20, 2015	Letter from Lieb R. A., FENOC: Supplemental Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, Reactor Vessel Internals Inspection Plan (TAC No. ME4640) and License Renewal Application Amendment No. 56. (ADAMS Accession No. ML15140A705)
May 28, 2015	Letter to Lieb R. A., FENOC: Davis-Besse Nulear Power Station - Inspection of Apparent Cause Evaluation Efforts for Propagation of Laminar Cracking in Reinforced Concrete Shield Building and Closure of Unresolved Item Involving Shield Building Laminar Cracking Licensing Basis. Inspection Report R 05000346/2014008. (ADAMS Accession No. ML15148A489)
June 5, 2015	Letter from Lieb R. A., FENOC: Supplemental Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, Reactor Vessel Internals Inspection Plan (TAC No. ME4640) and License Renewal Application Amendment No. 57. (ADAMS Accession No. ML15156B144)
June 12, 2015	Letter from Lieb R. A., FENOC: Supplemental Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 58. (ADAMS Accession No. ML15163A195)
June 29, 2015	Letter from Boles B.D., FENOC: License Renewal Application Amendment No. 59 - Annual Update (TAC No. ME4640). (ADAMS Accession No. ML15180A252)
July 13, 2015	Request For Withholding Information From Public Disclosure (TAC No. ME4640) (ADAMS Accession No. ML15189A056)
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APPENDIX C

PRINCIPAL CONTRIBUTORS

This appendix lists the principal contributors for the development of this supplemental safety evaluation report and their areas of responsibility.

APPENDIX C: PRINCIPAL CONTRIBUTORS	
Name	**Responsibility**
O. Aloysious	Reviewer-Reactor Systems
S. Cuadrado DeJesus	Reviewer-Structural
Y. Diaz-Sanabria	Management Oversight
A. Foli	Reviewer-Electrical
B. Fu	Reviewer-Reactor Systems
J. Gavula	Reviewer-Mechanical
D. Hoang	Reviewer-Electrical
W. Holston	Reviewer-Mechanical
R. Kalikian	Reviewer-Mechanical
G. Kulesa	Management Oversight
J. Lubinski	Management Oversight
T. Lupold	Management Oversight
J. Marshall	Management Oversight
M. Marshall	Management Oversight
J. Medoff	Reviewer-Reactor Systems
C. Miller	Management Oversight
S. Min	Reviewer-Reactor Systems
D. Morey	Management Oversight
R. Plasse	Project Manager
G. Thomas	Reviewer-Structural
J. Uribe	Reviewer-Mechanical
J. Wise	Reviewer-Mechanical
M. Yoo	Reviewer-Reactor Systems

APPENDIX D

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This appendix lists the references used throughout this supplemental safety evaluation report for review of the license renewal application (LRA) for Davis-Besse Nuclear Power Station.

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NRC FORM 335
(12-2010)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-2193, Supplement 1

2. TITLE AND SUBTITLE
Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station, Docket Number 50-346, FirstEnergy Nuclear Operating Company, Supplement

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	2016

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
See Appendix C

6. TYPE OF REPORT
Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)
Division of License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)
same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)
This document is a supplemental safety evaluation report (SSER) for the license renewal application (LRA) for Davis-Besse Nuclear Power Station (Davis-Besse) as submitted by FirstEnergy Nuclear Operating Company (FENOC). By letter dated August 27, 2010, FENOC submitted its LRA to the U.S. Nuclear Regulatory Commission (NRC) for renewal of the Davis-Besse operating licenses for an additional 20 years. The NRC staff (the staff) issued a safety evaluation report (SER) related to the license renewal of Davis-Besse Nuclear Power Station, dated September 3, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13248A267), which summarizes the results of its review of the LRA for compliance with the requirements of Title 10, Part 54, of the Code of Federal Regulations (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

This SSER documents the staff's review of supplemental information provided by the applicant since the issuance of the SER. This information includes annual updates required by 10 CFR 54.21(b) and updated information and commitments in response to the recent industry operating experience. This SSER supplements portions of SER Sections 1, 2, 3, 4, and Appendices.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
Safety Evaluation Report, SER, Davis-Besse Nuclear Power Plant, Davis-Besse, FirstEnergy Nuclear Operating Company, FENOC, License Renewal, License Renewal Application, LRA

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page)
unclassified
(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

Printed
on recycled
paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

@NRCgov

NUREG-2193
Supplement 1

Safety Evaluation Report Related to the License Renewal of
Davis-Besse Nuclear Power Station

April 2016