Steven M. Snider
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RA-22-0036
February 14, 2022

10 CFR 50.4
10 CFR Part 54

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, 50-287
Renewed License Numbers DPR-38, DPR-47, DPR-55
Subsequent License Renewal Application
Responses to NRC Request for Additional Information Set 2

References:

1. Duke Energy Letter (RA-21-0132) dated June 7, 2021, Application for Subsequent Renewed Operating Licenses, (ADAMS Accession Number ML21158A193)
2. NRC Letter dated July 22, 2021, Oconee Nuclear Station, Units 1, 2, and 3 - Determination of Acceptability and Sufficiency for Docketing, Proposed Review Schedule, and Opportunity for a Hearing Regarding Duke Energy Carolinas' Application for Subsequent License Renewal (ADAMS Accession Number ML21194A245)
3. NRC E-mail dated September 22, 2021, Oconee SLRA - Request for Additional Information B2.1.27-1 (ADAMS Accession Number ML21271A586)
4. Duke Energy Letter (RA-21-0281) dated October 22, 2021, Subsequent License Renewal Application, Response to Request for Additional Information B2.1.27-1 (ADAMS Accession Number ML21295A035)
5. NRC E-mail dated November 23, 2021, Oconee SLRA – Request for Additional Information Set 1 and Second Round Request for Additional Information RAI B2.1.27-1a (ADAMS Accession Number ML21327A277)
6. Duke Energy Letter (RA-21-0332) dated January 7, 2022, Oconee SLRA – Responses to NRC Request for Additional Information Set 1 and Second Round Request for Additional Information RAI B2.1.27-1a (ADAMS Accession Number ML22010A129)
7. NRC E-mail dated January 11, 2022, Oconee SLRA – Final Request for Additional Information - Set 2 (ADAMS Accession Number ML22012A043)

Ladies and Gentlemen:

By letter dated June 7, 2021 (Reference 1), Duke Energy Carolinas, LLC (Duke Energy) submitted an application for the subsequent license renewal of Renewed Facility Operating License Numbers DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station (ONS), Units 1, 2, and 3 to the U.S. Nuclear Regulatory Commission (NRC). On July 22, 2021 (Reference 2), the NRC determined that ONS subsequent license renewal application (SLRA) was acceptable and sufficient for docketing. In emails from Angela X. Wu (NRC) to Steve Snider (Duke Energy) dated September 22, 2021 and November 23, 2021 (References 3 and 5), the NRC transmitted specific requests for additional information (RAI) to support completion of the Safety Review. The responses (References 4 and 6) were provided to the NRC on October 22, 2021 and January 7, 2022. In an email from Angela X. Wu (NRC) to Steve Snider (Duke Energy) dated January 11, 2022 (Reference 7), the NRC transmitted RAI set 2 also to support completion of the Safety Review. This submittal provides those responses.

Enclosure 1 contains the responses to the RAI information for Set 2. Enclosure 1, Attachments 14P, 15P, 25P, 27P, 28P, 29P, 30P, 31P, 32P, 42P, and 43P contain proprietary information. Enclosure 2, Attachments 1 and 2 contain affidavits for the proprietary information.

Since Enclosure 1 contains proprietary information, it is supported by affidavits signed by the owner of the information (Enclosure 2). The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4) and consistent with NRC Regulatory Issue Summary 2014-11, Regulatory Requirements for Withholding of Proprietary Information from Public Disclosure. Accordingly, it is respectfully requested that the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390. A redacted, non-proprietary version is provided in Enclosure 1, Attachments 14, 15, 25, 27, 28, 29, 30, 31, 32, 42 and 43. Correspondence with respect to the copyright or proprietary aspects of the vendor/utility information or affidavits should be addressed to the vendor/utility representative identified in the respective affidavit.

SLRA changes are provided along with the affected SLRA section(s), SLRA page number(s), and SLRA mark-ups in each affected Enclosure 1 attachment. For clarity, deletions are indicated by strikethrough and inserted text by underlined red font.

The following commitments in Appendix A, Table A6.0-1, Subsequent License Renewal Commitments have been revised to align the commitments with the respective RAI responses:

- Commitment 8 (Enclosure 1, Attachment 34)
- Commitment 9 (Enclosure 1, Attachments 2 and 4),
- Commitment 16 (Enclosure 1, Attachment 36),
- Commitment 28 (Enclosure 1, Attachment 52),
- Commitment 30 (Enclosure 1, Attachments 8 and 52),
- Commitment 33 (Enclosure 1, Attachments 45 through 48, 50 and 52), and
- Commitment 48 (Enclosure 1, Attachment 10).

Should you have any questions regarding this submittal, please contact Paul Guill at (704) 382-4753 or by email at paul.guill@duke-energy.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 14, 2022.

Sincerely,

Steven M. Snider
Site Vice President
Oconee Nuclear Station

Enclosures:

Enclosure 1: Responses to NRC Request for Additional Information Set 2

Attachment	RAI Number
1	3.3.2.2.2-1
2	B2.1.9-1
3	B2.1.9-2
4	B2.1.9-3
5	A2.9-1
6	3.5.2.2.1.5-1
7	3.5.2.2.1.6-1
8	B2.1.30-1
9	3.5.2.2.1.2-1
10	B4.1-1
11	B4.1-2
12	4.6.1-1
13	4.6.3-1
14	4.2.3-1 – Non-Proprietary Version
14P	4.2.3-1 – Proprietary Version
15	4.2.3-2– Non-Proprietary Version
15P	4.2.3-2 – Proprietary Version
16	B2.1.15-1
17	B2.1.15-2
18	B2.1.10-1
19	B2.1.10-2
20	B2.1.10-3
21	B2.1.10-4
22	2.3.1.3-1
23	2.3.1.3-2
24	B2.1.11-1
25	3.5.2.2.2.6-1 – Non-Proprietary Version
25P	3.5.2.2.2.6-1 – Proprietary Version
26	3.5.2.2.2.6-2
27	3.5.2.2.2.2-3 – Non-Proprietary Version
27P	3.5.2.2.2.2-3 – Proprietary Version
28	3.5.2.2.2.6-4 – Non-Proprietary Version

Attachment	RAI Number
28P	3.5.2.2.2.6-4 – Proprietary Version
29	3.5.2.2.2.6-5 – Non-Proprietary Version
29P	3.5.2.2.2.6-5 – Proprietary Version
30	3.5.2.2.2.6-6 – Non-Proprietary Version
30P	3.5.2.2.2.6-6 – Proprietary Version
31	3.5.2.2.2.6-7 – Non-Proprietary Version
31P	3.5.2.2.2.6-7 – Proprietary Version
32	3.5.2.2.2.6-8 – Non-Proprietary Version
32P	3.5.2.2.2.6-8 – Proprietary Version
33	B2.1.8-1
34	B2.1.8-2
35	B2.1.16-1
36	B2.1.16-2
37	B2.1.16-3
38	B.2.3.2-1
39	3.4.2.2.2-1
40	B2.1.7-1
41	B2.1.7-2
42	B2.1.7-3 – Non-Proprietary Version
42P	B2.1.7-3 – Proprietary Version
43	B2.1.7-4 – Non-Proprietary Version
43P	B2.1.7-4 – Proprietary Version
44	B2.1.7-5
45	B2.1.33-1
46	B2.1.33-2
47	B2.1.33-3
48	B2.1.33-4
49	B2.1.33-5
50	3.5.1-092-1
51	3.5.1-093-1
52	B2.1.28-1

Enclosure 2: Affidavits

Attachment 1 Framatome Affidavit
Attachment 2 Duke Energy Affidavit

CC: W/O Enclosures:

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ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

Enclosure 1
Subsequent License Renewal Application
Responses to Requests for Additional Information Set #2

Requests for Additional Information (RAI) Attachments Index	
Attachment No.	**RAI No.**
1	3.3.2.2.2-1
2	B2.1.9-1
3	B2.1.9-2
4	B2.1.9-3
5	A2.9-1
6	3.5.2.2.1.5-1
7	3.5.2.2.1.6-1
8	B2.1.30-1
9	3.5.2.2.1.2-1
10	B4.1-1
11	B4.1-2
12	4.6.1-1
13	4.6.3-1
14, 14P	4.2.3-1
15, 15P	4.2.3-2
16	B2.1.15-1
17	B2.1.15-2
18	B2.1.10-1
19	B2.1.10-2
20	B2.1.10-3
21	B2.1.10-4
22	2.3.1.3-1
23	2.3.1.3-2
24	B2.1.11-1
25, 25P	3.5.2.2.2.6-1
26	3.5.2.2.2.6-2
27, 27P	3.5.2.2.2.2-3
28, 28P	3.5.2.2.2.6-4
29, 29P	3.5.2.2.2.6-5
30, 30P	3.5.2.2.2.6-6
31, 31P	3.5.2.2.2.6-7
32, 32P	3.5.2.2.2.6-8
33	B2.1.8-1
34	B2.1.8-2
35	B2.1.16-1
36	B2.1.16-2
37	B2.1.16-3
38	B.2.3.2-1
39	3.4.2.2.2-1
40	B2.1.7-1
41	B2.1.7-2

Requests for Additional Information (RAI) Attachments Index	
Attachment No.	**RAI No.**
42, 42P	B2.1.7-3
43, 43P	B2.1.7-4
44	B2.1.7-5
45	B2.1.33-1
46	B2.1.33-2
47	B2.1.33-3
48	B2.1.33-4
49	B2.1.33-5
50	3.5.1-092-1
51	3.5.1-093-1
52	B2.1.28-1

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 1
RAI 3.3.2.2.2-1

Enclosure 1, Attachment 1

RAI 3.3.2.2.2-1:

Regulatory Basis:
Title 10 of the Code of Federal Regulations (CFR) Section 54.21(a)(1) requires a license renewal application to contain an integrated plant assessment (IPA) that identifies and lists structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the period of extended operation. To complete its review and enable the staff to make a reasonable assurance finding on functionality of reviewed structures and components for the period of extended operation consistent with 10 CFR 54.21, the staff requires under 10 CFR 54.29(a) additional information be provided regarding the matters described below.

Background:
With Supplement 1 of the Subsequent License Renewal Application (SLRA) for Oconee Nuclear Station (ONS), Units 1, 2, and 3, Duke Energy Carolinas, LLC (Duke Energy or the applicant), revised its SLRA (ADAMS Accession No. ML21302A208). The applicant's revision added the letdown cooler tubes to the scope of subsequent license renewal (SLR). However, the applicant also stated that the letdown cooler assemblies are replaced on a specified frequency, therefore they are not subject to AMR in accordance with 10 CFR 54.21 (a)(1)(ii). Accordingly, the applicant removed AMR line items related to the head and the shell of the letdown coolers from SLRA Tables 2.3.2-3 and 3.2.2-3. Additionally, the applicant revised SLRA Section 3.3.2.2.2, which is related to the further evaluation of the ONS letdown coolers due to cracking as a result of stress corrosion cracking (SCC) and cyclic loading. The revision states, in part, that the ONS "letdown coolers are replaced on a specified time period and are short-lived components not subject to aging management."

However, the NRC staff observed that the ONS letdown coolers have been subject to replacements, design changes, changes in operation, and repairs due to frequent tube failures. Additionally, during initial license renewal, Duke credited several programs for managing the aging effects of the ONS letdown coolers during the period of extended operations (PEO).

Issue:
Even though the applicant has now determined that the ONS letdown coolers are not subject to AMR due to periodic replacement, the replacement intervals should account for the history of multiple failures of the ONS letdown coolers to ensure the integrity of reactor coolant pressure boundary of the ONS letdown coolers during the subsequent period of extended operation (SPEO).

Specifically, during the audit, the staff became aware that laboratory examinations were performed on an ONS letdown cooler which had developed a reactor coolant pressure boundary leak during the PEO and was removed from service. These examinations revealed that the leakage was due to SCC.

Request:
1. Explain any actions taken to mitigate SCC of the letdown coolers for the SPEO and the methodology used to establish a reasonable replacement interval for the letdown coolers.
2. Clarify the ONS programs which will be used to monitor the performance of the replacement letdown coolers so that there is a reasonable assurance that pressure boundary failure due to SCC of the ONS letdown coolers will not occur prior to scheduled replacement during the SPEO.

Response to RAI 3.3.2.2.2-1:

Request 1

The letdown coolers are passive components that perform a license renewal component intended function, but do not require an aging management review because they are replaced on a specified time period. Paragraph 54.21(a)(1)(ii) of 10 CFR 54 states that aging management is not required for components subject to replacement based on qualified life or specified time period. The specified time period for replacement has been established for the letdown coolers. The time period of 14 years was established based on both plant and industry operating experience as well as metallurgical testing. In addition, operational indicators are in place to assure continual oversight of the performance of the letdown cooler prior to scheduled replacement.

Request 2:

Operational performance of the letdown coolers is managed by monitoring for radiation through several means. These means, in part, serve to meet ONS Technical Specification 3.4.13. Operating experience used to establish the replacement frequency reflects that previous cooler leaks have not resulted in sudden, large leaks. Past leaks have started off as very small leaks that grow very slowly over time. Monitoring for radioactivity has proven effective at monitoring cooler leaks at low leak rates that occur prior to the leak rates being detectable. Radiation monitors are installed in the Component Cooling Water System that continuously monitor the system for radioactivity.

In addition, ONS Chemistry personnel routinely sample the Component Cooling Water monthly and in response to radiation alarms. These activities provide reasonable assurance that letdown cooler performance is monitored within the operating interval between scheduled replacement.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 2
RAI B2.1.9-1

Enclosure 1, Attachment 2

RAI B2.1.9-1:

<u>Regulatory Basis</u>:
Title 10 of the Code of Federal Regulations (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

<u>Background</u>:
SLRA Section B2.1.9, "Bolting Integrity," states that the Oconee Bolting Integrity AMP, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALL-SLR Report AMP XI.M18, "Bolting Integrity." For the "preventive actions" program element, the GALL-SLR Report AMP states that the use of molybdenum disulfide (MoS_2) as a lubricant has been shown to be a potential contributor to stress corrosion cracking (SCC) and should not be used.

In the program description, the SLRA claims that the program already includes preventive measures to prohibit the use of lubricant containing MoS_2. To verify this claim, the staff audited the program's procedures and references (e.g., MP/0/A/1800/003, MP/0/A/1200/108, MP/0/A/1800/003A, Power Chemistry Material Guide (PCMG) Program) to better understand how the Oconee Bolting Integrity AMP is being consistent with the GALL-SLR Report AMP.

<u>Issue</u>:
Based on the review of the procedures associated with the bolting integrity program, it is not clear how the program is consistent with the GALL-SLR Report AMP XI.M18 recommendation for including preventive actions that would preclude the use of MoS_2 as a lubricant in closure bolting for pressure retaining components.

During the review, the staff noted some inconsistencies in the preventive measures used by each procedure to prohibit the use of lubricant containing MoS_2. Some procedures specified that lubricant material must be selected in accordance with the PCMG program, and other procedures directed the use of "N-5000 or equivalent" as a lubricant apparently without any clear guidance on how the use of lubricant containing MoS_2 in closure bolting was restricted/limited.

It was also noted that Oconee follows the PCMG program to provide guidelines and limitations on materials that will be used in contact with safety related and non-safety related plant systems. However, this program was not described in the application for the bolting integrity program. Furthermore, it is not clear how this guidance will be sufficient to demonstrate consistency with the GALL-SLR Report since not all procedures clearly directed its use when selecting the lubricant for bolting material, nor is it clear those programs provided a clear action to prevent the use of lubricant material containing MoS_2.

<u>Request</u>:
Considering the issues identified above, clarify how procedures associated with the Bolting Integrity programs will be consistent with the GALL-SLR Report to demonstrate that clear preventive actions will be implemented to restrict the use of molybdenum disulfide (MoS_2) as a lubricant.

Response to RAI B2.1.9-1:

The Oconee *Bolting Integrity* AMP provides sufficient guidance in existing procedures to ensure that products containing molybdenum disulfide (MoS_2) are restricted from use as bolting thread lubricants. The nuclear chemical control procedure governs the use of all chemicals at Oconee, including bolting thread lubricants. The purpose of the nuclear chemical control process is to provide consistent practices for the nuclear fleet regarding all aspects of chemical control, including defining the method in which chemicals are approved for use on site and ensuring that only chemicals that have been approved for use are procured. In addition, the nuclear chemical control process drives implementation of PCMG requirements.

The PCMG provides additional requirements and limitations for materials to be used in contact with the safety related and non-safety related plant systems, structures, and components at Oconee. These requirements and limitations are based on the potential for initiation or promotion of degradation due to the chemical effect that certain substances have on the materials of construction of plant equipment. The PCMG establishes chemical impurity limits and provides for testing and categorization (approval categories) of chemicals used in the plant. The PCMG is implemented through an on-line database that is available to plant personnel and captures the results of the analyses and categorization of the various chemical materials used in the plant. The use of molybdenum disulfide as a thread lubricant has been shown to be a potential contributor to stress corrosion cracking. Therefore, the use of molybdenum disulfide as a thread lubricant on bolting is prohibited at Oconee as specified in the PCMG and, as such, a product containing MoS_2 is not, and would not be, approved for use as a bolting thread lubricant in the plant.

Maintenance procedures associated with the installation of closure bolting for pressure retaining components frequently specify a particular thread lubricant that has been approved for use in accordance with PCMG. Alternatively, maintenance procedures may state that a thread lubricant be selected in accordance with PCMG requirements. Although individual maintenance procedures may not directly reference the PCMG, the work planning process ensures that all chemicals used on station equipment, including bolting thread lubricants, meet the requirements of the PCMG. All chemicals provided for use on station equipment are issued to maintenance personnel through approved work orders. In accordance with the fleet work order planning procedure, work planners are required to ensure that chemicals issued for use in the field meet the requirements of the PCMG. Therefore, regardless of whether a maintenance procedure explicitly references the PCMG, the work planning process ensures that all thread lubricants issued to maintenance personnel comply with PCMG requirements.

To ensure that chemicals are not inadvertently used in unapproved applications, each chemical product used on site is labeled upon receipt to clearly display the PCMG approval category. In accordance with the PCMG, a product containing MoS_2 would not be labeled for use as a bolting thread lubricant in the plant. All site personnel are required to understand and comply with the requirements of the nuclear chemical control process and PCMG. Managers and supervisors are required to ensure that employees whose work activities include the use, storage and disposal of chemicals, comply with the requirements of the nuclear chemical control process and PCMG. Finally, technicians at Oconee receive instruction on the PCMG as part of the initial maintenance training program. The lesson familiarizes the student with information available in the PCMG and its use. The training stresses how improper application of chemical materials could have detrimental effects on plant components.

Based on the above, existing procedural guidance exists to ensure that products containing MoS_2 are restricted from use as bolting thread lubricants. The *Bolting Integrity* AMP is enhanced to specifically document the prohibition of the use of bolting thread lubricants containing MoS_2 in the fleet work planning procedure.

SLRA Revisions:

SLRA Appendix A2.9 (Page A-11 as revised by Supplement 1 (ML21302A208)) is revised as follows:

A2.9 Bolting Integrity

Enhancements

The *Bolting Integrity* AMP will be enhanced to:

5. Perform additional inspections of a minimum of 20% of similar bolting or five additional inspections, whichever is less, for each sample based inspection (each bolt, etc.) that does not meet acceptance criteria. If the additional inspections identify bolting that does not meet acceptance criteria, then an extent of condition and extent of cause analysis will be conducted to determine the further extent of inspections. Additional inspections of similar bolting (same material, environment, and aging effect(s)) will be performed at all three units and will occur within the same interval in which the original inspection was conducted. The corrective action program will be used to determine if changes to inspection frequency is appropriate if any inspection results indicate that loss of function will occur prior to the next scheduled inspection.

6. Revise applicable procedures and specifications to ensure that the use of bolting with measured yield strength greater than or equal to 150 kilo-pounds per square inch (ksi) or 1,034 megapascals (MPa) in newly designed applications is avoided when practical engineering design considerations allow.

7. **Revise the fleet work planning procedure to specifically prohibit the use of products containing molybdenum disulfide (MoS_2) as bolting thread lubricants.**

SLRA Table A6.0-1 (page A-76 as revised by Supplement 1 (ML21302A208)) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
9	*Bolting Integrity* program	5. Perform additional inspections of a minimum of 20% of similar bolting or five additional inspections, whichever is less, for each sample based inspection (each bolt, etc.) that does not meet acceptance criteria. If the additional inspections identify bolting that does not meet acceptance criteria, then an extent of condition and extent of cause analysis will be conducted to determine the further extent of inspections. Additional inspections of similar bolting (same material, environment, and aging effect(s)) will be performed at all three units and will occur within the same interval in which the original inspection was conducted. The corrective action program will be used to determine if changes to inspection frequency is appropriate if any inspection results indicate that loss of function will occur prior to the next scheduled inspection. 6. Revise applicable procedures and specifications to ensure that the use of bolting with measured yield strength greater than or equal to 150 kilo-pounds per square inch (ksi) or 1,034 megapascals (MPa) in newly designed applications is avoided when practical engineering design considerations allow. **7. Revise the fleet work planning procedure to specifically prohibit the use of products containing molybdenum disulfide (MoS_2) as bolting thread lubricants.**		

SLRA Appendix B2.1.9 (Page B-89 as revised by Supplement 1 (ML21302A208)) is revised as follows:

B2.1.9 BOLTING INTEGRITY

Enhancements

5. Perform additional inspections of a minimum of 20% of similar bolting or five additional inspections, whichever is less, for each sample based inspection (each bolt, etc.) that does not meet acceptance criteria. If the additional inspections identify bolting that does not meet acceptance criteria, then an extent of condition and extent of cause analysis will be conducted to determine the further extent of inspections. Additional inspections of similar bolting (same material, environment, and aging effect (s)) will be performed at all three units and will occur within the same interval in which the original inspection was conducted. The corrective action program will be used to determine if changes to inspection frequency is appropriate if any inspection results indicate that loss of function will occur prior to the next scheduled inspection. (Element 7)

6. Revise applicable procedures and specifications to ensure that the use of bolting with measured yield strength greater than or equal to 150 kilo-pounds per square inch (ksi) or 1,034 megapascals (MPa) in newly designed applications is avoided when practical engineering design considerations allow. (Element 2)

7. **Revise the fleet work planning procedure to specifically prohibit the use of products containing molybdenum disulfide (MoS_2) as bolting thread lubricants.**

Operating Experience

Based on a broad search of pertinent Oconee OE, the following examples provide objective evidence that the *Bolting Integrity* AMP will continue to be effective in managing aging effects for SSCs within the scope of the program so that intended functions will be maintained consistent with the current licensing basis for the SPEO. This broad search included the corrective action program, metallurgical laboratory reports, programmatic inspection results and effectiveness reviews for existing AMPs.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 3
RAI 2.1.9-2

Enclosure 1, Attachment 3

RAI B2.1.9-2:

Regulatory Basis:
Title 10 of the Code of Federal Regulations (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section B2.1.9, "Bolting Integrity," states that the Oconee Bolting Integrity AMP, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALL-SLR Report AMP XI.M18, "Bolting Integrity." To ensure consistency with the "detection of aging effects" program element, the SLRA included enhancement no. 4 to demonstrate that the program will manage the inspections of closure bolting in locations where the detection of joint leakage is precluded or for which leakage is difficult to detect.

For the "detection of aging effects" program element, the GALL-SLR Report AMP provides, in part, inspection criteria and guidance to demonstrate that submerged closure bolting, closure bolting in systems containing air or gas, and closure bolting in components that are not normally pressurized will be adequately managed by the program. For submerged closure bolting, the GALL-SLR Report recommends the use of visual inspection to detect loss of material during opportunistic maintenance activities (e.g., when made accessible, and when joints are disassembled). The SLRA does not state how integrity of the bolted joints will be maintained (through alternate means of inspections or testing) when opportunistic maintenance activities will not provide access to at least 20 percent of the population, or the applicable sample size for the site, over a 10-year period. In a similar way, for closure bolting in systems containing air or gas, the GALL-SLR Report recommends that the SLRA states how integrity of the bolted joint will be demonstrated through the proposed inspection method, and for the closure bolting in components that are not normally pressurized, it recommends that the SLRA states how the aging effects associated with the closure bolting will be managed based on the proposed inspection method. In addition, for the "acceptance criteria" program element, the GALL-SLR Report AMP also states, in part, that plant-specific acceptance criteria are established when alternative inspections or testing is conducted for submerged closure bolting or closure bolting where the piping systems contains air or gas for which leakage is difficult to detect.

Issue:
During the staff review of the SLRA, the staff noted that the SLRA does not state how the aging effects associated with closure bolting for components that are not normally pressurized will be detected and managed so that the intended function(s) will be maintained consistent with the current licensing basis. Specifically, the SLRA enhancement no. 4 seems to specifically address the detection of aging effects in submerged closure bolting and in closure bolting where the piping systems containing air or gas. Therefore, it is not clear how closure bolting from systems that are not normally pressurized will be adequately managed, and what alternate means of inspection and acceptance criteria will be implemented (e.g., checking the torque to the extent that the closure bolting is not loose) to ensure that the associated aging effects will be detected before a loss of function.

The staff also noted that SLRA enhancement no. 4 seeks to implement alternate means of inspection and testing when the minimum sample size is not met over a 10-year period. However, it is not clear what plant-specific acceptance criteria will be established for these alternative means of inspections and testing to demonstrate that these components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the subsequent period of extended operation.

Request:

1. State how the aging effects associated with the closure bolting for components that are not normally pressurized will be detected and adequately managed by the Bolting Integrity program during the subsequent period of extended operation (i.e., inspection methods and acceptance criteria that will be used). Update the SLRA as necessary to include this information.

2. For the alternate inspection and testing methods specified in the SLRA for submerged closure bolting or closure bolting where leakage is difficult to detect, clarify what the plant-specific acceptance criteria are that will be established for the Bolting Integrity program to ensure that the intended function(s) will be maintained consistent with the current licensing basis for the subsequent period of extended operation. Update the SLRA as necessary.

Response to RAI B2.1.9-2:

Request 1:

Closure bolting in systems that are not normally pressurized will be visually inspected during maintenance activities for degradation including loss of material, visible cracking, and loss of preload (loose or missing bolting). Bolt heads will be inspected when made accessible and bolt threads will be inspected when joints are disassembled (opportunistic inspection). A representative sample population for each ONS unit will consist of 20% of the population up to a maximum of 17 bolts for each material/ environment combination. Based on a review of existing periodic preventive maintenance activities, it is expected that sufficient opportunistic inspections will be performed to greatly exceed the representative sample size required by the AMP.

Closure bolting in air or gas filled systems that are not normally pressurized but contain air or gas (e.g., exhaust lines) will be managed consistent with the approach used for closure bolting on normally pressurized air or gas filled systems. Alternative inspections may include, (a) visual inspection for discoloration when leakage from inside the piping system would discolor the external surfaces of the component; (b) monitoring and trending of pressure decay when the bolted connection is located within an isolated boundary; (c) soap bubble testing on the external mating surface of the bolted component; or (d) thermography, when the temperature of the process fluid is higher than ambient conditions around the component. Other than visual inspections for discoloration, if these alternative inspections are required for air or gas filled systems that are not normally pressurized, then they would be performed while the system is pressurized to ensure leakage is detected. Visual inspections for discoloration do not require the system to be pressurized during the inspection since the intent of the inspection is to identify evidence of prior leakage rather than active leakage.

Closure bolting in water, steam, or oil filled systems that are not normally pressurized will be managed consistent with the approach used for closure bolting on normally pressurized water, steam, or oil filled systems. Standby systems that are not normally pressurized are typically maintained filled with fluid such that degradation of bolted joints would result in detectable leakage. All ASME Code Class 1, Class 2, or Class 3 systems, including safety-related systems that are normally maintained depressurized in standby, are subject to periodic VT-2 testing with the system pressurized such that leakage from bolted joints would be detected. In addition, although certain systems may not normally be pressurized, system pressurization occurs during testing and surveillance, infrequent plant operations such as startup and shutdown, and system makeup activities such that adequate opportunities exist to detect leakage from bolted joints during these activities. The periodic inspections required by this program include inspection for signs of past leakage (including discoloration/staining and damaged insulation) and loose or missing bolting and provide reasonable assurance that leakage from water, steam, or oil filled systems that are not normally pressurized will be detected.

Request 2:

Aging management of submerged closure bolting or closure bolting for systems containing air or gas where leakage is difficult to detect will primarily rely on opportunistic inspections performed when bolted joints are disassembled during maintenance activities. If opportunistic inspections do not provide an adequate sample size, then alternative inspections may be performed.

Alternative inspections for submerged bolting include diver inspections and remote video/photo inspections. Submerged bolted connections where diver inspections are performed will include visual inspections for degraded bolts, missing or broken bolts and, where possible, the torque of bolts verified to be hand tight. Remote video and photo inspections may be performed to inspect for degraded, loose, or missing bolts. Evidence of loose or missing bolting and significant loss of material (i.e., appreciable material loss that could adversely affect intended function) identified during inspections is unacceptable and will be entered into the corrective action program, if identified.

Alternative inspections for air or gas filled systems rely on alternative methods for detecting leakage since leakage of these systems cannot be detected visually and include (a) visual inspection for discoloration when leakage from inside the piping system would discolor the external surfaces of the component; (b) monitoring and trending of pressure decay when the bolted connection is located within an isolated boundary; (c) soap bubble testing on the external mating surface of the bolted component; or (d) thermography. The acceptance criteria for these systems is the same as for water, steam, or oil filled systems in that indications of leaking joints are unacceptable and, if identified, will be documented in the corrective action program. For exhaust lines, visual inspections for discoloration will provide indication of bolted joint leakage. For portions of air or gas systems within an isolated boundary, increased pressure decay rate will provide indication of bolted joint leakage. For other portions of air or gas systems, soap bubble testing allows for detection of leakage from bolted joints. Where the system process fluid is a different temperature than the surrounding environment, thermal imaging may be used to detect leaking joints as means to detect degraded bolted connections.

SLRA Revisions:

SLRA Revisions were not required.

ENCLOSURE 1

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ATTACHMENT 4
RAI 2.1.9-3

Enclosure 1, Attachment 4

RAI B2.1.9-3:

Regulatory Basis:
Title 10 of the Code of Federal Regulations (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section B2.1.9, "Bolting Integrity," states that the Oconee Bolting Integrity AMP, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALLSLR Report AMP XI.M18, "Bolting Integrity." To ensure consistency with the "detection of aging effects" program element, the SLRA included enhancement no. 3 to demonstrate that the program will perform volumetric inspections of non-ASME high-strength bolting greater than two inches in diameter in accordance with the methods described in ASME Code Section XI, Table IWB 2500 1, Examination Category BG1.

For the "parameters monitored or inspected," and "detection of aging effects" program elements, the GALL-SLR Report states that high strength closure bolting with actual yield strength greater than or equal to 150 ksi, and bolting for which yield strength is unknown, maybe subject to stress corrosion cracking (SCC), and it should be monitored for surface and subsurface discontinuities indicative of cracking. The GALL-SLR Report also states that for all closure bolting greater than 2 inches in diameter (regardless of code classification) with actual yield strength greater than or equal to 150 ksi and closure bolting for which yield strength is unknown, volumetric examination in accordance with ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, is performed.

Issue:
Regarding SLRA enhancement no. 3, it is not clear how this enhancement is consistent with the GALL-SLR Report recommendation to demonstrate that the aging effects of cracking due to SCC will be adequately managed for high-strength bolting. Specifically, the enhancement is not clear whether the proposed volumetric inspection applies to bolting with actual yield strength greater than or equal to 150 ksi and closure bolting for which yield strength is unknown.

Request:
Clarify how SLRA enhancement no. 3 to the Bolting Integrity program is consistent with the GALL-SLR Report recommendation to ensure that the aging effects of cracking due to SCC will be adequately manage for high-strength bolting with actual yield strength greater than or equal to 150 ksi and closure bolting for which yield strength is unknown.

Response to RAI B2.1.9-3:

ONS agrees that Enhancement No. 3 to the Bolting Integrity AMP should be clarified to define "high-strength bolting" as bolting with actual measured yield strength greater than or equal to 150 ksi (1,034 MPa). This clarification is provided below. The only bolting greater than 2 inches in diameter within the scope of the ONS Bolting Integrity AMP determined to be susceptible to stress corrosion cracking due to the use of high strength material is bolting which has an actual yield strength greater than 150 ksi installed in the Main Steam System main turbine stop and control valves and the Turbine and Auxiliaries System high and low pressure turbine casings. This was determined based on the actual measured yield strength or minimum specified yield strength documented in design and procurement specifications, fabrication and vendor drawings, and material test reports for bolting within the scope of the AMP.

The technical basis for GALL-SLR provided in NUREG-2221 states that “The provision for closure bolting for which yield strength is unknown was added to ensure that all closure bolting that is potentially high-strength would have volumetric inspections for cracking.” As documented in NUREG-1339, installed bolting could potentially have an actual yield strength that greatly exceeds the specified minimum yield strength. However, a large discrepancy between the specified minimum yield strength and actual yield strength would be due to non-aging issues such as installation errors (e.g., bolting of higher strength than specified was installed during construction) or manufacturing defects (e.g., improper heat treatment). As such, use of minimum specified yield strength values if actual yield strength test results are unavailable to identify bolting potentially susceptible to stress corrosion cracking is appropriate in the context of aging management provided adequate margin is included to account for normal variability between the specified minimum yield strength and actual yield strength.

If actual measured yield strength was unavailable, Oconee relied on a conservative minimum yield strength threshold of 120 ksi to identify bolting greater than 2 inches in diameter that is potentially susceptible to stress corrosion cracking due to the use of high strength material. The only bolting identified in the IPA performed during the development of the SLRA with a minimum specified yield that exceeds 120 ksi is the high strength bolting in the Main Steam System main turbine stop and control valves and the Turbine and Auxiliaries System high and low pressure turbine casings. There is reasonable assurance that bolting with a specified minimum yield strength less than 120 ksi has an actual yield strength of less than 150 ksi and, therefore, all bolting other than the previously identified high strength bolting was evaluated as non-high strength for SLR. This position is consistent with NRC and industry guidance regarding stress corrosion cracking of high strength bolting and is further supported based on empirical evidence from prior studies conducted on the difference between actual and minimum bolting yield strength and favorable operating experience at Oconee indicating no mechanical closure bolting failures due to stress corrosion cracking (SCC) since the implementation of an effective bolting integrity program.

SLRA Revisions:

SLRA Appendix A2.9 (page A-11) is revised as follows:

Enhancements

The *Bolting Integrity* AMP will be enhanced to:

3. Perform volumetric inspections of non-ASME high strength bolting **(with actual yield strength greater than or equal to 150 ksi (1,034 MPa))** greater than 2 inches in diameter in accordance with the method described in ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1.

SLRA Table A6.0-1 (page A-75) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
9	*Bolting Integrity* program	The *Bolting Integrity* AMP is an existing program that will be enhanced to: 1. Revise applicable procedures and specifications to include reference to EPRI Report 1015336, EPRI Report 1015337, and NUREG-1339, as appropriate. 2. Revise procedures governing the direct visual examination of bolted joints to include inspection parameters such as lighting, distance, and offset. 3. Perform volumetric inspections of non-ASME high strength bolting **(with actual yield strength greater than or equal to 150 ksi (1,034 MPa))** greater than 2 inches in diameter in accordance with the method described in ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1.	B2.1.9	Program enhancements for SLR will be implemented six months prior to the SPEO.

SLRA Appendix B2.1.9 (page B-88) is revised as follows:

Enhancements

3. Perform volumetric inspections of non-ASME high strength bolting **(with actual yield strength greater than or equal to 150 ksi (1,034 MPa))** greater than 2 inches in diameter in accordance with the method described in ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1.

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ATTACHMENT 5
A2.9-1

Enclosure 1, Attachment 5

RAI A2.9-1:

Regulatory Basis:
Title 10 of the Code of Federal Regulations (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section A2.9, "Bolting Integrity," provides the program description that will be used to supplement the UFSAR for the Bolting Integrity AMP. As stated, in part, in 10 CFR 54.21(d), the "FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by paragraphs (a) and (c) of this section." Table XI-01 of the GALL-SLR Report provides a recommended description of the Bolting Integrity program.

During the audit, the staff noted that existing procedures do not appear to directly manage the inspections of closure bolting in locations that preclude detection of joint leakage or where the system contains air or gas for which leakage is difficult to detect (e.g., air or gas systems, not normally pressurized). Therefore, enhancement no. 4 in SLRA Section B2.1.9 seeks to implement new actions, either as a new procedure or as an enhancement to existing procedure(s), to ensure that closure bolting in locations that preclude detection of joint leakage or where the leakage is difficult to detect is adequately managed during the subsequent period of extended operations, consistent with the GALL-SLR Report recommendations. These actions include the use of alternate means of inspections to ensure that aging effects can be detected for these components.

Issue:
During the staff review of the UFSAR supplement, the staff noted that the program description does not appear to contain a summary description of the AMP that is consistent with the program and actions described in SRLA Section B2.1.9 and/or the GALL-SLR Report. Specifically, the program description does not include the alternate means of inspections (e.g., testing – soap bubble or thermography testing) that will be used by the Bolting Integrity program to ensure that the effects of aging for closure bolting for which leakage is difficult to detect can be detected and adequately managed before a loss of function.

Request:
Update the summary description of the Bolting Integrity program to provide a description that is consistent with the program and actions described in SRLA Section B2.1.9.

Response to RAI A2.9-1:

For Appendix A of the Oconee SLRA, UFSAR Summary Descriptions are comprised of two parts, "Program Description" and "Enhancements". The discussion provided in the two parts are included within the UFSAR Summary Descriptions. The "Enhancements" section of the UFSAR Summary Description contains the information requested by the RAI. Therefore, an update to UFSAR Summary Description for Appendix A2.9, Bolting Integrity program is not needed.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 6
RAI 3.5.2.2.1.5-1

Enclosure 1, Attachment 6

RAI 3.5.2.2.1.5-1:

Regulatory Basis:
Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Table 3.5.1 claims that AMR item 3.5.1-027 is not applicable, and it further states, "Cracking due to cyclic loading of the Containment liner and penetrations is a time-limited aging analysis (TLAA), as defined in 10 CFR 54.3. The evaluation of this TLAA is addressed in Section 4.6. The associated NUREG-2191 aging items are not used."

SLRA Section 3.5.2.2.2.1.5 states that TLAAs for fatigue of the containment liner plate and main feedwater and main steam penetrations are addressed in SLRA Section 4.6. However, SLRA Section 3.5.2.2.2.1.5 does not address fatigue or fatigue waiver analyses of containment pressure-retaining boundary components other than those above, nor provides any further evaluation associated with AMR item 3.5.1-027 for managing cracking due to cyclic loading of containment pressure-retaining boundary components of steel, stainless steel or dissimilar metal weld (DMW) material that do not have a CLB fatigue analysis.

Issue:
The non-applicability claim of AMR item 3.5.1-027 appears to be not adequately justified and it is unclear how the other steel, stainless steel or DMW containment pressure-retaining boundary components subject to cyclic loading, but do not have a CLB fatigue analysis, covered by item 3.5.1-027 will be adequately managed for cracking due to cyclic loading.

Request:
1. List the steel, stainless steel or DMW containment-pressure-retaining boundary components at ONS covered by SLRA Table 3.5.1, item 027 (e.g., personnel airlock, equipment hatch, electrical penetration, mechanical penetration, penetration sleeves, penetration bellows, fuel transfer tube etc.) that are subject to cyclic loading but do not have a CLB fatigue analyses.
2. Justify the non-applicability claim of SLRA Table 3.5.1, item 3.5.1-027, for each of these components. Alternatively, describe how cracking due to cyclic loading (cumulative fatigue damage) will be adequately managed for each of these components pursuant to 10 CFR 54.21(a)(3), or justify why the aging effect does not require management pursuant to guidance in SRP-SLR Section 3.5.2.2.1.5, as modified by Interim Staff Guidance SLR-ISG-2021-03-Structures. Provide necessary conforming changes to the SLRA accordingly.

Response to RAI 3.5.2.2.1.5-1:

Request 1:

NUREG-2191 (GALL-SLR) Item II.A3.CP-37 (SRP Item 3.5.1-027) addresses cracking due to cyclic loading in an air – indoor uncontrolled/air – outdoor environment for metal liner, metal plate, airlock, equipment hatch, Control Rod Drive hatch, penetration sleeves and penetration bellows constructed of steel, stainless steel, and DMWs that do not contain a CLB fatigue analysis. SLR-ISG-2021-03-Structures, Updated Aging Management Criteria for Structures Portions of Subsequent License Renewal Guidance, updated Section 3.5.2.2.1.5 to include that a plant-specific further evaluation may be performed to demonstrate that cracking due to cyclic loading is an aging effect that does not require aging management for the component.

For the ONS Units 1, 2, and 3, the containment liner plate and penetrations, including mechanical, electrical, equipment and personnel-related penetrations, have a CLB fatigue analyses. The penetrations mentioned above include the associated penetration sleeves. ONS does not have penetration bellows that are part of the containment pressure boundary. Therefore, the NUREG-2191 Item II.A3.CP-37 (SRP Item 3.5.1-027) is not applicable for ONS. The materials of construction for these components are steel, stainless steel, and DMWs. As described in the response to RAI 4.6.1-1 (Attachment 12), a fatigue waiver analysis was performed on the liner plate and cold penetrations, including mechanical, electrical, equipment and personnel-related penetrations. Since initial design and construction, two penetrations, main steam and main feedwater, have received additional detailed design transient analyses due to higher temperature cycling concerns. These two are termed hot penetrations and are discussed in ONS SLRA Section 4.6.3. The remaining penetrations that do not experience higher temperature cycling are termed cold penetrations. For the containment liner and cold penetrations, the analysis was determined to meet all six criteria in the ASME Code, Section III, Subsection N-415.1. No detailed fatigue analysis is required for the containment liner or cold penetrations. The main steam and maintain feedwater fatigue analyses are addressed in ONS SLRA Section 4.6.3.

It should be noted that there are four penetrations that contain DMWs. These are the ONS Unit 1 Pressurizer and Reactor Coolant Sample line and the ONS Units 1, 2, and 3 Steam Generators 1A, 2B, and 3B Sample lines. Each sample line is ½ inch in diameter. At a typical piping penetration, the carbon steel liner is welded to a carbon steel dished head and dished head then welded to the exterior of the carbon steel piping to form the leak tight containment barrier. For these four penetrations, a stainless steel transition piece, also called a split ring, is welded to the stainless steel pipe and also to the carbon steel dished head. The DMW from the ASTM A316 (stainless steel) transition piece or split ring to the ASTM SA-516 Grade 70 (carbon steel) dished head is the component of interest. The weld between the stainless steel ½ inch process pipe and the ASTM A316 plate is owned by the process pipe for these four penetrations. ONS SLRA Table 3.5.2-2 is updated to include stainless steel as a material for the ONS penetrations and include the applicable AMR NUREG-2191 (GALL-SLR) line items.

The ONS Units 1, 2, and 3 fuel transfer tube penetrations contain a DMW between the carbon steel dished head and the stainless steel fuel transfer tube. This weld is part of the fuel transfer tube and is addressed in the ONS SLRA Table 3.3.2-17, Auxiliary System – Spent Fuel Cooling System – Aging Management Evaluation under the Piping component type. The blind flange is also addressed as a part of the Piping component type. The carbon steel portions of the fuel transfer tube penetrations are evaluated in the fatigue waiver analysis described above.

Request 2:

The ONS Units 1, 2, and 3 containment liners and penetrations, not including main steam and main feedwater, are addressed through a fatigue waiver analysis that is described in response to RAI 4.6.1-1 (Attachment 12). The main steam and main feedwater penetrations are addressed in Section 4.6.3 of the ONS SLRA and in response to RAI 4.6.3-1 (Attachment 13).

The fatigue waiver analysis performed for the containment liner and cold penetrations was performed in accordance with the ASME Code, Section III, Paragraph N-415, to validate that a detailed analysis would not be warranted. The analysis concluded that the liner plate and cold penetrations meet all six criteria in the ASME Code, Section III, Subsection N-415.1, and therefore a detailed fatigue analysis is not required.

The main steam and main feedwater penetrations do have a fatigue analysis as part of their design as described in ONS SLRA Section 4.6.3. The transient cycles considered in the main steam and main feedwater penetrations analyses were projected through 80-years of operation and the count found to be adequate for the SPEO. The *Fatigue Monitoring* AMP will monitor and track these applicable fatigue transients and ensure corrective action is taken prior to exceeding fatigue design limits.

SLRA Revisions:

SLRA Section 3.5.2.2.1.5 (page 3-1310) is revised as follows:

3.5.2.2.1.5 Cumulative Fatigue Damage

Evaluation

[3.5.1-009] – Containment Liner~~, Main Feedwater and Main Steam Containment Penetration~~ **and penetrations, including mechanical, electrical, equipment, and personnel-related penetrations** / Fatigue Cracking is a TLAA, as defined in 10 CFR 54.3. The evaluation of this TLAA is addressed in Section 4.6, "Containment Liner Plate, Metal Containments and Penetration Fatigue Analysis".

[3.5.1-027] – For the ONS Units 1, 2, and 3, the containment liner plate and penetrations, including mechanical, electrical, equipment and personnel-related penetrations, have a CLB fatigue analysis. The penetrations mentioned above include the associated penetration sleeves. ONS does not have penetration bellows that are part of the containment pressure boundary. Therefore, the NUREG-2191 Item II.A3.CP-37 (SRP Item 3.5.1-027) is not applicable for ONS. The materials of construction for these components are steel, stainless steel, and dissimilar metal welds. As described in SLRA Section 4.6.1, a fatigue waiver analysis was performed on the liner plate and cold penetrations, including mechanical, electrical, equipment and personnel-related penetrations. The analysis was determined to meet all six criteria in the ASME Code, Section III, Subsection N-415.1. No detailed fatigue analysis is required for the containment liner or cold penetrations. All penetrations except main steam and main feedwater are termed as cold penetrations. The main steam and main feedwater penetrations have been termed as hot penetrations. The main steam and main feedwater penetration fatigue analysis is addressed in SLRA Section 4.6.3.

SLRA Section 3.5.2.2.1.6 (page 3-1311) is revised as follows:

3.5.2.2.1.6 Cracking Due to Stress Corrosion Cracking

Evaluation

[3.5.1-010] – ONS containment does ~~not~~ have stainless steel **components and dissimilar metal welds** ~~penetration sleeves, penetration bellows, vent line bellows, or suppression chamber shell (interior surface)~~ as part of the containment pressure boundary. Stainless steel ~~high energy~~ pipes that penetrate the containment are connected to ~~carbon~~ **stainless** steel ~~penetration sleeves~~ **split rings and welded to a carbon steel dished head** with dissimilar metal welds.

Review of plant OE has not identified stress corrosion cracking associated with these welds. The *ASME Section XI, Subsection IWE* (B2.1.28) program, and *10 CFR Part 50, Appendix J* (B2.1.31) program manages the aging of these **stainless steel components and** dissimilar metal welds.

SLRA Table 3.5.1 (page 3-1328) is revised as follows:

Table 3.5.1 Summary of Aging Management Programs for Containments, Structures and Component Supports Evaluated in Chapters II and III of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-010	Penetration sleeves; penetration bellows	Cracking due to SCC	AMP XI.S1, “ASME Section XI, Subsection IWE,” and AMP XI.S4, “10 CFR Part 50, Appendix J”	Yes (SRP-SLR Section 3.5.2.2.1.6)	~~Not applicable. ONS containment does not have stainless steel (SS) penetration sleeves, penetration bellows, vent line bellows, or suppression chamber shell (interior surface) as part of the Containment pressure boundary.~~ **Consistent with NUREG-2191.** See further evaluation in Section 3.5.2.2.1.6. ~~The associated NUREG-2191 aging items are not used.~~

SLRA Table 3.5.2-2 (pages 3-1371) is revised as follows:

Table 3.5.2-2 Containments, Structures, and Component Supports - Reactor Building - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Moisture Barrier	SP	Elastomer, Rubber and Other Similar Materials	Air – Indoor Uncontrolled (External)	Loss of Sealing	ASME Section XI, Subsection IWE (B2.1.28)	II.A3.CP-40	3.5.1- 026	A
Penetrations	PB; SS	Steel	Air – Indoor Uncontrolled (External)	~~Cracking~~	~~10 CFR 50, Appendix J (B2.1.31)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
					~~ASME Section XI, Subsection IWE (B2.1.28)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
				Cumulative Fatigue Damage	TLAA	II.A3.C-13	3.5.1- 009	A
				Loss of Material	10 CFR 50, Appendix J (B2.1.31)	II.A3.CP-36	3.5.1- 035	A, 2
					ASME Section XI, Subsection IWE (B2.1.28)	II.A3.CP-36	3.5.1- 035	A, 2
			Air – Outdoor (External)	~~Cracking~~	~~10 CFR 50, Appendix J (B2.1.31)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
					~~ASME Section XI, Subsection IWE (B2.1.28)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
				Cumulative Fatigue Damage	TLAA	II.A3.C-13	3.5.1- 009	A
				Loss of Material	10 CFR 50, Appendix J (B2.1.31)	II.A3.CP-36	3.5.1- 035	A, 2

SLRA Table 3.5.2-2 (page 3-1372) is revised as follows:

Table 3.5.2-2 Containments, Structures, and Component Supports - Reactor Building - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Penetrations	PB; SS	Steel	Air – Outdoor (External)	Loss of Material	ASME Section XI, Subsection IWE (B2.1.28)	II.A3.CP-36	3.5.1- 035	A, 2
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	III.B5.T-25	3.5.1- 089	C
	PB; SS	**Stainless Steel**	**Air – Indoor Uncontrolled (External)**	**Cracking**	**10 CFR 50, Appendix J (B2.1.31)**	**II.A3.CP-38**	**3.5.1-010**	**A**
					ASME Section XI, Subsection IWE (B2.1.28)	**II.A3.CP-38**	**3.5.1-010**	**A**
				Cumulative Fatigue Damage	**TLAA**	**II.A3.C-13**	**3.5.1-009**	**A**

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 7
RAI 3.5.2.2.1.6-1

Enclosure 1, Attachment 7

RAI 3.5.2.2.1.6-1:

Regulatory Basis:
Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section 3.5.2.2.2.1.6, associated with AMR item 3.5.1-010 related to stress corrosion cracking (SCC), states that stainless steel high energy pipes that penetrate the containment are connected to carbon steel penetration sleeves with dissimilar metal welds, and the ASME Section XI, Subsection IWE program and 10 CFR Part 50, Appendix J program manage the aging of these dissimilar metal welds. However, SLRA Section B2.1.28 states that Appendix J AMP manages the aging of these dissimilar metal welds.

SLRA Section A2.28 states, “The program includes supplemental surface or enhanced examinations to detect cracking for specific pressure-retaining components. Containment liners and penetrations were analyzed for cyclic fatigue and do not require surface examinations in addition to visual examinations to detect cracking in stainless steel and dissimilar metal welds of penetration sleeves and components that are subject to cyclic loading.”

The above statements appear to imply that supplemental surface or enhanced examinations would apply to components with dissimilar metal welds for SCC since analysis for fatigue loading does not preclude cracking due to SCC. But the ASME Section XI, Subsection IWE program and the SLRA Section 3.5.2.2.1.6 do not appear to include any enhancements for the surface or enhanced examinations and do not identify the specific pressure-retaining components on which these examinations will be performed, as stated in SLRA Section A2.28. In addition, SRP-SLR Section 3.5.3.2.1.6 guidance states that containment inservice inspection (ISI) IWE and leak rate testing may not be sufficient to detect cracks, especially for dissimilar metal welds.

SLRA Table 3.5.1 claims AMR Item 3.5.1-010 to be not applicable, and SLRA Section 3.5.2.2.1.6 states that ONS containment does not have stainless steel penetration sleeves, penetrations bellows, vent line bellows, or suppression chamber shell (interior face) as part of the containment pressure boundary. However, as stated in SLRA Section 3.5.2.2.1.6, ONS does have penetration sleeves with dissimilar metal welds that could be subject to SCC.

Issue:
It is unclear how the ASME Section XI, Subsection IWE program and 10 CFR Part 50, Appendix J program will be sufficient to manage aging effects of dissimilar metal welds without additional appropriate examinations capable of detecting cracking due to SCC. Also, there are no AMR line items in SLRA Table 3.5.2-2 to manage SCC for penetrations with dissimilar metal welds.

SLRA Section B2.1.28 ASME Section XI, Subsection IWE program and SLRA Section 3.5.2.2.1.6 do not appear to include an enhancement with regard to supplemental surface or enhanced examinations and do not identify the specific pressure-retaining components on which these examinations will be performed, as stated in SLRA Section A2.28.

AMR Item 3.5.1-010, with corresponding GALL-SLR Report AMR item II.A3.CP-38, also applies to dissimilar metal welds which do exist at ONS. It appears that the non-applicability claim in SLRA Table 3.5.1, for AMR item 3.5.1-010 is not sufficiently justified or addressed in SLRA Section 3.5.2.2.1.6.

Request:

1. Explain how the ASME Section XI, Subsection IWE program and 10 CFR Part 50, Appendix J program examination/testing methods will be sufficient to manage dissimilar metal welds without additional examinations capable of detecting cracking due to SCC.
2. Clarify the specific pressure-retaining components that will be subject to supplemental surface or enhanced examinations and the examination frequency to detect cracking and provide appropriate enhancements to the SLRA Section B2.1.28 AMP to perform these examinations. Also, state the specific enhanced visual examination method (e.g., EVT-1) that may be performed in lieu of surface examinations.
3. Justify the non-applicability claim of SLRA Table 3.5.1, item 3.5.1-010 in SLRA Section 3.5.2.2.1.6. Alternatively, provide appropriate Table 2 AMR line items for the components and material that will be managed for SCC in accordance with GALL-SLR Table 3.5-1, item 3.5.1-010.
4. Clarify the discrepancy among SLRA Section 3.5.2.2.1.6, SLRA Section B2.1.28, and SLRA Section A2.28 regarding aging management of dissimilar metal welds, and supplemental surface or enhanced examinations to detect cracking for specific pressure-retaining components.
5. Revise SLRA as necessary to be consistent with responses to the above requests.

Response to RAI 3.5.2.2.1.6-1:

Background:

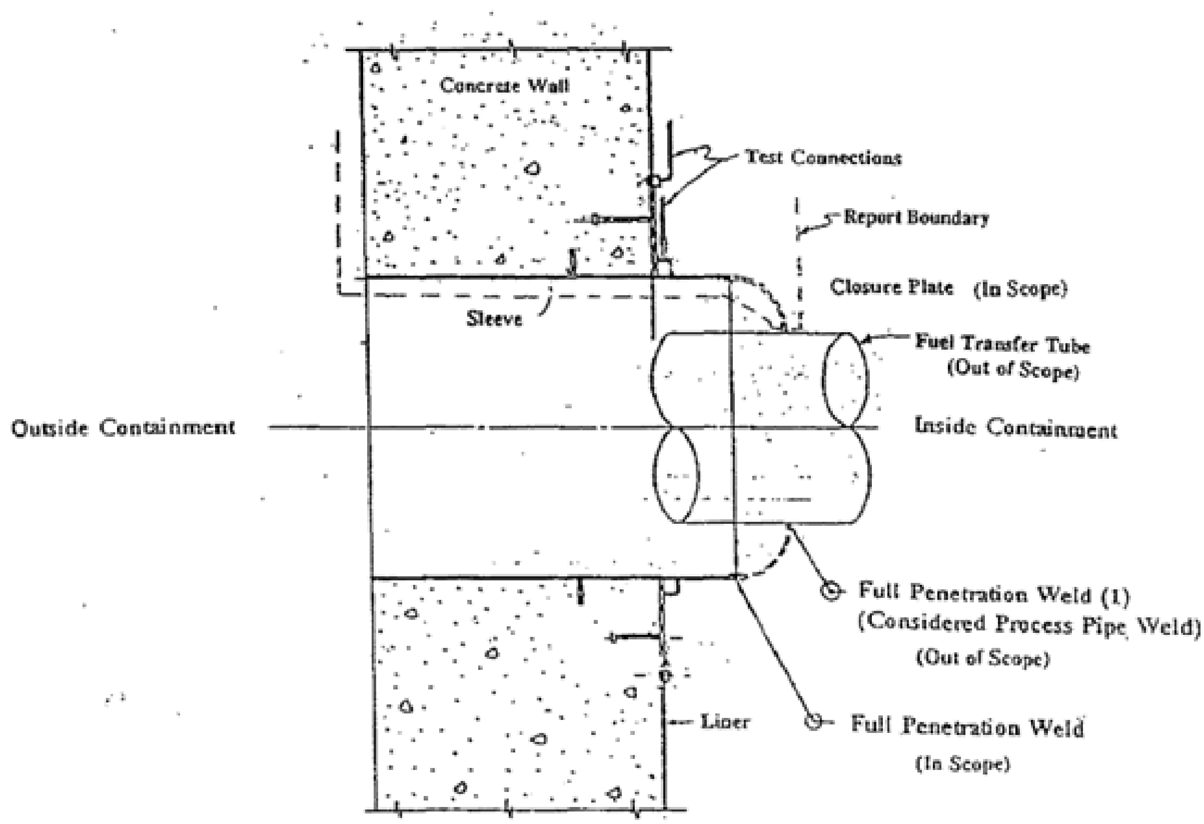
The components of interest in this response are certain dissimilar metal welds at select containment piping penetrations. Along with the containment liner, all containment penetrations are carbon steel. At a typical piping penetration, the liner is welded to a carbon steel dished head and the dished head is then welded to the exterior of the carbon steel piping to form the leak tight containment barrier. In a very limited number of cases the process piping is stainless steel. For these penetrations, a stainless steel transition piece, also called a split ring, is welded to the dished head and also to the stainless steel pipe. The stainless steel split ring and the dissimilar metal weld between the split ring and dished head are the components of interest. The weld between the stainless steel split ring and the process pipe is considered part of the process pipe and not in the scope of this response. The process pipe is included in the AMR with the appropriate mechanical systems.

The count of these dissimilar welds is very limited. Among the three units, there are dissimilar metal welds related to the containment liner and penetrations for four ½ inch NPS sample lines. ONS Unit 1 has two of these penetrations and Units 2 and 3 each have one of these configurations. The four lines are normally at the ambient temperatures of the Reactor Building (inside containment) and the Auxiliary Building (outside containment) as the lines only serve as periodic sampling lines.

The aging effect of interest is stress corrosion cracking (SCC) of stainless steel, focused on the dissimilar metal welds. For SCC of stainless steel to be an issue requires that the material be exposed to an adverse environment in the presence of a stressor such as high temperature. For these dissimilar metal welds, an adverse environment would be a concentration of chloride of sulfate contaminates, as well as, high stress levels and high temperatures. The area of containment where these welds are located does not have either of these conditions which could lead to SCC. Nevertheless, programmatic actions are in-place that will detect such aging should it occur.

Regarding the stainless steel fuel transfer tube and how it is connected to the containment liner in order to establish an essentially leak-tight barrier is represented in Figure 3.5.2.2.1.6-1. As shown in the figure, the closure between the transfer tube and the sleeve that is integrally welded to the containment liner consists of a carbon steel circular plate that is shop welded to the stainless steel fuel transfer tube and a short segment of carbon steel pipe to mate with the sleeve. In this case, the carbon-to-stainless steel weld is at the surface of the stainless tube. The design code jurisdictional boundary places ownership of the carbon to stainless steel weld with the fuel transfer tube. The fuel transfer tubes are evaluated with the spent fuel cooling system in ONS SLRA Section 2.4.2 with SCC addressed in ONS SLRA Table 3.3.2-17, Auxiliary Systems - Spent Fuel Cooling System - Aging Management Evaluation. Each unit has two fuel transfer tube penetrations.

Figure 3.5.2.2.1.6-1 – Fuel Transfer Tube Penetration



Request 1:

The ASME Section XI, Subsection IWE program and the 10 CFR Part 50, Appendix J program work together to provide techniques with which to manage detection of SCC of the limited length dissimilar metal welds associated with the containment penetrations for the four ½ inch NPS sample lines. The dissimilar metal welds between the carbon steel dished head and the stainless steel split ring at Penetrations 1 and 2 for Unit 1 and Penetration 2 for Units 2 and 3 are subject to a general visual examination once per inspection period per ASME Code Table IWE-2500-1, Examination Category E-A, Item No. E1.11. Cracking that could lead to failure of the containment integrity function can readily be detected by this visual examination. Past inspection results at these penetrations have revealed no signs of degradation, corrosion, or evidence of an environment conducive to SCC.

The 10 CFR Part 50, Appendix J program complements the ASME Section XI, Subsection IWE program by performing a Type A integrated leak rate test for containment that verifies the containment integrity function. Appendix J Type B local leak rate testing is not currently performed on these penetrations.

As described in the Response Background, the penetrations that contain stainless steel and dissimilar metal welds are not subject to environments that would contribute to SCC. These penetrations are considered cold penetrations for evaluation purposes. Therefore, the loads on these penetrations would not meet the criteria for causing SCC. The ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J AMPs are sufficient to manage cracking due to SCC for these components.

Request 2:

As described in the Response Background, the pressure-retaining component of interest is the dissimilar metal weld on four (4) ½ inch NPS stainless steel sample lines that penetrate containment. Because of the environments associated with these few welds, no supplemental surface or enhanced examinations are required. Aging management of these few locations will be managed by the ASME Section XI, Subsection IWE (B2.1.28) program and the 10 CFR Part 50, Appendix J (B2.1.31) program as discussed in Response 1 to this RAI.

Request 3:

NUREG-2192 (SRP), Table 3.5.1, ID 10 is the source for the entry in SLRA Table 3.5.1, item 3.5.1-010. The NUREG specifically focuses the components of interest to penetration sleeves and penetration bellows for which stress corrosion cracking could be an issue. Oconee has penetrations that contain stainless steel materials as described in the background of this response. Therefore, ONS SLRA Section 3.5.2.2.1.6 is revised to describe these components in the evaluation of SRP Item 3.5.1-010. ONS SLRA Table 3.5.1 is updated to include a similar discussion for SRP Item 3.5.1-010. See below for SLRA revisions.

Request 4:

SLRA Section 3.5.2.2.1.6, SLRA Section B2.1.28, and SLRA Section A2.28 have been reviewed and several portions realigned to address the stainless steel split rings and dissimilar metal welds. Addressed in these revisions are the aging management of dissimilar metal welds, management of cyclic fatigue in response to RAI 4.6.1-1 (Attachment 12) and the misstatement regarding supplemental or enhanced examinations which are not required as discussed in response to Request 2 of this RAI.

Request 5:

See below for SLRA revisions. Note that the changes to pages 3-1311, 3-1328, 3-1371, and 3-1372 are also contained in Enclosure 1, Attachment 6.

SLRA Revisions:

SLRA Section 3.5.2.2.1.6 (page 3-1311) is revised as follows:

3.5.2.2.1.6 Cracking Due to Stress Corrosion Cracking

Evaluation

[3.5.1-010] – ONS containment does ~~not~~ have stainless steel **components and dissimilar metal welds** ~~penetration sleeves, penetration bellows, vent line bellows, or suppression chamber shell (interior surface)~~ as part of the containment pressure boundary. Stainless steel ~~high energy~~ pipes that penetrate the containment are connected to ~~carbon~~ **stainless** steel ~~penetration sleeves~~ **split rings and welded to a carbon steel dish head** with dissimilar metal welds.

Review of plant OE has not identified stress corrosion cracking associated with these welds. The *ASME Section XI, Subsection IWE* (B2.1.28) program, and *10 CFR Part 50, Appendix J* (B2.1.31) program manages the aging of these **stainless steel components and** dissimilar metal welds.

SLRA Table 3.5.1 (page 3-1328) is revised as follows:

Table 3.5.1 Summary of Aging Management Programs for Containments, Structures and Component Supports Evaluated in Chapters II and III of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-010	Penetration sleeves; penetration bellows	Cracking due to SCC	AMP XI.S1, "ASME Section XI, Subsection IWE," and AMP XI.S4, "10 CFR Part 50, Appendix J"	Yes (SRP-SLR Section 3.5.2.2.1.6)	~~Not applicable. ONS containment does not have stainless steel (SS) penetration sleeves, penetration bellows, vent line bellows, or suppression chamber shell (interior surface) as part of the Containment pressure boundary.~~ **Consistent with NUREG-2191.** See further evaluation in Section 3.5.2.2.1.6. ~~The associated NUREG-2191 aging items are not used.~~

SLRA Table 3.5.2-2 (pages 3-1371) is revised as follows:

Table 3.5.2-2 Containments, Structures, and Component Supports - Reactor Building - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Moisture Barrier	SP	Elastomer, Rubber and Other Similar Materials	Air – Indoor Uncontrolled (External)	Loss of Sealing	ASME Section XI, Subsection IWE (B2.1.28)	II.A3.CP-40	3.5.1- 026	A
Penetrations	PB; SS	Steel	Air – Indoor Uncontrolled (External)	~~Cracking~~	~~10 CFR 50, Appendix J (B2.1.31)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
					~~ASME Section XI, Subsection IWE (B2.1.28)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
				Cumulative Fatigue Damage	TLAA	II.A3.C-13	3.5.1- 009	A
				Loss of Material	10 CFR 50, Appendix J (B2.1.31)	II.A3.CP-36	3.5.1- 035	A, 2
					ASME Section XI, Subsection IWE (B2.1.28)	II.A3.CP-36	3.5.1- 035	A, 2
			Air – Outdoor (External)	~~Cracking~~	~~10 CFR 50, Appendix J (B2.1.31)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
					~~ASME Section XI, Subsection IWE (B2.1.28)~~	~~II.A3.CP-38~~	~~3.5.1- 010~~	~~A~~
				Cumulative Fatigue Damage	TLAA	II.A3.C-13	3.5.1- 009	A
				Loss of Material	10 CFR 50, Appendix J (B2.1.31)	II.A3.CP-36	3.5.1- 035	A, 2

SLRA Table 3.5.2-2 (page 3-1372) is revised as follows:

Table 3.5.2-2 Containments, Structures, and Component Supports - Reactor Building - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Penetrations	PB; SS	Steel	Air – Outdoor (External)	Loss of Material	ASME Section XI, Subsection IWE (B2.1.28)	II.A3.CP-36	3.5.1- 035	A, 2
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	III.B5.T-25	3.5.1- 089	C
	PB; SS	**Stainless Steel**	**Air – Indoor Uncontrolled (External)**	**Cracking**	**10 CFR 50, Appendix J (B2.1.31)**	**II.A3.CP-38**	**3.5.1-010**	**A**
					ASME Section XI, Subsection IWE (B2.1.28)	**II.A3.CP-38**	**3.5.1-010**	**A**
				Cumulative Fatigue Damage	**TLAA**	**II.A3.C-13**	**3.5.1-009**	**A**

SLRA Appendix A2.28 (page A-30) is revised as follows:

A2.28 ASME XI, Subsection IWE

Program Description

The *ASME Section XI, Subsection IWE* AMP is an existing condition monitoring program that manages cracking, loss of material, loss of sealing, loss of preload, and loss of leak tightness. This program is in accordance with ASME Section XI, Subsection IWE, consistent with 10 CFR 50.55a *"Codes and Standards*," with supplemental recommendations. The *ASME Section XI, Subsection IWE* program includes periodic visual, surface, and volumetric examinations, where applicable, of the metallic pressure-retaining components of the concrete containment for signs of degradation, damage, irregularities including discernible liner plate bulges, and for coated areas, distress that might be indicative of degradation of the underlying metal shell or liner, and corrective actions. Acceptability of inaccessible areas of the concrete containment steel liner is evaluated when conditions found in accessible areas, indicate the presence of, or could result in, flaws or degradation in inaccessible areas.

~~The program includes supplemental surface or enhanced examinations to detect cracking for specific pressure-retaining components.~~ Containment liners and penetrations were analyzed for cyclic fatigue and**, except for the main steam and feedwater penetrations, were exempted from further fatigue consideration by fatigue waiver.** ~~do not require surface examinations in addition to visual examinations to detect cracking in stainless steel and dissimilar metal welds of penetration sleeves and components that are subject to cyclic loading.~~ **For the containment liner,** ~~A~~ **a** one-time volumetric examination of metal liner surfaces that are inaccessible from one side will be performed if triggered by plant-specific OE. This supplemental volumetric examination consists of a sample of one-foot square locations that include both randomly selected and focused areas most likely to experience degradation based on operating experience and/or relevant considerations such as environment. Inspection results will be compared with prior recorded results in acceptance of components for continued service.

In conformance with 10 CFR 50.55a(g)(4)(ii), the ~~*Containment Inservice Inspection*~~ ***ASME Section XI, Subsection IWE*** AMP will be updated during each successive 120 month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified 12 months before the start of the inspection interval.

SLRA Appendix B2.1.28 (page B-198 and B-199) is revised as follows:

B2.1.28 ASME XI, Subsection IWE

Program Description

There are no stainless steel penetration bellows installed as part of the containment pressure boundary. ~~Stainless steel high energy pipes that penetrate the containment are connected to carbon steel penetration sleeves with dissimilar metal welds.~~ **Stainless steel pipes that penetrate the containment are welded to a stainless steel split ring that is welded to a carbon steel dished head. The dished head is welded to the carbon steel containment liner. A dissimilar metal weld exists between the carbon steel dished head and stainless steel split ring.** Plant OE has not identified any stress corrosion cracking associated with these welds. The ***ASME Section XI, Subsection IWE along with*** *10 CFR Part 50, Appendix J* AMP manages the aging of these dissimilar metal welds. The containment steel liner and penetrations were analyzed for cyclic fatigue and**, except for the main steam and feedwater penetrations, were exempted from further fatigue consideration by fatigue waiver** ~~do not require surface examinations in addition to visual examinations. This evaluation is a TLAA and the projected number of cycles is less than the design cycles for 80 years. Therefore, cracking due to fatigue is managed as a TLAA.~~ **For the containment liner,** ~~A~~ **a** one-time volumetric examination of metal liner surfaces that are inaccessible from one side will be performed if triggered by plant-specific OE. This supplemental volumetric examination consists of a sample of one-foot square locations that include both randomly selected and focused areas most likely to experience degradation based on OE and/or relevant considerations such as environment.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 8
RAI 2.1.30-1

Enclosure 1, Attachment 8

RAI B2.1.30-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(3) requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
The "preventive actions" program element of GALL-SLR AMP XI.S3 states: "Operating experience and laboratory examinations show that the use of molybdenum disulfide (MoS_2) as a lubricant is a potential contributor to stress corrosion cracking (SCC), especially when applied to high-strength bolting. Thus, molybdenum disulfide and other lubricants containing sulfur should not be used."

The "parameters monitored or inspected" program element of GALL-SLR AMP XI.S3 recommends that high strength bolting (actual measured yield strength greater than or equal to 150 ksi) in sizes greater than 1 inch nominal diameter should be monitored for SCC.

SLRA Section B2.1.30 states that the ONS ASME Section XI, Subsection IWF AMP is an existing program with enhancements that will be consistent with the ten elements of the GALL-SLR AMP XI.S3, ASME Section XI, Subsection IWF. The SLRA AMP does not take exception to any program element including "preventive actions."

SRP-SLR Section 1.2.1 states, in part: "If a GALL-SLR Report AMP is selected to manage aging, the applicant may take one or more exceptions to specific GALL-SLR Report AMP program elements. Exceptions are portions of the GALL-SLR Report AMP that the applicant does not intend to implement, which the staff will review on a case-by-case basis. Any deviation or exception to the GALL-SLR Report AMP should be described and justified."

SLR-ONS-AMPR-XI.S3, "ASME Section XI, Subsection IWF AMP Evaluation Report, ONS Units 1, 2 and 3," Revision 2, paragraph 4.2.a under Preventive Actions, on page 10 of 41 states: "The use of molybdenum disulfide as a lubricant on bolting is prohibited at Oconee as specified in the Power Chemistry Materials Guide (PCMG). Station procedures specify the use of thread lubricants, such a Loctite N-5000, that have low levels of halogens and sulfur to minimize the potential for stress corrosion cracking. (Reference: AD-EN-ALL-0045, PCMG Manual Section 1.10.7)."

Section 4.2.c of SLR-ONS-AMPR-XI.S3, Revision 2, on page 10/11 of 41 states in part:

> The ONS ASME Section XI, Subsection IWF AMP consists of ASTM A490 bolting for the reactor vessel anchor studs and the replacement steam generator anchor studs. Bolting material selection is governed through the design change procedures and design specifications for the plant. The use of lubricants and sealants is controlled by the Nuclear Chemical Control process through the PCMG and by station maintenance procedures. Station procedures specify the use of thread lubricants and sealants…

Section 1.10.7 of the Power Chemistry Material Guide (PCMG) Program, Revision 31, states in part: "The PCMG program controls chemical content of consumables to preclude initiation of stress corrosion cracking of austenitic stainless steel, and prescribes contaminant limits for thread lubricants and sealants, which are approved for use. NOTE: Molybdenum Disulfide containing materials are restricted from use on bolting materials."

Since high-strength bolting is used and may continue to be used at ONS, and molybdenum disulfide (MoS_2) or other lubricants containing sulfur may have been used at ONS, as recommended in the GALL-SLR, SLRA Section B2.1.30 includes enhancement 5 (SLR Commitment # 30(5) in SLRA Table A6.0-1) to the "detection of aging effects" program element to perform volumetric examinations, once in each 10-year period during the SPEO, on a representative sample of high-strength bolting greater than one inch nominal diameter to detect cracking for NSSS [Nuclear Steam Supply System] component supports.

Issue:

1. While the PCMG through a footnote (as above) appears to only restrict (and not prohibit as claimed in the AMP evaluation report SLR-ONS-AMPR-XI.S3) use of molybdenum disulfide (MoS_2) containing materials at ONS, the possibility appears to exist that MoS_2 lubricant may have been used at ONS in the past and may continue to be used in the future. Also, the PCMG does not mention or prohibit use of other lubricants containing sulfur. Further, during the audit, the staff noted that document ER-CHM-00005 "Nuclear Chemical [Approved] List" includes Molykote BR-2, Molykote 3452, Molykote (R) Z Powder as approved lubricants at ONS, which contain MoS_2 as an ingredient. The staff also noted that, in response to audit breakout question 5(a), the applicant stated that it cannot be confirmed whether molybdenum disulfide has been used on bolting within the IWF program at ONS in the past. Based on the above, the "preventive actions" element of the SLRA AMP does not appear to have adequately addressed the use at ONS of molybdenum disulfide and other lubricants using sulfur which are identified in the GALL-SLR Report as potential contributors to stress corrosion cracking (SCC) in high-strength bolting, and the staff could not verify the consistency claim.

   The "preventive actions" program element of the SLRA AMP does not include an enhancement corresponding to the GALL-SLR AMP XI.S3 recommendation that MoS_2 and other lubricants containing sulfur should not be used, and therefore it does not appear to be consistent with the corresponding program element of the GALL-SLR XI.S3 AMP in that regard. Further, the SLRA AMP does not appear to take or justify an exception to the "preventive actions" program element with regard to use of molybdenum disulfide and other lubricants containing sulfur.

2. The staff notes that VT-3 visual inspections of the currently implemented ASME Section XI, Subsection IWF program may not detect SCC aging effect, prior to loss of intended function, in high-strength bolting from the use of molybdenum disulfide or other lubricants containing sulfur. The staff also noted that adequate preventive measures may not be currently in place and, for the subsequent period of extended operation (SPEO), may not inhibit this aging effect. Hence, it is possible that this aging effect may be present prior to entering and during the SPEO due to possible past use and the possibility of continued use of high-strength bolts with lubricants containing sulfur, including molybdenum disulfide. Hence, it is possible that such an aging effect may remain undetected until SLRA AMP B2.1.30 volumetric examinations of a sample of susceptible high-strength are performed during the SPEO, which could be as much towards the end of the first 10-year interval in the SPEO. The SRP-SLR Branch Technical Position RLSB-1 Section A.1.2.3.4, however, states that "detection of aging effects" should occur before there is a loss of the SC-intended function(s). Therefore, for the period of time between the start of the SPEO and when the volumetric examinations are performed, which

could be up to the end of the first 10-year interval in to the SPEO, it is not clear how the aging effect of cracking due to SCC will be detected and managed prior to a loss of intended function.

3. If ONS uses MoS_2 or other lubricants containing sulfur coated high-strength bolts susceptible to SCC into the SPEO, there is the potential to increase the population of installed high-strength bolts (i.e., install additional high-strength bolts as replacement bolting) susceptible to SCC. It is not clear how the sample for volumetric examination representing the entire population of high-strength bolts will be established. It is also not clear how the program will assess the sample size and scope to ensure that it continues to monitor suspect high-strength bolts coated with sulfur based lubricants, especially those that have used/are using MoS_2.

Request:

1. Provide an enhancement and corresponding SLR Commitment that would make the “preventive actions” program element consistent, as claimed in the SLRA, with the GALL-SLR AMP XI.S3 regarding the recommendation that molybdenum disulfide and other lubricants containing sulfur should not be used in order to prevent SCC in high strength bolting. Alternatively, describe and justify an exception taken to the “preventive actions” program element of the GALL-SLR AMP XI.S3 regarding the recommendation that molybdenum disulfide and other lubricants containing sulfur should not be used to prevent SCC in high-strength bolting.

2. Since the first volumetric examinations per enhancement 5 (SLRA Commitment 30(5)) to the SLRA B2.1.30 AMP are planned for some time into the SPEO (could be as much as towards the end of the first 10-year interval in the SPEO), provide information on whether and how the timing of implementation of enhancement 5 volumetric examinations would assure that cracking due to SCC will be detected for the population of existing high strength bolts in a manner such that this aging effect can be managed prior to loss of intended function, consistent with SRP-SLR Branch Technical Position RLSB-1 Section A.1.2.3.4, during the SPEO.

3. Discuss how the “parameters monitored or inspected” program element will identify and assess the adequacy of the representative high-strength bolting sample inspected for cracking due to SCC for existing and/or when additional susceptible high-strength bolts are installed.

4. Update applicable portions of the SLRA, as necessary, consistent with responses to the above requests.

Response to RAI B2.1.30-1:

Request 1:

Existing procedural guidance exists to ensure that products containing molybdenum disulfide (MoS_2) are restricted from use as bolting thread lubricants. The ASME Section XI, Subsection IWF is enhanced to specifically document the prohibition of the use of bolting thread lubricants containing MoS_2 and other lubricants containing sulfur that could contribute to SCC in the fleet work planning procedures.

The nuclear chemical control procedure governs the use of all chemicals at Oconee, including bolting thread lubricants. The purpose of the nuclear chemical control process is to provide consistent practices for the nuclear fleet regarding all aspects of chemical control, including defining the method in which chemicals are approved for use on site and ensuring that only chemicals that have been approved for use are procured. In addition, the nuclear chemical control process drives implementation of Power Chemistry Material Guide (PCMG) requirements.

The PCMG provides additional requirements and limitations for materials to be used in contact with the safety related and non-safety related plant systems, structures, and components at Oconee. These requirements and limitations are based on the potential for initiation or promotion of degradation due to the chemical effect that certain substances have on the materials of construction of plant equipment. The PCMG establishes chemical impurity limits and provides for testing and categorization (approval categories) of chemicals used in the plant. The PCMG is implemented through an online database that is available to plant personnel and captures the results of the analyses and categorization of the various chemical materials used in the plant. The use of molybdenum disulfide as a thread lubricant has been shown to be a potential contributor to stress corrosion cracking. Therefore, the use of molybdenum disulfide as a thread lubricant on bolting is prohibited at Oconee as specified in the PCMG and, as such, a product containing MoS_2 is not, and would not be, approved for use as a bolting thread lubricant in the plant.

All chemicals provided for use on station equipment are issued to maintenance personnel through approved work orders. In accordance with the fleet work order planning procedure, work planners are required to ensure that chemicals issued for use in the field meet the requirements of the PCMG. Therefore, the work planning process ensures that all thread lubricants issued to maintenance personnel comply with PCMG requirements. While Molykote BR-2, Molykote 3452, Molykote (R) Z Powder are approved for use at Oconee, these are not used as a thread lubricant. The Molykote BR-2 is used as lubricant for reactor trip breakers. Molykote 3452 does not contain MoS_2 and is used in switchyard power breakers. Molykote ® Z Powder is lubricant for rotating parts.

To ensure that chemicals are not inadvertently used in unapproved applications, each chemical product used on site is labeled upon receipt to clearly display the PCMG approval category. In accordance with the PCMG, a product containing MoS_2 would not be labeled for use as a bolting thread lubricant in the plant. All site personnel are required to understand and comply with the requirements of the nuclear chemical control process and PCMG. Managers and supervisors are required to ensure that employees whose work activities include the use, storage and disposal of chemicals, comply with the requirements of the nuclear chemical control process and PCMG. Finally, technicians at Oconee receive instruction on the PCMG as part of the initial maintenance training program. The lesson familiarizes the student with information available in the PCMG and its use. The training stresses how improper application of chemical materials could have detrimental effects on plant components.

Based on the above, existing procedural guidance exists to ensure that products containing molybdenum disulfide (MoS_2) are restricted from use as bolting thread lubricants. The ASME Section XI, Subsection IWF is enhanced to specifically document the prohibition of the use of bolting thread lubricants containing MoS_2 and other lubricants containing sulfur that could contribute to SCC in the fleet work planning procedures.

Request 2:

The use of MoS_2 as a thread lubricant is currently restricted by the PCMG. An enhancement is being added to the ASME Section XI, Subsection IWF to specifically document the prohibition of the use of bolting thread lubricants containing MoS_2. ONS ASME Section XI, Subsection IWF AMP currently monitors bolting for cracking using visual inspections. Laboratory tests have shown aqueous environments release H2S, which causes SCC. ONS ASME Section XI, Subsection IWF supports use

ASTM A490 bolting greater than 1 inch nominal diameter for the reactor vessel anchor studs and the replacement steam generator anchor studs. The ONS ASME Section XI, Subsection IWF supports use ASTM A325 bolting for the pressurizer support bolts and reactor coolant pump motor supports. These bolts are not in locations that are prone to high moisture environments. OE exists for SCC for the Unit 3 steam generator primary manway, but plant specific operating experience has not identified SCC for structural high strength bolts used at ONS. Based on the plant operating experience, existing restriction, environment and enhancement for prohibition of the use of bolting thread lubricants containing MoS_2, there is reasonable assurance that performing the first volumetric inspection during the first interval of the SPEO will detect any cracking due to SCC prior to a loss of intended function.

Request 3:

The “parameters monitored or inspected” program element high-strength bolting sample shall include the bolting that is most susceptible to age-related degradation based on time in service and environment. Consistent with the enhancement provided in Response #1, MoS_2 is explicitly prohibited at ONS prior to the start of the SPEO. This ensures that any high strength bolting installed as new or replacement bolting at ONS would not be exposed to this potential contributor to SCC and would not be representative of the condition of the original population. As such, the original population would represent the most susceptible locations and adequacy of the sample size will be assured.

Request 4:

The SLRA is updated as shown in the supplements below.

SLRA Revisions:

SLRA Appendix A2.30 (page A-33) is revised as follows:

Enhancements

The *ASME Section XI, Subsection IWF* AMP will be enhanced to:

7. **Revise the fleet work planning procedure to specifically prohibit the use of products containing molybdenum disulfide (MoS_2) and other lubricants containing sulfur that could contribute to SCC as bolting thread lubricants.**

Table A6.0-1 (page A-97) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
30	*ASME Section XI, Subsection IWF* program	**7. Revise the fleet work planning procedure to specifically prohibit the use of products containing molybdenum disulfide (MoS_2) and other lubricants containing sulfur that could contribute to SCC as bolting thread lubricants.**	B2.1.30	Program enhancements for SLR will be implemented and a one-time inspection of an additional 5% of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports is conducted within 5 years prior to the SPEO, and is to be completed prior to the SPEO. Other enhancements are completed 6 months prior to the SPEO or no later than the last refueling outage prior to the SPEO.

SLRA Appendix B2.1.30 (page B-211) is revised as follows:

Enhancements:

7. Revise the fleet work planning procedure to specifically prohibit the use of products containing molybdenum disulfide (MoS_2) and other lubricants containing sulfur that could contribute to SCC as bolting thread lubricants. (Element 2)

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 9
RAI 3.5.2.2.1.2-1

Enclosure 1, Attachment 9

RAI 3.5.2.2.1.2-1:

Regulatory Basis:
Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in the SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section 3.5.2.2.1.2 states that the only high temperature piping penetrating the reactor building shell are the main steam lines and feedwater lines. The main steam penetrations (shown in USFAR Figure 3-20) are designed with cooling fans and stacks. SLRA Section 3.5.2.2.1.2 also states that a review of ONS operating experience (OE) reflects that localized concrete temperatures at the main steam penetrations have marginally exceeded 200°F. Volumetric non-destructive concrete testing was performed to address the exposure to elevated temperature concern. This testing determined that there was no adverse impact to the concrete strength due to the marginally higher temperatures in these areas.

SLRA Section B 2.1.29, operating experience No. 3 states, "In December of 2012, while performing temperature readings at the Unit 1 main steam penetration inside of the enclosed penetration area, it was discovered that the maximum inner surface of the concrete around the piping at the penetration was 227°F."

It appears that the main steam penetrations rely on cooling fans and stacks to maintain acceptable temperatures. However, the staff found that, in the SLRA, cooling fans and stacks are not in the scope of subsequent license renewal, and they are not subject to aging management review.

During the audit, the staff also found that the outside main steam penetrations of the Units 2 and 3 have experienced high concrete temperatures indicated in the document "Reactor Building Concrete Subject to Elevated Temperatures," file no. OSC-10898, Revision 0, which concluded that the temperature history of the concrete around the Units 2 and 3 main steam penetrations has caused no significant reduction in concrete strength based on Olson Engineering, Inc.'s assessment of Unit 1 penetration #28.

Issue:
It is unclear how localized concrete temperatures at the main steam penetrations can be adequately maintained without aging management of the cooling fans and stacks and concrete temperature monitoring at the main steam penetrations to identify a problem with the cooling fans and stacks.

It is unclear what the elevated concrete temperatures at the Units 2 and 3 main steam penetrations were, and how it was determined that the conclusions from the Unit 1 assessment were applicable to the elevated concrete temperatures at the Units 2 and 3 main steam penetrations.

Request:

1. Evaluate whether cooling fans and stacks should be in the scope of subsequent license renewal and subject to aging management review.
   a. If cooling fans and stacks are within the scope of subsequent license renewal, explain how aging management will be accomplished (i.e., identify an appropriate AMP and enhancements, provide acceptance criteria with basis, summarize plant-specific evaluations and corrective actions, and develop AMR line items etc.)
   b. If cooling fans and stacks are not within the scope of subsequent license renewal, explain how localized concrete temperatures at the main steam penetrations will be adequately maintained, and provide the technical basis (I.e., tests and/or calculations) to justify the higher temperatures if the localized concrete temperatures are exceeded.
2. Describe high concrete temperature OE at the Units 2 and 3 main steam penetrations. Explain how it was determined that the conclusions of the Unit 1 assessment apply to the elevated concrete temperatures at the Units 2 and 3 main steam penetrations.
3. Based on the responses to the above requests, update the SLRA accordingly.

Response to RAI 3.5.2.2.1.2-1:

Request 1:

As discussed in Section 2.1.3 of the Oconee Subsequent License Renewal Application, subsequent license renewal (SLR) scoping was performed in accordance with guidance of NEI 17-01, including the guidance of NEI 95-10 Rev. 6, which is incorporated by reference in NEI 17-01. In accordance with NEI 95-10, equipment used in the establishment of initial conditions is generically excluded from the scope of license renewal unless that equipment meets the criteria of 10 CFR 54.4. Equipment that is used in the establishment of initial conditions includes non-safety related equipment, augmented with a suitable surveillance or monitoring program, to maintain safety-related equipment or plant conditions within limits consistent with event assumptions. As discussed in the Statement of Considerations (SOC) to the license renewal rule (60FR22461), equipment used to establish initial conditions are generically excluded from the scope of license renewal as the Commission concluded that current activities for such systems, structures, and components, including licensee programs and the regulatory process, are sufficient and that no additional evaluation is necessary for license renewal.

The function of the main steam penetration fans is to maintain the temperature of concrete within limits during normal operation. The fans are required to be operating when the main steam lines are above 150°F. Since the fans are supplied with non QA-1 power, they are not available during a design basis event.

The main steam penetration fans do not meet the criteria of 54.4:

- These fans are not relied upon to mitigate the consequences of a design basis event (i.e., as defined by the plant CLB) and do not meet the (a)(1) scoping criteria.
- These fans are not relied upon to demonstrate compliance with regulations for the Fire Protection (50.48), Environmental Qualification (50.49), Pressurized Thermal Shock (50.61), Anticipated Transient without Scram (50.62), or Station Blackout (50.63) and do not meet the (a)(3) scoping criteria.
- The failure of the fans will not prevent satisfactory accomplishment of a function described in 10 CFR 54(a)(1)(i), (ii) or (iii) and do not meet the 10 CFR 54.4(a)(2) criteria.

Therefore, the main steam penetration fans are not within the scope of subsequent license renewal.

The localized concrete temperature at the main steam penetration is adequately maintained through operating procedures. Each main steam penetration is supplied with two cooling fans. When cooling is required, one fan is normally operating while the second fan is a backup. The status of the main steam penetration fans is monitored in the Control Room. A fan failure alarm is initiated in the Control Room on the following conditions: 1) a loss of power to the fans, 2) low air flow in the fans' discharge line, or 3) when the standby fan fails to start. The immediate corrective actions to a fan failure alarm are to verify the standby fan is operating. Corrective actions are initiated to repair the inoperable fan. If both fans are lost, an engineering evaluation is required. If both fans on a penetration are determined to be inoperable, a cold shutdown is initiated within 48 hours.

The ASME Code, Section III, Division 2, Subsection CC, "Concrete Reactor Vessels and Containments", in Paragraph CC-3440(a) states that concrete in local areas, such as around penetrations, is allowed to have increased temperatures not to exceed 200°F. For accidents or other short term periods, paragraph CC-3440(b) allows concrete temperatures to not exceed 350°F. Higher temperatures are allowed by Paragraph CC-3440(c) if tests evaluating the reduction in concrete strength are provided and any reduced strength is applied to design allowables.

A vendor was hired to perform nondestructive testing of the exterior Unit 1 reactor building concrete wall based on the elevated temperatures found at the main steam penetration. The data collected by the vendor demonstrated relative concrete soundness around the penetration relative to concrete soundness remote to the penetration. The test results indicate that the majority (90.2%) of test locations are considered sound with "Excellent" surface wave velocities. The remaining 9.8% of test locations, which is comprised of "Good" surface wave velocities with one "Fair" surface wave velocity, were approximately evenly distributed above the penetration where the heat from the Main Steam pipe would be greatest and below the penetration where the heat input from the Main Steam pipe would be minimal. The conclusion is that the temperature history of the concrete around the penetration has caused no significant reduction in concrete strength.

The tests found no reduction of strength due to the elevated temperatures. Therefore, using the precedent provided in Paragraph CC-3440 (c), there is no impact to the design allowables. As a result, the operating temperatures at the Main Steam penetration that exceed the values provided in Paragraph CC-3440 (a) are acceptable.

Request 2:

The outside Main Steam penetrations of Units 2 and 3 have the same configuration and conditions as that of Unit 1. It is therefore conservatively assumed that Units 2 and 3 have experienced elevated concrete temperatures as well. All 3 units were constructed using the same plant specific specification for concrete for the reactor building. The specification provides guidance for acceptable concrete design strength, slump, aggregate, cement, placing, temperature and curing among other requirements. This specification provided assurance that the Reactor Building concrete for all 3 units are representative of each other. In addition, the ASME Section XI IWL program performs inspections of the Reactor Building concrete. Elevated temperatures can cause surface scaling and cracking. There is no operating experience at Units 2 and 3 that found signs of degradation at the penetration. Vendor's assessment of Unit 1 Main Steam Penetration demonstrated relative concrete soundness around the penetration relative to concrete soundness remote to the penetration. The test results indicate that the majority (90.2%) of test locations are considered sound with "Excellent" surface wave velocities. The remaining 9.8% of test locations, which is comprised of "Good" surface wave velocities with one "Fair" surface wave velocity, were approximately evenly distributed above the penetration where the heat from the Main Steam pipe would be greatest and below the penetration where the heat input from the Main Steam pipe would be minimal. A visual inspection was performed at the penetration and the areas of minor spalling were determined to not be a result of elevated temperature. The concrete was also sounded with a small ball-peen hammer and a standard flat-head screwdriver to assess if the surface hardness of the concrete had degraded or become brittle relative to areas away from the penetration. These sounding techniques showed no indication of detectable concrete degradation. The Units 2 and 3 concrete design is the same as Unit 1 and the main steam operations of the three units is similar with Unit 1 operating the longest; therefore, the Unit 1 main steam penetration concrete has received the most thermal exposure. Since the concrete at the Unit 1 penetrations was found acceptable it was concluded that the temperature history of the concrete around the Units 2 and 3 Main Steam penetrations has caused no significant reduction in concrete strength.

Request 3:

There are no updates to the SLRA required.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 10
RAI B4.1-1

Enclosure 1, Attachment 10

RAI B4.1-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Background:
Section B4.1, "Secondary Shield Wall (SSW) Tendon Surveillance," of the SLRA starts, "the program manages for loss of material, cracking, and loss of tendon prestress by conducting visual inspections and tendon liftoff tests in accordance with station procedures (MP/A1400/021, Reference 017, "Tendon - SSW – Surveillance"). These are performed on three randomly selected horizontal tendons every other outage." The SLRA further states, "the primary strength of the removable sections of the wall is provided by horizontal tendons as well as conventional reinforcing bars in each panel."

Based on its review of drawings during the audit, the staff noted that there are three types of tendon groups in the SSW, namely: vertical, horizontal (lower, middle, and upper) and diagonal ties. These tendon groups help resist the pressure and jet loads resulting from postulated pipe ruptures. In addition, the diagonal tie tendons provide a critical boundary support function for the entire SSW structure to the permanent shield wall as shown in ONS Plant Drawing, No. O-0070-A, Revision 14, "Reactor Building SSWs North Elevation Unit 1, South Elevation Units 2 & 3.".

Issue:
It is not clear why the SSW tendon inspections are limited to the horizontal tendons only.

Request:
Provide justification for why vertical and diagonal tie tendons are not considered for visual inspections and tendon liftoff tests in accordance with station procedures. Include an explanation of how aging management is conducted on the vertical and diagonal tendons or explain why aging management is not necessary for these tendons.

Response to RAI B4.1-1:

For Oconee SLR, based on the different functions of the three types of secondary shield wall tendons, horizontal, vertical, and diagonal, Duke agrees that a program enhancement to include a sample from the vertical and diagonal tendons is warranted. Duke also agrees that several program refinements would better assure program quality during the SPEO.

The SSW Tendon Surveillance program is enhanced to perform secondary shield wall tendon lift off testing on each of the three units for one (1) horizontal, one (1) diagonal, and one (1) vertical tendon. They will be lifted or pulled from one end every second refueling outage or every 48 months. These three tendons, per unit, will be selected based on the existing lift off data that is available as well as practical access limitations. Following initial sampling, these three tendons will become "common" tendons for lift off testing. On a practical level, this enhancement will ensure the implementation of tendon testing is more predictable during a refueling outage and interferences are minimized. On a programmatic level, the common tendon concept for the lift off testing allows for trending the prestressing losses and projecting the tendon prestress losses through the next two operating cycles, i.e. 48 months. A visual inspection will also be performed on both ends of the common tendons every other refueling outage, during the same outage as the lift off tests.

Additionally, on alternate intervals to the lift off tests, the program is enhanced to perform visual inspections on tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, one (1) middle horizontal, one (1) lower horizontal, one (1) diagonal, and one (1) vertical tendon set, per unit, every second refueling outage or 48 months. Enhancing the program to include this general visual inspection will also ensure that a complete approach to aging management is done for these secondary shield wall tendon components.

This set of enhancements has been captured in the revisions below.

SLRA Revisions:

SLRA Appendix A5.1 (page A-72) is revised as follows:

A5.1 Secondary Shield Wall Tendon Surveillance

Program Description

The *Secondary Shield Wall Tendon Surveillance* AMP is an existing condition monitoring program that manages aging effects associated with the tendons and tendon anchorage in the reactor building secondary shield wall. The secondary shield wall tendon system assures the structural adequacy of the secondary concrete shield wall, which provides structural support, shelter and protection to safety related SSCs. There are no preventive or mitigative actions associated with this program. The program manages for loss of material, cracking, and loss of tendon prestress by conducting visual inspections and tendon lift-off tests in accordance with station procedures. **Lift-off testing is** ~~These are~~ performed on three ~~randomly selected horizontal~~ tendons**, one (1) horizontal, one (1) diagonal, and one (1) vertical,** every other **refueling** outage. **These three tendons, per unit, will be selected based on the existing lift-off data and practical access limitations. They will be the common tendons for lift-off testing. Prestressing forces will be trended and projected through the next two operating cycles, i.e. every 48 months. Visual inspections will also be performed on both ends of the common tendons on the same frequency as the lift off tests.**

Visual inspections will be also performed on tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, one (1) middle horizontal, one (1) lower horizontal, one (1) diagonal, and one (1) vertical, per unit, every other refueling outage, i.e. every 48 months, occurring on alternate outages from the lift off tests. Acceptance criteria outlined in station specifications and procedures ensures appropriate corrective actions are taken based on the observed and/or measured conditions. If an inspection identifies a degraded condition associated with a tendon or the tendon anchorage, the corrective action program is utilized to facilitate repair or replacement activities.

Enhancements

The *Secondary Shield Wall Tendon Surveillance* AMP will be enhanced to:

1. **Perform lift-off testing on each of the three units on one (1) horizontal, one (1) diagonal, and one (1) vertical tendon, pulled from one end, every second refueling outage, i.e. every 48 months. These three tendons, per unit, will be selected based on the existing lift-off data that is available as well as practical access limitations, and they will be the common tendons for lift off testing.**

2. Enhance station procedures to include a review of previous visual inspection and lift-off data results for the tendons selected for inspection. **Revise station procedures to use the common tendon lift-off data to trend the prestressing forces and to project tendon prestress losses through the next two operating cycles, I.e. every 48 months.**

3. Perform a visual inspection on both ends of the common tendons every other refueling outage, i.e. 48 months, during the same outage as the lift off tests.

4. Perform visual inspections on tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, (1) middle horizontal, (1) lower horizontal, (1) diagonal, and (1) vertical, per unit, every second refueling outage, i.e. every 48 months, occurring on alternate outages from the lift-off tests.

SLRA Appendix Table A6.0-1 (page A-115) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
48	*Secondary Shield Wall Tendon Surveillance* program	1. Perform lift-off testing on each of the three units on one (1) horizontal, one (1) diagonal, and one (1) vertical tendon, pulled from one end, every second refueling outage, i.e. every 48 months. These three tendons, per unit, will be selected based on the existing lift-off data that is available as well as practical access limitations, and they will be the common tendons for lift off testing. 2. Enhance station procedures to include a review of previous visual inspection and lift-off data results for the tendons selected for inspection. Revise station procedures to use the common tendon lift-off data to trend the prestressing forces and to project tendon prestress losses through the next two operating cycles, I.e. every 48 months. 3. Perform a visual inspection on both ends of the common tendons every other refueling outage, i.e. 48 months, during the same outage as the lift off tests. 4. Perform visual inspections on tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, (1) middle horizontal, (1) lower horizontal, (1) diagonal, and (1) vertical, per unit, every second refueling outage, i.e. every 48 months, occurring on alternate outages from the lift-off tests.	B4.1	Program enhancements for SLR will be implemented no later than 6 months prior to the SPEO, or no later than the last refueling outage prior to the SPEO.

SLRA Appendix B4.1 (page B-307) is revised as follows:

B4.1 Secondary Shield Wall Tendon Surveillance

Program Description

The *Secondary Shield Wall Tendon Surveillance* AMP is an existing condition monitoring program that manages aging effects associated with the tendons and tendon anchorage in the reactor building secondary shield wall. The secondary shield wall tendon system assures the structural adequacy of the secondary concrete shield wall, which provides structural support, shelter and protection to safety related SSCs. There are no preventive or mitigative actions associated with this program. The program manages for loss of material, cracking, and loss of tendon prestress by conducting visual inspections and tendon lift-off tests in accordance with station procedures. **Lift-off testing is** ~~These are~~ performed on three ~~randomly selected horizontal~~ tendons**, one (1) horizontal, one (1) diagonal, and one (1) vertical,** every other **refueling** outage**. These three tendons, per unit, will be selected based on the existing lift-off data and practical access limitations. They will be the common tendons for lift-off testing. Prestressing forces will be trended and projected through the next two operating cycles, i.e. every 48 months. Visual inspections will also be performed on both ends of the common tendons on the same frequency as the lift off tests.**

Visual inspections will be also performed on tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, one (1) middle horizontal, one (1) lower horizontal, one (1) diagonal, and one (1) vertical, per unit, every other refueling outage, i.e. every 48 months, occurring on alternate outages from the lift off tests.

Acceptance criteria outlined in station specifications and procedures ensures appropriate corrective actions are taken based on the observed and/or measured conditions. If an inspection identifies a degraded condition associated with a tendon or the tendon anchorage, the corrective action program is utilized to facilitate repair or replacement activities.

SLRA Appendix B4.1 (page B-308) is revised as follows:

Parameters Monitored or Inspected – Element 3

The *Secondary Shield Wall Tendon Surveillance* program utilizes visual inspections to inspect for loss of material and cracking of tendon anchorage, bearing plates, and shims. Tendon wires are inspected for loss of material and breakage, and button-heads are inspected for cracks, splits or broken button-heads. The program also monitors for loss of preload (prestress) by comparing measured lift-off forces to the established minimum required force for each tendon group, as described in station procedures.

Oconee performs visual inspections and lift-off tests on three randomly selected horizontal tendons every other refueling outage. **The program will be enhanced to perform lift off tests on three tendons; one (1) horizontal, one (1) diagonal, and one (1) vertical tendon every other refueling outage, i.e. every 48 months. These three tendons, per unit, will be selected based on the existing lift off data and practical access limitations. They will be the common tendons for lift off testing. Visual inspections will also be performed on both ends of the common tendons on the same frequency as the lift off tests.**

Additional enhancements include performing visual inspections on the tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, one (1) middle horizontal, one (1) lower horizontal, one (1) diagonal, and one (1) vertical, per unit, every other refueling outage, i.e. every 48 months, occurring on alternate outages from the lift off tests.

Detection of Aging Effects – Element 4

The *Secondary Shield Wall Tendon Surveillance* AMP detects aging effects prior to the loss of the secondary shield wall tendon intended function. This AMP is based on performing secondary shield wall tendon surveillances in accordance with site specific procedures.

Visual inspections and lift-off tests are performed on three randomly selected horizontal tendons every other refueling outage. **The program will be enhanced to perform lift off tests on three tendons; one (1) horizontal, one (1) diagonal, and one (1) vertical tendon every other refueling outage, i.e. every 48 months. These three tendons, per unit, will be selected based on the existing lift off data and practical access limitations. They will be the common tendons for lift off testing. Visual inspections will also be performed on both ends of the common tendons on the same frequency as the lift off tests.**

Additional enhancements include performing visual inspections on the tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, one (1) middle horizontal, one (1) lower horizontal, one (1) diagonal, and one (1) vertical, per unit, every other refueling outage, i.e. every 48 months, occurring on alternate outages from the lift off tests.

Visual inspections include looking for moisture, discoloration, foreign matter, rust, corrosion, splits or cracks in the button-heads, broken or missing wires, and other obvious damage.

Lift-off testing is performed on a selected number of tendons every other refueling outage. Lift-off forces are measured and compared to established acceptance criteria. **The program will be enhanced to trend and project prestressing forces through the next two operating cycles, i.e. every 48 months.**

Monitoring and Trending – Element 5

Condition monitoring for degradation of tendon wires and anchorage is performed by periodic visual inspections, looking for loss of material and cracking. The program also requires monitoring loss of prestress by comparing the measured lift-off forces to the established minimum required force for each tendon group, as specified in station procedures. Station procedures will be enhanced to include reviewing previous visual inspection results for tendon wires and anchorages condition, and lift-off results for the tendons selected for inspection. **A programmatic enhancement will use the common tendons to trend the prestressing losses and to project the tendons prestress losses through the next two operating cycles, i.e. every 48 months.**

SLRA Appendix B4.1 (page B-312) is revised as follows:

Enhancements

The following enhancements will be implemented in the following program elements:
Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4), and Monitoring and Trending (Element 5)

1. **Perform lift-off testing on each of the three units on one (1) horizontal, one (1) diagonal, and one (1) vertical tendon, pulled from one end, every second refueling outage, i.e. every 48 months. These three tendons, per unit, will be selected based on the existing lift off data that is available as well as practical access limitations, and they will be the common tendons for lift off testing. (Elements 3, 4, and 5)**
2. Enhance station procedures to include a review of previous visual inspection and lift-off data results for the tendons selected for inspection. **Revise station procedures to use the common tendon lift-off data to trend the prestressing forces and to project tendon prestress losses through the next two operating cycles, i.e. every 48 months.** (Element **4 and** 5)
3. **Perform a visual inspection on both ends of the common tendons every other refueling outage, i.e. 48 months, during the same outage as the lift off tests. (Elements 3, 4, and 5)**
4. **Perform visual inspections on tendon caps and stressing hardware on both ends of the following randomly selected tendons; one (1) upper horizontal, (1) middle horizontal, (1) lower horizontal, (1) diagonal, and (1) vertical, per unit, every second refueling outage, i.e. every 48 months, occurring on alternate outages from the lift-off tests. (Elements 3, 4, and 5)**

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 11
RAI B4.1-2

Enclosure 1, Attachment 11

RAI B4.1-2:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Background:
SRP-SLR Section A.1.2.3.5 provides the branch technical position for reviewing element 5, “monitoring and trending,” for plant-specific AMPs. Section A.1.2.3.5 notes that “results of inspections in the prior period of extended operation are used to provide input to trending results,” and that “trending is a comparison of the current monitoring results with previous results in order to make predictions for the future.” The SRP-SLR further notes that where practical, degradation should be projected to the next scheduled inspection.

Issue:
SLRA Section B4.1, Element 5, discusses monitoring and trending of the prestress in the SSW tendons. The SLRA notes that prestress is monitored by comparing the measured lift-off forces to the established minimum required force for each tendon group. The SLRA also notes that plant procedures will be enhanced to include a review of previous lift-off data results for the tendons selected for inspection; however, no discussion is provided regarding predicting future results or projecting the prestressing losses to the next scheduled inspection.

Request:
Explain how prestressing losses are projected for the SSW tendons or provide justification for not performing trending assessments of the SSW tendon losses to make projections to the next inspection.

Response to RAI B4.1-2:

An enhancement to the *Secondary Shield Wall Tendon Surveillance* program has been identified in response to RAI B4.1-1 that will account for the projection of prestressing losses. The enhancement will assure that lift off testing is performed on each unit for one (1) horizontal, one (1) diagonal, and one (1) vertical tendon to address the loss of prestress for the SSW tendons. These three tendons will become the common tendons for subsequent lift off testing.

Using the common tendon concept for the lift off testing allows for trending of the prestressing losses and projecting the tendon prestress losses through the next two operating cycles, i.e. every 48 months, for the common tendons and by extrapolation for the full SSW tendon set.

SLRA Revisions:

See the SLRA Revisions provided in the response to RAI B4.1-1 in Enclosure 1, Attachment 10.

ENCLOSURE 1

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ATTACHMENT 12
RAI 4.6.1-1

Enclosure 1, Attachment 12

RAI 4.6.1-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(c) requires the applicant to evaluate time-limited aging analyses (TLAA) and disposition them in accordance with (c)(1)(i), (c)(1)(ii), or (c)(1)(iii). 10 CFR 54.21(d) requires that the FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and evaluation of the TLAA for the period of extended operation determined by 54.21(a) and 54.21(c), respectively.

Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
SLRA Section 4.6.1, “Containment Liner Plate” under TLAA disposition for the containment liner plate states, in part: “As described in UFSAR Section 3.8.1.5.3, the only portions of the liner plate that contain fatigue analysis are those thickened portions at the penetrations.” A similar statement is made in UFSAR supplement for the TLAA in SLRA Section A4.6.1.

SLRA Section 4.6.1 and its UFSAR supplement description in SLRA Section A4.6.1 concluded that thermal fatigue of the containment liner plate would be acceptable for the subsequent period of extended operation (SPEO), and further states that the effects of fatigue of the containment liner plate will be adequately managed by the Fatigue Monitoring AMP (B3.1) for the SPEO in accordance with 10 CFR 54.21(c)(1)(iii).

SLRA Section 4.6.1 states, in part: “The Fatigue Monitoring AMP will track cycles for significant fatigue transients listed in Table 4.3.1-1 and ensure corrective action is taken prior to potentially exceeding design limits.”

The acceptance criteria for FSAR Supplement in SRP-SLR Section 4.6.2.2 states:

> The specific criterion for meeting 10 CFR 54.21(d) is that the summary description of the evaluation of TLAAs for the subsequent period of operation in the FSAR Supplement is sufficiently comprehensive, such that later changes can be controlled by 10 CFR 50.59. The description contains information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

Issue:

1. The staff understands that the liner plate is thickened at the penetrations to address stress concentrations at discontinuities. However, the staff notes from review of UFSAR Section 3.8.1.5.3 (p 3.8.1-30 and -31) and the SLRA Section 4.6.1 under the title “TLAA description” that the fatigue loads stated therein apply to and were considered in the design of the entire liner plate (not only the thickened liner plate), which appear inconsistent with the above referenced statement from the SLRA.

2. From the SLRA Section 4.6.1 and A4.6.1 descriptions, it is not clear what fatigue parameter (e.g., cumulative fatigue damage, cracking due to cyclic loading, fatigue waiver analyses parameter(s)) is managed by the Fatigue Monitoring Program, the relevant transients that are monitored and how monitoring those parameters assure adequate fatigue management of the containment liner plate consistent with the disposition of 10 CFR 54.21(c)(1)(iii).

3. The staff is unable to make its determination that the UFSAR supplement provides an adequate summary description for the TLAA equivalent to that in SRP-SLR Table 4.6-1 for the TLAA disposition in accordance with 10 CFR 54.21(c)(1)(iii).

Request:

1. Clarify or correct the referenced statements from SLRA 4.6.1 disposition and A4.6.1 in Issue 1, regarding that fatigue analysis are only for thickened portions of the liner, which appears inconsistent with the description in the UFSAR and SLRA 4.6.1 TLAA description. Clearly state whether or not the TLAA evaluation described in SLRA Section 4.6.1 and A4.6.1 applies to the containment liner plate (i.e., portions backed by concrete) in its entirety.

2. Identify the fatigue parameter (e.g., cumulative fatigue damage, cracking due to cyclic loading) that is managed by the Fatigue Monitoring Program in accordance with the TLAA disposition for the containment liner plate, the relevant transients that are monitored by the program and how monitoring those transients assure adequate fatigue management of the containment liner plate.

3. Provide a revised SLRA Section A4.6.1 UFSAR supplement summary description, consistent with the disposition of the containment liner plate fatigue TLAA in SLRA Section 4.6.1, with sufficient information that includes the specific fatigue evaluation parameter that is managed by the Fatigue Monitoring Program, the specific relevant transients monitored by the program, and how monitoring those transients assures adequate management of cumulative fatigue damage of the containment liner plate.

4. If a TLAA, as defined in 10 CFR 54.3, does not exist in the current licensing basis (CLB) for the containment liner plate information presented in SLRA Section 4.6.1, state so and describe how the aging effect of cumulative fatigue damage or cracking due to cyclic loading will be adequately managed for the subsequent period of extended operation pursuant to 10 CFR 54.21(a)(3) and the guidance for SRP-SLR Table 3.5-1, item 027 and its associated further evaluation in SRP-SLR Section 3.5.2.2.1.5, as modified in Interim Staff Guidance (ISG) SLR-ISG-2021-03-Structures (ADAMS Accession No. ML20181A381).

5. Provide applicable and necessary updates to the SLRA consistent with the responses to the requests above.

Response to RAI 4.6.1-1:

Request 1:

The fatigue loads described in UFSAR Section 3.8.1.5.3 and ONS SLRA Section 4.6.1 apply to the containment liner plate in its entirety. For 60-years of operation the containment liner plate was evaluated against these fatigue loads and it was determined that the analysis remained valid for the period of extended operation. For 80-years of operation the accumulated effects of the loading conditions on the containment liner and penetrations, not including main steam and main feedwater, were evaluated in accordance with the ASME Code, Section III, Paragraph N-415, to determine the need for a detailed fatigue analysis. The evaluation considered the combined loading of thermal cycling due to reactor building interior temperature varying during the startup and shutdown of the reactor coolant system and Type A integrated leak rate tests required by 10 CFR 50, Appendix J, including any Type A tests that may be performed if major modifications or repairs are made to the containment pressure boundary which provides an essentially leak tight barrier. These loadings are the governing design aspect of the liner and are assumed to be 500 cycles. Under this specified number of cycles, a fatigue waiver analysis was completed that determined no detailed fatigue analysis is required as all the conditions in the ASME Code, Section III, Subsection N-415.1 are shown to be satisfied. Section 4.6.1 is revised as shown in the attached SLRA markup.

Request 2:

As described above, a fatigue waiver analysis was performed for the containment liner plate and penetrations, excluding main steam and main feedwater, which are addressed in both SLRA Section 4.6.3 and further in the response to RAI 4.6.3-1. The disposition for SLRA Section 4.6.1 is revised as shown in the attached SLRA markup.

Request 3:

A revised SLRA Section A4.6.1 is included below. This supplement describes the fatigue waiver evaluation for the containment liner plate and cold penetrations, including mechanical, electrical, equipment and personnel-related penetrations.

Request 4:

A TLAA does exist for the containment liner plate and cold penetrations. For 60-years of operation, the evaluation of this TLAA is described in NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, Section 4.2.1. For 80-years of operation a fatigue waiver evaluation was performed in accordance with the ASME Code, Section III, Paragraph N-415. The conclusion of the evaluation was that a detailed fatigue analysis was not required for these components. Revisions to ONS Sections 4.6.1 and A4.6.1 are provided below.

Request 5:

The SLRA Supplements to ONS SLRA Table 4.1.4-1, Section 4.6.1, and Section A4.6.1 are provided below.

SLRA Revisions:

SLRA Table 4.1.4-1 (page 4-5) is revised as follows:

Table 4.1.4-1: Oconee Time-Limited Aging Analyses Categories and Dispositions

TLAA CATEGORY	ANALYSIS	DISPOSITION (Note 1)	SECTION
ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT	Environmental Qualification of Electrical Equipment Program	(iii)	4.4
CONCRETE CONTAINMENT TENDON PRESTRESS	Concrete Containment Tendon Prestress	(iii)	4.5
CONTAINMENT LINER PLATE, METAL CONTAINMENTS AND PENETRATIONS FATIGUE ANALYSES	Containment Liner Plate	~~(iii)~~ **(ii)**	4.6.1
	Metal Containment	Not Applicable	4.6.2
	Containment Penetrations Fatigue Analyses	(iii)	4.6.3

SLRA Section 4.6.1 (page 4-99) is revised as follows:

4.6.1 CONTAINMENT LINER PLATE AND COLD PENETRATIONS

TLAA Description:

The interior surface of the containments for Oconee Units 1, 2, and 3 are lined with a ¼" thick welded steel plate to provide an essentially leak-tight barrier. At all penetrations, the liner plate is thickened to **1/2" to** reduce stress concentrations. The following fatigue loads were considered in the design of the liner plate, as described in USFAR Section 3.8.1.5.3, and are considered to be time-limited aging analyses for the purpose of license renewal:

(a) Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 cycles for the plant life of 40 years. The Oconee liners plates are positioned on the interior of the Containment and are shielded from the outdoor environments by an approximately 3-3/4 ft wall of **reinforced** concrete. The change in temperature from the outer surface of the concrete to the inner surface would be insignificant from a thermal fatigue impact. Furthermore, increasing the 40 cycles to 80 cycles for 80 years of operation would be considered insignificant in comparison to the assumed 500 thermal cycles due to variations of containment interior temperature during heatup and cooldown of the reactor coolant system.

(b) The combined loading ~~of~~ **on the containment liner includes 1)** thermal cycling due to reactor building interior temperature varying during the startup and shutdown of the reactor coolant system and **2) the** Type A integrated leak rate tests required by 10 CFR 50, Appendix J, including any Type A tests that may be performed if major modifications or repairs are made to the containment pressure boundary. The number of cycles for this loading is assumed to be 500 cycles. This loading is the governing design aspect that is the focus to be evaluated for the SPEO.

(c) Thermal cycling due to the LOCA will be assumed to be one cycle. This is not a time-limited design parameter.

(d) Thermal load cycles in the piping system are somewhat isolated from the liner plate penetrations by concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the plant life~~ in Section 4.6.3~~.

TLAA Evaluation:

Steel material cycling, both thermal and mechanical (pressure) was an aspect considered in the design of the entire containment liner, including the ¼ inch welded steel plate liner and the various penetrations through this liner. These penetrations can be divided into lower temperature or cold penetrations and a few higher temperature or hot penetrations. The cold penetrations were considered as those that experience a temperature generally less than 150 F. The set of these cold penetrations include mechanical, electrical, and equipment and personnel-related penetrations. The higher temperature or hot penetrations are limited to the main steam and feedwater penetrations and are addressed in Section 4.6.3.

For the welded liner and the cold penetrations, ~~T~~**t**he associated temperature and pressure changes from reactor coolant system heatups and cooldowns and Type A integrated leak rate test are the main contributors to **cycling of the steel material. Table 4.6.1-1 describes the materials of construction used in the liner plate and cold penetrations for ONS Unit 1, 2, and 3** ~~fatigue of the containment liner plate~~.

Table 4.6.1-1: ONS Liner Plate and Cold Penetrations Materials of Construction

ASTM	Material	Type	Components
A36	Carbon Steel	Containment Liner	Plate
SA-516 Grade 70	Carbon Steel	Containment Liner Mech. Penetrations Elec Penetrations	Plate
A-333 Grade 6	Carbon Steel	Mech. Penetrations Elec. Penetrations	Pipe
A-155	Carbon Steel	Mech. Penetrations Elec. Penetrations	Pipe
A36	Carbon Steel	Elec Penetrations	Rod
SA-516 Grade 70	Carbon Steel	Mech. Penetrations Elec. Penetrations	Pipe Cap/Dished Head
SA-106 Grade C	Carbon Steel	Mech Penetrations	Pipe
A-350 Grade LF2	Carbon Steel	Elect. Penetrations	Forging
A-234 Grade WPB	Carbon Steel	Elec. Penetrations	Fittings
SA-182 F304	Stainless Steel	Elect. Penetrations	Flange
SA-316	Stainless Steel	Mech Penetrations	Plate

The design of the containment liner ~~relied on~~ **considered** a cumulative number of 500 cycles for these transients~~, which control the other design parameters of the liner plate~~. **The practical outcome of this consideration was the thickening of the welded plate at each penetration to assure margin in the design and was considered good engineering practice at that time. No documented analytical basis was identified that addressed these aspects of cycling for the liner and the cold penetrations.**

SLRA Section 4.6.1 (page 4-100) is revised as follows:

When evaluating the continued validity of this design consideration, an understanding of other governing limits in the plant design is useful. The ONS UFSAR, Table 5-2, provides the 40-year design transient number of 360 heatup and cooldown cycles for the reactor coolant system. The projected number of cycles for each ONS unit through 80 years of operation has been determined to be less that the original 360 cycle design limits as shown in Table 4.3.1-1. Periodic Type A integrated leak rate tests are additional major sources of load changes. ~~These Type A loads are considered within the set of design loads whose cumulative total was assumed to be 500 cycles.~~ For license renewal it was projected that there would be a total of 11 Type A tests performed per unit, with 7 being documented for each unit ~~in~~ **through** 1998. This would result in approximately 25 Type A tests per each unit to the end of the SPEO. **These Type A loads are considered within the set of design loads whose cumulative total was assumed to be 500 cycles.**

Combining the total number of cycles for outdoor temperature variations, the total number of Type A tests, and the projected heatups and cooldowns from Table 4.3.1-1 would result in a transient cycle count less than 500 cycles. ~~Therefore, it is concluded that thermal fatigue of the containment liner~~

~~would be acceptable for the SPEO. The *Fatigue Monitoring* AMP will track cycles for significant fatigue transients listed in Table 4.3.1-1 and ensure corrective action is taken prior to potentially exceeding fatigue design limits.~~ **Therefore, the 500 cycles considered in the design still bounds the actual cycles for the containment liner plate and cold penetrations and remain valid.**

The ONS UFSAR, Section 3.8.1.5.3, states that ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, was considered in the design of the liner plate and cold penetrations. For 80-years of operation, the accumulated effects of containment liner and cold penetration loading conditions were evaluated to contemporary standards in accordance with the ASME Code, Section III, Paragraph N-415, to validate that a detailed fatigue analysis would not be warranted. The containment liner plate and cold penetration materials of construction listed in Table 4.6.1-1 above, in particular the carbon steel SA-516 Grade 70 material, meets all six criteria in the ASME Code, Section III, Subsection N-415.1 for the 500 applied design cycles for these components. This evaluation is bounding for the liner plate and cold penetrations, including mechanical, electrical, equipment and personnel-related penetrations. No detailed fatigue analysis is required for the containment liner or cold penetrations due to stresses induced by pressure and temperature changes.

TLAA Disposition: 10 CFR 54.21(c)(1)~~(iii)~~**(ii)**

~~The effects of fatigue on the intended function(s) of the containment liner plate will be adequately managed by the *Fatigue Monitoring* AMP (B3.1) for the SPEO.~~

~~As described in UFSAR Section 3.8.1.5.3, the only portions of the liner plate that contain fatigue analysis are those thickened portions at the penetrations. The containment penetration fatigue analysis is contained in Section 4.6.3 and validates these sections of the liner plate. Therefore, this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).~~

The fatigue waiver analysis associated with the containment liner plate and cold penetrations meets all six criteria in the ASME Code, Section III, Subsection N-415.1 and will remain valid for the subsequent period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

SLRA Appendix A4.6.1 (pages A-65 and A-66) is being revised as follows:

A4.6.1 Containment Liner Plate **and Cold Penetrations**

The interior surface of the containments for ONS Units 1, 2, and 3 are lined with a ¼" thick welded steel plate to provide an essentially leak-tight barrier. At all penetrations, the liner plate is thickened to **½" to** reduce stress concentrations. As described in UFSAR 3.8.1.5.3 *Liner Plate*, the ~~only portions of the liner plate that contain fatigue analysis are those thickened portions at the penetrations.~~ **liner plate design considered a specific set of design transients. Since initial design and construction, two penetrations, main steam and main feedwater, have received additional detailed design transient analyses due to higher temperature cycling concerns. These two are termed hot penetrations and are discussed in Section 4.6.3. The remaining penetrations that do not experience higher temperature cycling are termed cold penetrations. The transient cycling considerations of the containment liner and the cold penetrations, as described in UFSAR 3.8.1.5.3 have been evaluated as a TLAA for subsequent license renewal.** ~~The containment penetration fatigue analysis is contained in Section 4.6.3.~~

The associated temperature and pressure changes from reactor coolant system heatups and cooldowns and Type A integrated leak rate test are the main contributors to fatigue of the containment liner plate. Combining the total number of cycles for outdoor temperature variations, the total number of Type A tests, and the projected heatups and cooldowns would result in a transient cycle count less than 500 cycles. ~~Therefore, it is concluded that thermal fatigue of the containment liner would be acceptable for the SPEO. The effects of fatigue of the containment liner plate will be adequately managed by the *Fatigue Monitoring* AMP (B3.1) for the SPEO in accordance with 10 CFR 54.21(c)(1)(iii).~~ **A fatigue waiver evaluation was completed for the containment liner plate and cold penetrations, including mechanical, electrical, equipment and personnel-related penetrations, in accordance with ASME Section III, Paragraph N-415. The evaluation determined that a detailed fatigue analysis was not needed and will remain valid for the subsequent period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).**

ENCLOSURE 1

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ATTACHMENT 13
RAI 4.6.3-1

Enclosure 1, Attachment 13

RAI 4.6.3-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(c) requires the applicant to evaluate time-limited aging analyses (TLAA) and disposition them in accordance with (c)(1)(i), (c)(1)(ii), or (c)(1)(iii). 10 CFR 54.21(d) requires that the FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and evaluation of the TLAA for the period of extended operation determined by 54.21(a) and 54.21(c), respectively.

Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
SLRA Section 4.6.3 states that the effects of fatigue on the intended functions of the containment main steam penetrations and main feedwater penetrations will be adequately managed by the Fatigue Monitoring AMP (B.3.1) for the subsequent period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

The UFSAR Supplement in SLRA Section A4.6.3 “Containment Penetrations Fatigue Analysis,” as amended by Supplement 2 dated November 11, 2021 (ADAMS Accession No. ML21315A012), states in part:

> The transient cycles considered in the main steam and feedwater penetrations analyses were projected for 80 years of operation and the count found to be adequate for the SPEO. The *Fatigue Monitoring* (A3.1) aging management program will monitor and track the relevant transients to manage fatigue of the main steam and feedwater penetrations during the SPEO in accordance with 10 CFR 54.21(c)(1)(iii).

The acceptance criteria for FSAR Supplement in SRP-SLR Section 4.6.2.2 states:

> The specific criterion for meeting 10 CFR 54.21(d) is that the summary description of the evaluation of TLAAs for the subsequent period of operation in the FSAR Supplement is sufficiently comprehensive, such that later changes can be controlled by 10 CFR 50.59. The description contains information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

The staff verifies that the applicant has provided an FSAR Supplement with information equivalent to that in SRP-SLR Table 4.6-1 for disposition under 10 CFR 54.21(c)(1)(iii).

Issue:
The fatigue TLAA evaluation in SLRA Section 4.6.3, and its UFSAR supplement description in SLRA Section A4.6.3 (as amended) do not state what specific fatigue evaluation parameter (e.g., cumulative fatigue damage (cumulative usage factor or CUF), cracking due to cyclic loading, etc.) is adequately managed by the Fatigue Monitoring Program by monitoring the relevant transient cycles. Further, the UFSAR Supplement summary description does not state, to assure program effectiveness, the specific relevant transients that will be monitored and what fatigue evaluation parameter is maintained within

what acceptance criteria (e.g., CUF maintained less than or equal to 1, or require corrective actions prior to that limit) by monitoring those transients against the stated allowable cycle counts or require corrective actions when cycle count limits are approached.

The staff needs additional information to make its determination that the UFSAR supplement provides an adequate summary description for the TLAA equivalent to that in SRP-SLR Table 4.6-1.

Request:

1. State the specific fatigue evaluation parameter that will be adequately managed by the Fatigue Monitoring Program (SLRA B3.1) and by monitoring what specific transient cycles for the fatigue TLAA evaluation for containment penetrations in SLRA Section 4.6.3.

2. Provide a revised SLRA Section A4.6.3 UFSAR supplement summary description consistent with the disposition of the main steam and main feedwater piping penetrations fatigue TLAA in SLRA Section 4.6.3, with sufficiently comprehensive information that includes the specific fatigue evaluation parameter (with limit) that is managed by the Fatigue Monitoring Program, the specific relevant transients monitored by the program and how monitoring those parameters assures adequate management of cumulative fatigue damage including triggering corrective action.

Response to RAI 4.6.3-1:

Request 1:

ONS SLRA Section 4.6.3 addresses the fatigue TLAA evaluations for the main steam and main feedwater penetrations. As stated in Section 4.6.3, the original design for these components was to the 1965 Edition of ASME Section III. This was revised for the main steam and main feedwater penetrations to ASME Section III, Subsection NE and NC, 1992 Edition with 1992 Addenda. Fatigue loads were considered in the current analysis of the main steam and main feedwater penetrations and a fatigue usage was calculated for these components. Therefore, cumulative fatigue damage (cumulative usage factor or CUF) is the specific fatigue parameter that will be managed by the *Fatigue Monitoring* AMP.

The initial fatigue evaluations for the main steam and main feedwater penetration utilized the 40-year design allowable cycles which resulted in a usage factor greater than unity (greater than 1.0). A refined set of allowable cycles was used to achieve a CUF that is less than 1.0 and is detailed in ONS SLRA Table 4.6.3-1. For the main steam penetrations, the transients that are tracked by the *Fatigue Monitoring* program are heatups/cooldowns (Transients 1A and 1B) and the total number of reactor trips, which is the addition of Transients 8A, 8B, 8C and 8D from ONS SLRA Table 4.3.1-1. For the main feedwater penetrations the transients that are tracked by the *Fatigue Monitoring* program are heatups and cooldowns (Transients 1A and 1B). The 80-year projections for these transients remain below the refined allowable cycles for these penetrations.

Request 2:

SLRA Appendix A4.6.3 is being revised. See below for revisions.

SLRA Revisions:

SLRA Appendix A4.6.3 (page A-66) is revised as follows:

A4.6.3 Containment Penetrations Fatigue Analysis

The interior surface of the containment is lined with welded steel plate to provide an essentially leak tight barrier. **These penetrations can be divided into lower temperature or cold penetrations and a few higher temperature or hot penetrations.** At all penetrations, the liner plate is thickened to reduce stress concentrations of the liner plate. ONS Units 1, 2 and 3 process lines that penetrate the primary containment and experience significant thermal expansion and contraction**, the higher temperature, or hot penetrations,** are solidly anchored to the containment wall. These high temperature lines penetrating the containment wall and liner plate are the main steam and feedwater lines. **Fatigue evaluations performed on the main steam and main feedwater penetrations utilize a refined number of allowable cycles to achieve a cumulative usage factor less than unity. For the main steam penetration, the allowable number of heatups and cooldowns is 262 cycles each. The maximum number of total reactor trips for the main steam penetration is 202 cycles. In addition, the main steam penetration was qualified for 5 operating basis earthquake (OBE) events. For the main feedwater penetration, the allowable number of heatups and cooldowns is 249 cycles each. In addition, the main feedwater penetration was qualified for 3 OBE events. Monitoring and tracking of these transients will ensure that the aging effect of cumulative fatigue damage is managed for these penetrations.**

The transient cycles considered in the main steam and feedwater penetrations analyses were projected for 80 years of operation and the count found to be adequate for the SPEO. **The *Fatigue Monitoring* (A3.1) AMP will monitor and track these applicable fatigue transients and ensure corrective action is taken prior to exceeding fatigue design limits.** ~~The Fatigue Monitoring (A3.1) aging management program will monitor and manage fatigue of the main steam and feedwater penetrations during the SPEO in accordance with 10 CFR 54.21(c)(1)(iii).~~ **The effects of fatigue on the main steam and main feedwater penetrations will be adequately managed by the *Fatigue Monitoring* AMP (A3.1) for the SPEO in accordance with 10 CFR 54.21(c)(1)(iii).**

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 14
RAI 4.2.3-1
[**NON-PROPRIETARY]**

Enclosure 1, Attachment 14
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 4.2.3-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.3 defines the criteria to qualify a certain analysis as a Time-Limiting Aging Analysis (TLAA). For each analysis that is determined to meet the definition of a TLAA, 10 CFR 54.21(c) requires the applicant to evaluate the TLAA in the Subsequent License Renewal Application (SLRA) to demonstrate that the TLAA either: (i) will remain valid for period of extended operation; (ii) has been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) of the component(s) assessed in the TLAA will be adequately managed during the period of extended operation..

Background:
The applicant dispositioned the Pressurized Thermal Shock (PTS) TLAA in accordance with 10 CFR 54.21(c)(ii). Applicable to extended beltline locations, Section 3.4.1 of ANP-3898P, Revision 0, discusses how the Cu wt% content was determined for ONS Unit 1 and Unit 2 RPV inlet (INF) and outlet (ONF) nozzle forgings and transition forgings, and Unit 3 RVP INF. The staff notes that Section 3.4.1 of ANP-3898P does not include a discussion regarding the Cu wt% value for the RPV ONF and transition forging for Unit 3.

Issue:
The staff lacks clarity regarding how the applicant determined a generic Cu wt% value for the Unit 3 RPV ONF and transition forging. These forgings show a different Cu wt% value in Table 5-12 of ANP-3898P, Revision 0, compared to the generic value determined in Section 3.4.1 for extended beltline materials. During its review of ANP-3898P and supporting documents, the staff was not able to identify how the applicant determined the applicable Cu wt% value for the Unit 3 RPV ONF and transition forging.

Request:
Describe how the generic Cu wt% value for Unit 3 RPV ONF and transition forging was determined and provide a justification for using this value.

Response to RAI 4.2.3-1:

As described in ANP-3898P, Revision 0, the ONS Unit 3 RPV outlet nozzle forgings and transition forging were procured from Klockner-Werke, a European supplier, with no reported copper content in the certified material test report (CMTR). **[**

]b,c,d,e

The mean copper content for these ingots was found to be **[ ]**$_{b,c,e}$ weight percent, the sample standard deviation is **[ ]**$_{b,c,e}$ and the one-sided factor (k) is **[ ]**$_{b,c,e}$. The generic value is calculated to be **[ ]**$_{b,c,e}$ weight percent (mean plus kσ).

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

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ATTACHMENT 15
RAI 4.2.3-2
[NON-PROPRIETARY]

Enclosure 1, Attachment 15
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 4.2.3-2:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.3 defines the criteria to qualify a certain analysis as a Time-Limiting Aging Analysis (TLAA). For each analysis that is determined to meet the definition of a TLAA, 10 CFR 54.21(c) requires the applicant to evaluate the TLAA in the Subsequent License Renewal Application (SLRA) to demonstrate that the TLAA either: (i) will remain valid for period of extended operation; (ii) has been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) of the component(s) assessed in the TLAA will be adequately managed during the period of extended operation.

Background:
Section 5.4.1 of ANP-3898P, Revision 0, establishes the initial RTNDT (generic mean) and σI (standard deviation) values for ONS Unit 3 RPV ONF and transition forging. The applicant states that these forgings were fabricated by a non-US supplier (Klockner-Werke), and the generic mean and standard deviation values were determined from the material data set in Table 5-2 of ANP-3898P, Revision 0. ANP-3898P, Revision 0, states that the actual suppliers of the forgings to Rotterdam Dockyards for the plants listed in Table 5-2 are not known but likely included Klockner-Werke, Fried-Krupp Huttenwerke AG, and Rheinstahl Huttenwerke AG based on a review of Table 2 of PWROG-17090 (ADAMS Accession Number ML20023E238).

Issue:
The staff lacks clarity regarding how the forgings in Table 5-2 of ANP-3898P, Revision 0, are applicable or representative of the ONS Unit 3 RPV ONF and transition forging. In addition, the applicant states that it reviewed PWROG-17090 as part of determining the data set in Table 5-2 of ANP-3898P, Revision 0, but it is not clear to the staff how the applicant’s review yielded the dataset provided in Table 5-2. Specifically, the staff performed a search of the applicable materials in Table 5-2 within PWROG-17090 and was not able to reference them within the document.

Request:
Justify how the forgings in Table 5-2 of ANP-3898P, Revision 0, are applicable or representative to ONS Unit 3 RPV ONF and transition forging. As part of your response, include a discussion of the source information for these forgings and why their selection is conservative, applicable, or representative for ONS Unit 3.

Response to RAI 4.2.3-2:

Reg. Guide 1.99, Revision 2 states if measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a generic mean and standard deviation for the class. **[**

]b,c,d,e demonstrate the Rotterdam Dockyard (European supplier) forging specific generic values are the most applicable available data set for the ONS Unit 3 RPV ONF and transition forging, which were procured from Klockner-Werke (European supplier).

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 16
RAI B2.1.15-1

Enclosure 1, Attachment 16

RAI B2.1.15-1:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Oconee Subsequent License Renewal Application (SLRA) Tables 3.5.2-1 (Auxiliary Building) and 3.5.2-3 (Turbine Building) list a “masonry wall” component type, with a fire protection intended function and include two materials: “masonry walls” and “concrete block.” For masonry walls materials, the tables credit both the Fire Protection and the Masonry Walls programs to manage cracking. However, for concrete block materials these tables only credit the Masonry Walls program to manage cracking of the concrete block masonry walls with a fire barrier intended function and do not include the Fire Protection program.

Issue:
It is not clear whether: a) the different treatment of masonry walls and concrete block materials noted above was an oversight, b) the listed masonry walls constructed of concrete block do not have a fire barrier intended function and, consequently, do not need to be included in the Fire Protection program, or c) inspections done by the Masonry Walls program can ensure the fire barrier intended function of the concrete block masonry walls is being maintained.

Request:
Clarify which of the issues discussed above is applicable to this situation and if appropriate provide a discussion and any changes to the SLRA. If the inspections done by the Masonry Walls program are being credited for ensuring the fire barrier intended function of the concrete block masonry walls is maintained during the subsequent period of extended operation, then include additional information (e.g., inspections, acceptance criteria, and corrective actions are equivalent to those in the Fire Protection program; inspections are performed on the same frequency as required by the Fire Protection program; and the credited program procedures have been updated, if necessary, to address the fire barrier intended function).

Response to RAI B2.1.15-1:

The Masonry Walls are constructed of concrete block. The material types concrete block and masonry walls are used interchangeably. The Masonry Walls provide structural support; shelter, protection; and fire barrier intended functions. NUREG-2191(GALL SLR) provides alignments for masonry walls in Section III and Section VII. The Section III alignment addresses the structural support and shelter, protection intended functions, but the fire barrier function is not addressed with the Section III alignment. The Section VII alignment addresses the fire barrier intended function but does not address the structural intended functions. Both alignments are utilized to capture all potential aging mechanisms for the Masonry Walls that contain a fire barrier function. The material is shown as concrete block for the Section III alignment and masonry walls for Section VII alignment to match the SLR-SRP.

The Turbine Building and Auxiliary building share an interior concrete block wall (masonry wall) that serves the functions identified above.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 17
RAI B2.1.15-2

Enclosure 1, Attachment 17

RAI B2.1.15-2:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
SLRA Section 2.4.8.2 states, “Armaflex is a flexible insulation material which is installed in penetrations in the floor and ceiling of east and west penetration rooms (of each unit) for pressure boundary conditions. The NRC has accepted Armaflex as a fire barrier and has exempted it from testing and rating requirements normally required for fire barriers.”

The table in Section 4.4 of Enclosure 9.5, “Fire Barrier Penetration Configuration Identification,” in implementing procedures MP/1/A/1705/018, MP/2/A/1705/018, and MP/3/A/1705/018, “Fire Protection – Penetration – Fire and Flood Barrier – Inspection and Minor Repair,” state that Armaflex is approved for “Flood only in East Penetration room floors.”

Issue:
SLRA Section 2.4.8.2 states that Armaflex is a credited fire barrier material; however, the implementing procedures state it is only approved for flood barriers. The NRC staff notes that neither SLRA Table 2.4.8-2 nor SLRA Table 3.5.2-23 identify Flood Barrier as an intended function for component type “Fire Barriers – Penetration Seals.” In addition, the staff notes that the implementing procedures referenced above include other sealant types that are approved for both fire and flood.

Based on the implementing procedures referenced above, it appears that flood barriers are inspected at the same frequency as fire barriers; however, it is unclear to the NRC staff whether the “detection of aging effects,” “acceptance criteria,” and “corrective actions” program elements for fire barriers are bounded by the comparable program element associated with flood barriers. The staff notes that SLRA Table 3.5.2-23 lists the aging effects requiring management for elastomeric fire barriers as “hardening, loss of strength, or shrinkage,” using the Fire Protection program, whereas the corresponding aging effects for flood barriers is only “loss of sealing,” using the Structures Monitoring program.

Request:

1. Discuss and address the apparent disparity between the SLRA statement and the implementing procedures relative to the approved function(s) of Armaflex (e.g., fire barrier, flood barrier, or both).

2. If the inspections done by the Structures Monitoring program are being credited for ensuring the fire barrier intended function of the elastomeric penetration seals is maintained during the subsequent period of extended operation, then include additional information (e.g., “detection of aging effects,” “acceptance criteria,” and “corrective actions” program elements for elastomeric fire barriers in the Fire Protection program are bounded by the corresponding program elements in the Structures Monitoring program associated with flood barriers).

Response to RAI B2.1.15-2:

Request 1:

As noted in the implementing procedures, Armaflex is approved for use as a flood barrier in limited applications (i.e., the East Penetration room floors). Armaflex does not serve a fire barrier function. The SLRA is amended to correct this discrepancy.

Request 2:

The Armaflex is not a credited fire barrier. The Armaflex does not serve a fire barrier function, therefore no applicable aging effects for the Armaflex associated with fire barrier function exist. The Structures Monitoring program is not credited for ensuring the intended function of fire barrier materials.

SLRA Revisions:

SLRA Section 2.4.8.2 (page 2-340 as revised by Supplement 3 (ML21349A005)) is revised as follows:

System Description

Fire barriers are located in safety and non-safety buildings to protect equipment within the scope of SLR from fire. Armaflex is a flexible insulation material which is installed in penetrations ~~in the floor and ceiling~~ of east and west penetration rooms (of each unit) **but is not credited as a fire barrier.** ~~for pressure boundary conditions. The NRC has accepted Armaflex as a fire barrier and has exempted it from testing and rating requirements normally required for fire barriers.~~

ENCLOSURE 1

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ATTACHMENT 18
RAI B2.1.10-1

Enclosure 1, Attachment 18

RAI B2.1.10-1:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Oconee Subsequent License Renewal Application (SLRA) Section 2.3.1.4 states that the evaluation boundary for the steam generator (SG) components includes “mechanical sleeves.”

In addition, SLRA Supplement 1, dated October 28, 2021 (ADAMS Accession No. ML21302A208), revised SLRA Section B2.1.10, “Steam Generators,” to clarify that continued acceptability evaluations will be performed during the subsequent period of extended operation for “steam generator components such as tubes, plugs, secondary side components, sleeves, tube supports, primary side cladding of heads (interior surfaces), tubesheets and tube-to-tubesheet welds.”

Issue:
Sleeving is not an NRC-approved repair method at Oconee; therefore, it is unclear to the NRC staff why the SLRA refers to “mechanical sleeves” and “sleeves.”

Request:
Discuss why the SLRA refers to “mechanical sleeves” and “sleeves,” in relation to steam generator components or alternatively revise SLRA Sections 2.3.1.4 and B2.1.10 to remove reference to “mechanical sleeves” and “sleeves,” respectively.

Response to RAI B2.1.10-1:

ONS SLRA Section 2.3.1.4 is being revised to remove the reference to “mechanical sleeves” and Appendix B2.1.10 is being revised to remove the reference to “sleeves”.

SLRA Revisions:

SLRA Section 2.3.1.4 (page 2-66) is being revised as follows:

System Evaluation Boundary

The evaluation boundary for the steam generator components includes the hemispherical heads, secondary shell, tubes, plugs, ~~mechanical sleeves,~~ tubesheets, primary nozzles, main and auxiliary feedwater nozzles, steam outlet nozzles, instrumentation nozzles, main and auxiliary feedwater nozzles, steam outlet nozzles, instrumentation nozzles, all associated pressure retaining bolting, and integral attachments. The main and auxiliary feedwater headers and riser piping are non-Class 1 items and are addressed in Section 2.3.4.

SLRA Appendix B2.1.10 (page B-93 and Supplement 1 (ML21302A208)) is being revised as follows:

Depending on the results of the visual examinations, more detailed inspections may be performed. A condition report is generated in the corrective action program to address any indications of degradation on steam generator components. Steam generator components such as, degraded plugs, tube-to-tubesheet welds, heads (interior surfaces), tubesheets (primary side), and secondary side internals are evaluated for continued acceptability on a case-by-case basis as part of the ONS Steam Generators aging management program. Similarly, inspection/maintenance activity events that potentially damage tubesheet and tube-to-tubesheet weld primary side surfaces are also addressed on a case-by-case basis through entry into the corrective action program. Evaluations are performed using industry guidance contained in NEI 97-06 and the associated EPRI Steam Generator Guidelines. These evaluations for continued acceptability will be performed during the subsequent period of extended operation for steam generator components such as, tubes, plugs, secondary side components, ~~sleeves,~~ tube supports, primary side cladding of heads (interior surfaces), tubesheets and tube-to-tubesheet welds.

ENCLOSURE 1

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ATTACHMENT 19
RAI B2.1.10-2

Enclosure 1, Attachment 19

RAI B2.1.10-2:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
SLRA Supplement 1, dated October 28, 2021 (ADAMS Accession No. ML21302A208), revised SLRA Table 3.1.2-4 by adding two different items under Plant Specific Note 1, which state: “The environment of Treated Water is equal to the environment of Secondary Feedwater for the Tube Support Plate Assembly (support rods),” and “The Auxiliary Feedwater Nozzle Flanges are insulated with stainless steel metal reflective insulation (Reference: Drawing OM 241-37 Sheet 1 and OSS-0241.00-00-0005).” In addition, SLRA Supplement 3, dated December 15, 2021 (ADAMS Accession No. ML21349A005), revised SLRA Table 3.1.2-4 by adding another Plant Specific Note 1, which states, “The environment of Secondary Feedwater is considered the same as Treated Water for the Steam Generator components.”

The NRC staff notes that SLRA Supplement 1 also included a Plant Specific Note 2.

Issue:
Based on SLRA Supplements 1 and 3, there will be three different items identified as Plant Specific Note 1.

Request:
Revise SLRA Table 3.1.2-4 to correct the numbering of the plant specific notes and, if necessary, revise the language where they are referenced in the SLRA Table 3.1.2-4.

Response to RAI B2.1.10-2:

The ONS SLRA is being revised to correct the numbering of the plant specific notes and references on all applicable pages of SLRA Table 3.1.2-4.

SLRA Revisions:

SLRA Table 3.1.2-4 (page 3-193) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Auxiliary Feedwater Nozzle Flanges	Pressure Boundary	Stainless Steel	Air – Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B2.1.20)	VIII.H.S.452b	3.4.1-104	A**, 2**
				Loss of Material	One-Time Inspection (B2.1.20)	VIII.H.S.451b	3.4.1-103	A**, 2**
			Air with Borated Water Leakage (External)	None	None	IV.E.RP-05	3.1.1-107	A
			Treated Water (Internal)	Cracking	One-Time Inspection (B2.1.20)	None	None	~~A~~ **F**
					Water Chemistry (B2.1.2)	None	None	~~A~~ **F**
				~~Cumulative Fatigue Damage~~	~~TLAA~~	~~IV.C2.R-18~~	~~3.1.1-005~~	~~C~~
				Loss of Material	One-Time Inspection (B2.1.20)	VIII.G.SP-87	3.4.1-085	A
					Water Chemistry (B2.1.2)	VIII.G.SP-87	3.4.1-085	A
			Steam (Internal)	**Cumulative Fatigue Damage**	**TLAA**	**IV.C2.R-18**	**3.1.1-005**	**C**

SLRA Table 3.1.2-4 (page 3-194) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Auxiliary Feedwater Nozzle Inlet Header	Pressure Boundary	Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	VIII.H.S.29	3.4.1-034	A
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.4.1-049	A
			Treated Water (Internal)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	A, 3
				Loss of Material	One-Time Inspection (B2.1.20)	VIII.G.SP-74	3.1.1-014	A
					Water Chemistry (B2.1.2)	VIII.G.SP-74	3.1.1-014	A
				Wall Thinning	Flow-Accelerated Corrosion (B2.1.8)	IV.D2.R-38	3.1.1-061	A

SLRA Table 3.1.2-4 (page 3-195) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Baffle Assemblies	Flow Distribution	Steel	Secondary Feedwater (External)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-162	3.1.1-072	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-162	3.1.1-072	~~A~~ C
	Structural Support	Steel	Secondary Feedwater (External)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-162	3.1.1-072	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-162	3.1.1-072	~~A~~ C

SLRA Table 3.1.2-4 (page 3-196) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Main Feedwater Nozzle Inlet Headers	Pressure Boundary	Steel	Secondary Feedwater (Internal)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	A
				Loss of Material	One-Time Inspection (B2.1.20)	IV.D2.RP-153	3.1.1-083	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-153	3.1.1-083	~~A~~ C
				Wall Thinning	Flow-Accelerated Corrosion (B2.1.8)	IV.D2.R-38	3.1.1-061	A

SLRA Table 3.1.2-4 (page 3-197) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Primary Manway and Inspection Opening Covers and Backing Plates	Pressure Boundary	Nickel Alloy	Reactor Coolant (Internal)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.RP-47	3.1.1-042	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-47	3.1.1-042	~~A~~ C
				Cumulative Fatigue Damage	TLAA	IV.D2.R-222	3.1.1-008	~~A~~ C
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	~~A~~ C

SLRA Table 3.1.2-4 (page 3-198) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Primary Manway and Inspection Opening Covers and Backing Plates	Pressure Boundary	Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	~~A~~ C
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
Primary Nozzles	Pressure Boundary	Steel (with Stainless Steel Cladding)	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	A
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
			Reactor Coolant (Internal)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.RP-47	3.1.1-042	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-47	3.1.1-042	~~A~~ C
				Cumulative Fatigue Damage	TLAA	IV.D2.R-222	3.1.1-008	A
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A

SLRA Table 3.1.2-4 (page 3-199) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Secondary Manway and Handhole Opening Covers	Pressure Boundary	Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	~~A~~ C
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
			Secondary Feedwater (Internal)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	~~A~~ C
				Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.R-31	3.1.1-044	A, 3
					One-Time Inspection (B2.1.20)	IV.D2.RP-153	3.1.1-083	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-153	3.1.1-083	~~A~~ C

SLRA Table 3.1.2-4 (page 3-200) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Secondary Side Nozzles (vent, drain, and instrumentation)	Pressure Boundary	Steel	Secondary Feedwater (Internal)	Loss of Material	One-Time Inspection (B2.1.20)	IV.D2.RP-153	3.1.1-083	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-153	3.1.1-083	~~A~~ C
Shell Assembly	Pressure Boundary	Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	~~A~~ C
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
			Secondary Feedwater (Internal)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	A
				Loss of Material	One-Time Inspection (B2.1.20)	IV.D2.RP-153	3.1.1-083	A
					Water Chemistry (B2.1.2)	IV.D2.RP-153	3.1.1-083	A

SLRA Table 3.1.2-4 (page 3-201) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Steam Outlet Nozzle	Pressure Boundary	Steel	Steam (Internal)	Loss of Material	One-Time Inspection (B2.1.20)	IV.D2.RP-153	3.1.1-083	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-153	3.1.1-083	~~A~~ C
				~~Wall Thinning~~	~~Flow-Accelerated Corrosion (B2.1.8)~~	~~IV.D2.R-38~~	~~3.1.1-061~~	~~A~~
Tube Plugs	Pressure Boundary	Nickel Alloy	Reactor Coolant (External)	Cracking	Steam Generators (B2.1.10)	IV.D2.R-40	3.1.1-070	A
					Water Chemistry (B2.1.2)	IV.D2.R-40	3.1.1-070	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-46	3.1.1-002	~~A~~ C
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	~~A~~ C
Tube Support Plate Assembly (spacers, nuts, keys, and wedges)	Structural Support	Steel	Secondary Feedwater (External)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-162	3.1.1-072	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-162	3.1.1-072	~~A~~ C
Tube Support Plate Assembly (support rods)	Structural Support	Stainless Steel	Secondary Feedwater (External)	Cracking	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ C

SLRA Table 3.1.2-4 (page 3-202) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Tube Support Plate Assembly (support rods)	Structural Support	Stainless Steel	Secondary Feedwater (External)	Cracking	Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ **C**
			Treated Water (External)	Cumulative Fatigue Damage	TLAA	IV.C2.R-18	3.1.1-005	C, 1
			Treated Water ~~(Internal)~~ **(External)**	Loss of Material	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 1~~ **C**
					Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 1~~ **C**
Tube Support Plate Assembly (tube support plates)	Structural Support	Stainless Steel	Secondary Feedwater (External)	Cracking	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ **C**
					Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ **C**
				Cumulative Fatigue Damage	TLAA	IV.C2.R-18	3.1.1-005	C
			~~Treated Water (Internal)~~ Secondary Feedwater (External)	Loss of Material	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 2~~ **C**
					Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 2~~ **C**

SLRA Table 3.1.2-4 (page 3-203) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Tubes	Pressure Boundary	Nickel Alloy	Reactor Coolant (Internal)	Cracking	Water Chemistry (B2.1.2)	IV.D2.R-44	3.1.1-070	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-46	3.1.1-002	A
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	~~A~~ C
			Secondary Feedwater (External)	Cracking	Steam Generators (B2.1.10)	IV.D2.R-442	3.1.1-125	A
						IV.D2.R-47	3.1.1-069	A
					Water Chemistry (B2.1.2)	IV.D2.R-47	3.1.1-069	A
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-233	3.1.1-077	A
Tubesheet	Pressure Boundary	Stainless Steel	Secondary Feedwater (External)	Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-162	3.1.1-072	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-162	3.1.1-072	~~A~~ C
		Steel with Nickel Alloy Cladding	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	~~A~~ C
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A

SLRA Table 3.1.2-4 (page 3-204) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Tubesheet	Pressure Boundary	Steel with Nickel Alloy Cladding	Reactor Coolant (Internal)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.RP-47	3.1.1-042	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-47	3.1.1-042	~~A~~ C
				Cumulative Fatigue Damage	TLAA	IV.D2.R-222	3.1.1-008	C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.R-440	3.1.1-127	A
					Water Chemistry (B2.1.2)	IV.D2.R-440	3.1.1-127	A
Tube-to-Tube Sheet Welds	Structural Support	Nickel Alloy	Reactor Coolant (External)	Cracking	Water Chemistry (B2.1.2)	IV.D2.RP-185	3.1.1-025	A
			Reactor Coolant ~~(Internal)~~ **(External)**	Cracking	Steam Generators (B2.1.10)	IV.D2.RP-185	3.1.1-025	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-46	3.1.1-002	~~A~~ C
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A
			Secondary Feedwater ~~(External)~~ **(Internal)**	Cracking	Steam Generators (B2.1.10)	IV.D2.R-47	3.1.1-069	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.R-47	3.1.1-069	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-233	3.1.1-077	~~A~~ C

SLRA Table 3.1.2-4 (page 3-205) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Upper and Lower Heads	Pressure Boundary	Steel (with Stainless Steel Cladding)	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	V.E.E-44	3.2.1-040	A
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
			Reactor Coolant (Internal)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.RP-47	3.1.1-042	A
					Water Chemistry (B2.1.2)	IV.D2.RP-47	3.1.1-042	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-222	3.1.1-008	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.R-440	3.1.1-127	A
					Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A
						IV.D2.R-440	3.1.1-127	A

Plant Specific Notes:

~~None~~ **1. The environment of Treated Water is equal to the environment of Secondary Feedwater for the Tube Support Plate Assembly (support rods).**
2. The Auxiliary Feedwater Nozzle Flanges are insulated with stainless steel metal reflective insulation (Reference: Drawing OM 241-37 Sheet 1 and OSS-0241.00-00-0005).
3. The environment of Secondary Feedwater is considered the same as Treated Water for the Steam Generator components.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 20
RAI B2.1.10-3

Enclosure 1, Attachment 20

RAI B2.1.10-3:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
The NRC staff has questions regarding the use of the Industry Standard Notes in SLRA Table 3.1.2-4 for the following:

- Aging Management Review (AMR) item 3.1.1-005 for managing cumulative fatigue damage of the steel auxiliary feedwater nozzle inlet header exposed internally to treated water (Note A).
- AMR item 3.1.1-083 for managing loss of material of the steel secondary manway and handhole opening covers exposed internally to secondary feedwater (Note A).
- Cracking of stainless steel tube support plate assembly (support rods) exposed externally to secondary feedwater (Note E).
- Loss of material of the stainless steel tube support plate assembly (support rods) and tube support plate assembly (tube support plates) exposed externally and internally, respectively, to treated water (Note J).
- Cracking of stainless steel tube support plate assembly (tube support plates) exposed externally to secondary feedwater (Note E).
- AMR item 3.1.1-072 for managing loss of material of the steel tubesheet exposed externally to secondary feedwater (Note A).

Issue:
AMR item 3.1.1-005 in Volume 1 of NUREG-2191, "Generic Aging Lessons Learned for subsequent License Renewal (GALL-SLR) Report" (ADAMS Accession No. ML17187A031), manages cumulative fatigue damage of steel steam generator components exposed to secondary feedwater or steam. However, as noted above, the environment for the steel auxiliary feedwater nozzle inlet header is treated water (internal). It is unclear to the NRC staff that Industry Standard Note A is correct since its use means, in part, that the environment is consistent with the GALL-SLR. The staff notes that SLRA Supplement 3, dated December 15, 2021 (ADAMS Accession No. ML21349A005), revised SLRA Table 3.1.2-4 by adding a Plant Specific Note 1, which states, "The environment of Secondary Feedwater is considered the same as Treated Water for the Steam Generator components."

AMR item 3.1.1-083 in the GALL-SLR manages loss of material of steel steam generator components: shell assembly exposed to secondary feedwater or steam. However, as noted above, the component is steel secondary manway and handhole opening covers exposed internally to secondary feedwater. It is unclear to the NRC staff that Industry Standard Note A is correct since its use means, in part, that the component is consistent with the GALL-SLR.

SLRA Supplement 3 revised SLRA Table 3.1.2-4 to cite AMR item 3.1.1-071 to manage cracking of stainless steel tube support plate assembly (support rods) exposed externally to secondary feedwater by the Steam Generators program and changed the Industry Standard Note from E to C. AMR item 3.1.1-071 in the GALL-SLR manages cracking by the Steam Generators and Water Chemistry programs. However, SLRA Table 3.1.2-4 was not revised to make similar changes to the aging management evaluation item for managing cracking of stainless steel tube support plate assembly (support rods) exposed externally to secondary feedwater by the Water Chemistry program, which also cites Industry Standard Note E.

SLRA Table 3.1.2-4 cites Industry Standard Note J for managing loss of material of the tube support plate assembly (support rods) and tube support plate assembly (tube support plates) exposed externally and internally, respectively, to treated water by the Steam Generators and Water Chemistry programs. Industry Standard Note J is defined in the SLRA as “Neither the component nor the material and environment combination is evaluated in NUREG-2191.” However, the use of Industry Standard Note J is unclear because the NRC staff notes that AMR item 3.1.1-071 (IV.D1.RP-226) manages loss of material of stainless steam generator structural: U-bend supports including anti-vibration bars exposed to secondary feedwater or steam by the Steam Generators and Water Chemistry programs. The staff notes that Supplement 1, dated October 28, 2021 (ADAMS Accession No. ML21302A208), revised SLRA Table 3.1.2-4 by adding plant specific notes that indicate that the environment of treated water is equal to the environment of secondary feedwater for these components. In addition, the staff notes, citing AMR item 3.1.1-071 with an Industry Standard Note C for managing cracking would be similar to citing ARM item 3.1.1-071 for managing loss of material as discussed above.

SLRA Table 3.1.2-4 cites the Steam Generators and Water Chemistry programs to manage cracking of stainless steel tube support plate assembly (tube support plates) exposed externally to secondary feedwater and cites Industry Standard Note E. The NRC staff notes that SLRA Supplement 3 revised SLRA Table 3.1.2-4 to cite AMR item 3.1.1-071 to manage cracking of stainless steel tube support plate assembly (support rods) exposed externally to secondary feedwater by the Steam Generators program and changed the Industry Standard Note from E to C. Therefore, it is unclear to the NRC staff why no AMR item was cited for managing cracking of the stainless steel tube support plate assembly (tube support plates) and why Industry Standard Note E is cited.

AMR item 3.1.1-072 in GALL-SLR manages loss of material of steel steam generator: tube bundle wrapper and associated supports and mounting hardware exposed to secondary feedwater or steam by the Steam Generators and Water Chemistry programs. SLRA Supplement 3 revised SLRA Table 3.1.2-4 to change Industry Standard Note A to C for the steel tubesheet since it is a different component than the component for AMR item 3.1.1-072. However, SLRA Supplement 3 only made this change for the Steam Generators program.

<u>Request:</u>

1. Given that the environment for the steel auxiliary feedwater nozzle inlet header is different than the environments for AMR item 3.1.1-005, please discuss the use of Industry Standard Note A, or alternatively, revise SLRA Table 3.1.2-4 to cite an industry standard note that indicates differences with the GALL-SLR or reference Plant Specific Note 1 that was added in Supplement 3.

2. Given that steel secondary manway and handhole opening covers is different than the component for AMR item 3.1.1-083, please discuss the use of Industry Standard Note A or alternatively revise SLRA Table 3.1.2-4 to use Industry Standard Note C.

3. Discuss why AMR item 3.1.1-071 was not cited for managing cracking of stainless steel tube support plate assembly (support rods) exposed externally to secondary feedwater by the Water Chemistry program, and discuss the use of Industry Standard Note E. Alternatively, revise SLRA Table 3.1.2-4 to cite AMR item 3.1.1-071 and Industry Standard Note C.

4. Discuss the use of Industry Standard Note J for loss of material of the stainless steel tube support plate assembly (support rods) and tube support plate assembly (tube support plates) exposed externally and internally, respectively, to treated water. Alternatively, revise SLRA Table 3.1.2-4 to cite AMR item 3.1.1-071 (IV.D1.RP-226) and Industry Standard Note C.

5. Discuss why no AMR item is cited for managing cracking of the stainless steel tube support plate assembly (tube support plates) and why Industry Standard Note E is cited. Alternatively, revise SLRA Table 3.1.2-4 to cite AMR item 3.1.1-071 (IV.D1.RP-384) and Industry Standard Note C.

6. Discuss the use of Industry Standard Note A for AMR item 3.1.1-072 cited for managing loss of material of the steel tubesheet exposed externally to secondary feedwater by the Water Chemistry program. Alternatively, revise SLRA Table 3.1.2-4 to use Industry Standard Note C.

Response to RAI B2.1.10-3:

Request 1:

As noted by the staff, the internal environment for the Auxiliary Feedwater Nozzle Inlet Header in the ONS SLRA, Table 3.1.2-4 is Treated Water. In order to use Industry Standard Note A, a plant specific note is being added to AMR line item 3.1.1-005 (page 3-194 and page 3-205) for the steel Auxiliary Feedwater Nozzle Inlet Header to describe that the environment of Treated Water is equal to that of Secondary Feedwater. This note (Plant Specific Note 3) clarifies that this GALL-SLR AMR line item IV.D2.R-33 (SRP Item 3.1.1-005) line item is consistent with GALL.

Request 2:

SLRA Table 3.1.2-4 (page 3-199) is being revised to use Industry Standard Note C for the Secondary Manway and Handhole Opening Covers AMR item 3.1.1-083 for the *One-Time Inspection* and *Water Chemistry* programs.

Request 3:

SLRA Table 3.1.2-4 (page 3-202) is being revised to cite AMR item 3.1.1-071 and Industry Standard Note C for managing cracking of the stainless steel Tube Support Plate Assembly (support rods) exposed externally to secondary feedwater by the *Water Chemistry* program.

Request 4:

SLRA Table 3.1.2-4 (page 3-202) is being revised to cite AMR item 3.1.1-071 (IV.D1.RP-226) and Industry Standard Note C for loss of material of the stainless steel Tube Support Plate Assembly (support rods) and Tube Support Plate Assembly (tube support plates) for the *Steam Generators* and *Water Chemistry* programs. Note that a correction was also made to the environment for these line items.

Request 5:

SLRA Table 3.1.2-4 (page 3-202) is revised to cite AMR item 3.1.1-071 (IV.D1.RP-384) and Industry Standard Note C for managing cracking of the stainless steel Tube Support Plate Assembly (support rods) by the *Water Chemistry* program.

Request 6:

SLRA Table 3.1.2-4 (page 3-203) is revised to use Industry Standard Note C for AMR item 3.1.1-072 cited for managing loss of material of the steel Tubesheet exposed externally to secondary feedwater by the *Water Chemistry* program.

SLRA Revisions:

SLRA Table 3.1.2-4 revisions are shown in their entirety in response to RAI B2.1.10-2 (Enclosure 1, Attachment 19). Specifically, pages 3-194, 3-199, 3-202, 3-203, and 3-205 are repeated here for ease of review.

SLRA Table 3.1.2-4 (page 3-194) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Auxiliary Feedwater Nozzle Inlet Header	Pressure Boundary	Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	VIII.H.S.29	3.4.1-034	A
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.4.1-049	A
			Treated Water (Internal)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	A, 3
				Loss of Material	One-Time Inspection (B2.1.20)	VIII.G.SP-74	3.1.1-014	A
					Water Chemistry (B2.1.2)	VIII.G.SP-74	3.1.1-014	A
				Wall Thinning	Flow-Accelerated Corrosion (B2.1.8)	IV.D2.R-38	3.1.1-061	A

SLRA Table 3.1.2-4 (page 3-199) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Secondary Manway and Handhole Opening Covers	Pressure Boundary	Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	~~A~~ C
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
			Secondary Feedwater (Internal)	Cumulative Fatigue Damage	TLAA	IV.D2.R-33	3.1.1-005	~~A~~ C
				Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.R-31	3.1.1-044	A, 3
					One-Time Inspection (B2.1.20)	IV.D2.RP-153	3.1.1-083	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-153	3.1.1-083	~~A~~ C

SLRA Table 3.1.2-4 (page 3-202) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Tube Support Plate Assembly (support rods)	Structural Support	Stainless Steel	Secondary Feedwater (External)	Cracking	Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ **C**
			Treated Water (External)	Cumulative Fatigue Damage	TLAA	IV.C2.R-18	3.1.1-005	C, 1
			Treated Water ~~(Internal)~~ **(External)**	Loss of Material	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 1~~ **C**
					Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 1~~ **C**
Tube Support Plate Assembly (tube support plates)	Structural Support	Stainless Steel	Secondary Feedwater (External)	Cracking	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ **C**
					Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-384**	~~None~~ **3.1.1-071**	~~E~~ **C**
				Cumulative Fatigue Damage	TLAA	IV.C2.R-18	3.1.1-005	C
			~~Treated Water (Internal)~~ Secondary Feedwater (External)	Loss of Material	Steam Generators (B2.1.10)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 2~~ **C**
					Water Chemistry (B2.1.2)	~~None~~ **IV.D1.RP-226**	~~None~~ **3.1.1-071**	~~J, 2~~ **C**

SLRA Table 3.1.2-4 (page 3-203) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Tubes	Pressure Boundary	Nickel Alloy	Reactor Coolant (Internal)	Cracking	Water Chemistry (B2.1.2)	IV.D2.R-44	3.1.1-070	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-46	3.1.1-002	A
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	~~A~~ C
			Secondary Feedwater (External)	Cracking	Steam Generators (B2.1.10)	IV.D2.R-442	3.1.1-125	A
						IV.D2.R-47	3.1.1-069	A
					Water Chemistry (B2.1.2)	IV.D2.R-47	3.1.1-069	A
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-233	3.1.1-077	A
Tubesheet	Pressure Boundary	Stainless Steel	Secondary Feedwater (External)	Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-162	3.1.1-072	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-162	3.1.1-072	~~A~~ C
		Steel with Nickel Alloy Cladding	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	~~A~~ C
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A

SLRA Table 3.1.2-4 (page 3-205) is being revised as follows:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Upper and Lower Heads	Pressure Boundary	Steel (with Stainless Steel Cladding)	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	V.E.E-44	3.2.1-040	A
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.D2.R-17	3.1.1-049	A
			Reactor Coolant (Internal)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.RP-47	3.1.1-042	A
					Water Chemistry (B2.1.2)	IV.D2.RP-47	3.1.1-042	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-222	3.1.1-008	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.R-440	3.1.1-127	A
					Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A
						IV.D2.R-440	3.1.1-127	A

Plant Specific Notes:

~~None~~ **1. The environment of Treated Water is equal to the environment of Secondary Feedwater for the Tube Support Plate Assembly (support rods).**
2.The Auxiliary Feedwater Nozzle Flanges are insulated with stainless steel metal reflective insulation (Reference: Drawing OM 241-37 Sheet 1 andOSS-0241.00-00-0005).
3.The environment of Secondary Feedwater is considered the same as Treated Water for the Steam Generator components.

ENCLOSURE 1

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ATTACHMENT 21
RAI B2.1.10-4

Enclosure 1, Attachment 21

RAI B2.1.10-4:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
SLRA Table 3.1.2-4 includes an aging management evaluation for nickel alloy and steel primary manway and inspection opening covers and backing plates. During the audit of the SG program, Duke Energy Carolinas, LLC, clarified that the primary manway and inspection opening covers and backing plates are steel with nickel alloy cladding.

Issue:
The AMR items cited for the nickel alloy primary manway and inspection opening covers and backing plates are for steel (with stainless steel or nickel alloy cladding) primary side components: upper and lower heads, and tubesheet welds. While SLRA Supplement 2, dated November 11, 2021 (ADAMS Accession No. ML21315A012), revised SLRA Table 3.1.2-4 to cite Industry Standard Note C for the AMR items cited for the nickel alloy primary manway and inspection opening covers and backing plates, the material does not appear to be accurately reflected for these components or to be consistent with the GALL-SLR.

AMR items 3.1.1-124 and 3.1.1-049 cited for the steel primary manway and inspection opening covers and backing plates are for steel piping and piping components and for external surfaces of steel once-through SG components, respectively. While SLRA Supplement 2 also revised SLRA Table 3.1.2-4 to cite Industry Standard Note C for AMR item 3.1.1-124, the material does not appear to be accurately reflected for these components or to be consistent with the GALL-SLR. No changes were included in any supplements for AMR item 3.1.1-049. The NRC staff notes that Table 3.1.2-4 states that other components are clad.

Request:
Revise SLRA Table 3.1.2-4 to indicate that the material of the primary manway and inspection opening covers and backing plates is “steel (with nickel alloy cladding)” or explain why this revision is not appropriate.

Response to RAI B2.1.10-4:

The ONS Units 1, 2, and 3 primary manway and inspection covers are constructed of SA-533 Type B Class 1 (steel). The backing plates for the ONS Units 1, 2, and 3 primary manway and inspection covers (diaphragms) are constructed of SB-168 UNS N06690 (nickel alloy). None of these components contain cladding material.

The configuration of the primary manway and inspection opening consists of the backing plate, or diaphragm, being placed in the opening and the cover bolted on top of the backing plate. The backing plate, or diaphragm, is in contact with the reactor coolant and the cover is exposed to an air – indoor uncontrolled environment.

Therefore, ONS SLRA Table 3.1.2-4 shows the component “Primary Manway and Inspection Opening Covers and Backing Plates” with the correct material and environment combination. Table 3.1.2-4 does not require revision.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 22
RAI 2.3.1.3-1

Enclosure 1, Attachment 22

RAI 2.3.1.3-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations,* Part 54, "Requirements for renewal of operating licenses for nuclear power plants," is designed to elicit application information that will enable the NRC staff to perform an adequate safety review and the Commission to make the necessary findings. Reliability of application information is important and advanced by requirements that license applications be submitted in writing under oath or affirmation and that information provided to the NRC by a license renewal applicant or requirement to be maintained by NRC regulations be complete and accurate in all material respects. Information that must be submitted in writing under oath or affirmation includes the technical information required under 10 CFR 54.21(a) related to assessment of the aging effects on structures, systems, and components subject to an aging management review. Thus, both the general submission requirements for license renewal applications and the specific technical application information requirements require that submission of information material to NRC's safety findings (see 10 CFR 54.29, "Standards for issuance of a renewed license") be submitted by an applicant as part of the application.

Background:

By letter dated June 7, 2021, Duke Energy Carolinas, LLC (Duke Energy) submitted to the U.S. Nuclear Regulatory Commission (NRC or staff) an application to renew the Renewed Facility Operating License for the Oconee Power Station, Units 1, 2 and 3 licenses pursuant to Section 104b of the Atomic Energy Act of 1954, as amended, and Part 54 of Title 10 of the *Code of Federal Regulations,* Part 54, "Requirements for renewal of operating licenses for nuclear power plants."

Issue:
In SLRA Section 2.3.1.3, Reactor Coolant System: SLRA Table 2.3.1-3 – Reactor Coolant System – Pressurizer, the intended function for the listed components is to form part of the pressure boundary. However, thermal cycling is not included as an intended function for following components:

- Pressurizer; Surge Line Nozzle.
- Pressurizer; Surge Line Nozzle Safe End.
- Pressurizer; Surge Line Nozzle Safe End Weld.

Request:
Explain whether thermal cycling should be included as an intended function for these components. If not, provide justification.

Response to RAI 2.3.1.3-1:

Thermal cycling should not be included as an intended function for the Pressurizer Surge Line Nozzle, Surge Line Nozzle Safe End, or Surge Line Nozzle Safe End Weld.

NEI 95-10, Industry Guidelines for Implementing The Requirements of 10 CFR Part 54 – The License Renewal Rule, describes that intended functions define the plant process, condition or action that must be accomplished in order to perform or support a safety function for responding to a design basis event or to perform or support a specific requirement of one of the five regulated events in 10 CFR Part 54.4(a)(3). At a system level, the intended functions may be thought of as the functions of the system that are the bases for including this system within the scope of license renewal as specified in 10 CFR Part 54.4(a)(1) - (3).

NUREG-2192, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants, lists in Table 2.1-5 the following typical passive component-intended functions: Absorb Neutrons, Electrical Continuity, Filter, Heat Transfer, Insulate (electrical), Insulate (thermal), Leakage Boundary (spatial), Pressure Boundary, Spray, Structural Integrity (attached), Structural Support, and Throttle.

NUREG-2191, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report, describes Thermal Cycling as an Aging Effect/Mechanism.

SLRA Revisions:

None

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ATTACHMENT 23
RAI 2.3.1.3-2

Enclosure 1, Attachment 23

RAI 2.3.1.3-2:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations,* Part 54, "Requirements for renewal of operating licenses for nuclear power plants," is designed to elicit application information that will enable the NRC staff to perform an adequate safety review and the Commission to make the necessary findings. Reliability of application information is important and advanced by requirements that license applications be submitted in writing under oath or affirmation and that information provided to the NRC by a license renewal applicant or requirement to be maintained by NRC regulations be complete and accurate in all material respects. Information that must be submitted in writing under oath or affirmation includes the technical information required under 10 CFR 54.21(a) related to assessment of the aging effects on structures, systems, and components subject to an aging management review. Thus, both the general submission requirements for license renewal applications and the specific technical application information requirements require that submission of information material to NRC's safety findings (see 10 CFR 54.29, "Standards for issuance of a renewed license") be submitted by an applicant as part of the application.

Background:

By letter dated June 7, 2021, Duke Energy Carolinas, LLC (Duke Energy) submitted to the U.S. Nuclear Regulatory Commission (NRC or staff) an application to renew the Renewed Facility Operating License for the Oconee Power Station, Units 1, 2 and 3 licenses pursuant to Section 104b of the Atomic Energy Act of 1954, as amended, and Part 54 of Title 10 of the *Code of Federal Regulations,* Part 54, "Requirements for renewal of operating licenses for nuclear power plants."

Issue:
Regarding SLRA Section 2.3.1.3, Reactor Coolant System: SLRA Table 2.3.1-3 - Reactor Coolant System – Pressurizer, if the spray head meets any one of the situations as described below, it may require the inclusion of the pressurizer spray head in the scope of license renewal:

a. During fire events as required by 10 CFR 50 Appendix R evaluation, the pressurizer spray head is used to achieve the reactor cooldown to meet the Technical Specifications LCO 3.4.3.
b. If the spray head is failed, it will damage the surrounding safety-related components.

Request:
Explain if the pressurizer spray head is excluded from the scope of license renewal and provide justification by specifically addressing the related concerns presented in Table 2.3-1 of the Standard Review Plan (NUREG-2192).

Response to RAI 2.3.1.3-2:

The pressurizer spray head is included with the scope of SLR. As indicated in Table 2.3.1-3 of the SLRA, the pressurizer spray head is subject to AMR and has a pressure boundary intended function.

Table 2.3-1 of the Standard Review Plan (NUREG-2192) specifically addresses the following scoping considerations for pressurizer spray head components: 1) pressure control to achieve cold shutdown during certain fire events, and 2) damage to surrounding safety grade components in the event of a failure of the spray head. Each consideration is addressed below:

- The pressurizer spray head is not relied upon for compliance with Fire Protection (50.48) regulations. The pressurizer power operated relief valve (PORV) is relied upon for RCS pressure control for meeting NFPA 805 requirements. Therefore, the pressurizer spray head does not meet the 10CFR54.4(a)(3) scoping for the Fire Protection (50.48) (or any other regulated event).
- Analyses performed for a complete failure of the pressurizer spray head and spray line and damage due to impact on surrounding safety related components and for loose parts generated from the spray line concluded that damage will not result in a loss of a function as described in parts (i), (ii), or (iii) of 10CFR 54.4(a)(1). Therefore, the pressurizer spray line (including the spray head) does not meet the 10CFR54.4(a)(2) scoping criteria.

SLRA Revisions:

None

ENCLOSURE 1

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ATTACHMENT 24
RAI B2.1.11-1

Enclosure 1, Attachment 24

RAI B2.1.11-1:

Regulatory Basis
Title 10 of the *Code of Federal Regulations* Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. Title 10 of the *Code of Federal Regulations* Section 54.21(d) requires that the FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by 10 CFR 54.21(a) and (c), respectively. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
SRP-SLR:
The SRP-SLR discusses how applicants meet the 10 aging management program elements in Appendix A.1.2.3, "Aging Management Program Elements."

SRP-SLR Appendix A.1.2.3.9, "Administrative Controls" states:

2. Administrative controls are addressed through the QA program that is used to meet the requirements of 10 CFR Part 50, Appendix B, associated with managing the effects of aging (e.g., document control, special processes, and test control). Appendix A.2 describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the administrative controls element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program.

SRP-SLR Appendix A.2:

Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1) BTP (IQMB-1) describes an acceptable process for implementing the corrective actions, the confirmation process, and administrative controls of aging management programs for SLR.

SRP-SLR A.2.2 Branch Technical Position:

2. For nonsafety-related SCs that are subject to an AMR for SLR, an applicant has the option to expand the scope of its 10 CFR Part 50 Appendix B program to include these SCs and to address corrective actions, the confirmation process, and administrative controls for aging management during the subsequent period of extended operation. The reviewer verifies that the applicant has documented such a commitment in the Final Safety Analysis Report supplement in accordance with 10 CFR 54.21(d).

3. If an applicant chooses an alternative means to address corrective actions, the confirmation process, and administrative controls for managing aging of nonsafety-related SCs that are

subject to an AMR for SLR, the applicant's proposal is reviewed on a case-by-case basis following the guidance in BTP RLSB-1 (Appendix A.1 of this SRP-SLR).

GALL-SLR:
An example summary program description of the QA program for the FSAR supplement is shown in Table A-01 below.

Table A-01. FSAR Supplement Summary for Quality Assurance Programs for Aging Management Programs

GALL-SLR AMP	**GALL-SLR Program**	**Description of Program**	**Implementation Schedule**
GALL-SLR Appendix A	Quality Assurance	The QA program, developed in accordance with the requirements of 10 CFR Part 50, Appendix B, provides the basis for the corrective actions, confirmation process, and administrative controls elements of AMPs. The scope of this existing QA program is expanded to also include nonsafety-related SCs subject to AMPs.	Existing program

Oconee SLRA:

B2.1.11 Open-Cycle Cooling Water System:
Enhancement 7 to the Oconee *Open-Cycle Cooling Water System* AMP states that, “The *Open-Cycle Cooling Water System* AMP is an existing program that will be consistent with NUREG-2191 Section XI.M20, *Open-Cycle Cooling Water System* with enhancements and exceptions, as described below”:

7. Incorporate programmatic guidance contained in engineering support documents into controlled plant procedures subject to administrative controls in accordance with the Duke Energy QA Program. (Element 9)

Appendix A1.0:

Quality Assurance for Aging Management Programs

The Quality Assurance (QA) Program is described in Topical Report DUKE-QAPD-001-A, “Quality Assurance Program Description, Operating Fleet” which implements the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.” The QA Program is consistent with the summary in Appendix A.2, “Quality Assurance for AMPs (Branch Technical Position IQMB-1)” of NUREG-2192. The QA Program provides the basis for the corrective actions, confirmation process, and administrative controls elements of AMPs. The scope of the existing QA Program is expanded to include non-safety related structures and components that are subject to an AMR for LR. The QA Program is applicable to the safety related and non-safety related structures, components, and commodity groups that are subject to AMR.

Issue:
Although the SLRA states that the existing QA Program is expanded to include non-safety related structures and components that are subject to an AMR for LR, the statement only addresses license renewal (LR) and does not appear to be expanded to include subsequent license renewal (SLR) and, in addition, there is no such commitment listed in Table A6.0-1 of the Oconee SLRA.

Request:

1. Revise the description of the expanded QA Program in the SLRA to be applicable to SLR and include the commitment in Table A6.0-1 of the SLRA, or provide an alternative means to address corrective actions, the confirmation process, and administrative controls for managing aging of non-safety-related SCs that are subject to an AMR for SLR.

2. It is unclear to the staff why an enhancement to element 9 is needed for only the Open-Cycle Cooling Water System aging management program. Clarify the extent to which this enhancement is needed in the Oconee SLRA.

Response to RAI B2.1.11-1:

Request 1:

The discussion provided in Oconee SLRA Section Appendix A.0, specifically the statement: "The scope of the existing QA Program is expanded to include non-safety related structures and components that are subject to an AMR for LR" is revised. The use of "LR" in this statement is referring to the License Renewal Rule, 10 CFR Part 54. Accordingly, the SLRA discussions of application of Duke's QA program (SLRA Appendix A1.0 and SLRA Appendix B1.3) are revised for clarification purposes.

The QA Program described in Topical Report DUKE-QAPD-001-A, "Quality Assurance Program Description, Operating Fleet" is consistent with Appendix A.2, "Quality Assurance for AMPs (Branch Technical Position IQMB-1)" of NUREG-2192. Consequently, a commitment in Table A6.0-1 of the SLRA is not needed.

Request 2:

Enhancement 7 to Element 9 of the Open-Cycle Cooling Water System AMP addresses an issue unique to this particular AMP at Oconee. Certain aspects of the AMP that are relied upon to demonstrate consistency with NRC recommendations in GALL-SLR XI.M20 are defined in engineering support documents. Engineering support documents are intended to be informal working level tools for Oconee engineering staff and are not subject to the administrative controls defined by 10 CFR Part 50 Appendix B. Engineering support documents typically provide a summary of significant operating experience, plans for future enhancements, and the maintenance strategy, including additional guidance not included in procedures, and the basis for inspections and testing. As such, these documents aid engineering staff in ensuring AMP effectiveness through long-term monitoring and trending, and also help ensure institutional knowledge is maintained when a program is reassigned to a new engineer.

The intent of Enhancement 7 is to ensure that all attributes of the Open-Cycle Cooling Water System AMP relied upon to demonstrate consistency with GALL-SLR guidance are included within AMP implementing documents that are subject to 10 CFR Part 50 Appendix B administrative controls (e.g., procedures and preventive maintenance work orders). The engineering support documents related to implementation of the Open-Cycle Cooling Water System AMP are the only instance identified during ONS SLRA development where documents that are not subject to 10 CFR Part 50 Appendix B administrative controls are relied upon to demonstrate compliance with NRC guidance contained in

GALL-SLR. Therefore, this enhancement is only applicable to the Open-Cycle Cooling Water System AMP.

SLRA Revisions:

Request 1:

SLRA Appendix A1.0 (pages A-2 and A-3) is revised as follows:

Quality Assurance for Aging Management Programs

The Quality Assurance (QA) Program is described in Topical Report DUKE-QAPD-001-A,"Quality Assurance Program Description, Operating Fleet" which implements the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The QA Program is consistent with the summary in Appendix A.2, "Quality Assurance for AMPs (Branch Technical Position IQMB-1)" of NUREG-2192. The QA Program provides the basis for the corrective actions, confirmation process, and administrative controls elements of AMPs. The scope of the existing QA Program is expanded to include non-safety related structures and components that are subject to an AMR ~~for LR~~. The QA Program is applicable to the safety related and non-safety related structures, components, and commodity groups that are subject to AMR.

SLRA Appendix B1.3 (pages B-4) is revised as follows:

<u>Corrective Actions:</u>

Results that do not meet acceptance criteria are addressed as conditions adverse to quality or significant conditions adverse to quality under Sections 17.3.1.6, 17.3.2.13, and D17.3.2.13 *"Corrective Action,"* of DUKE-QAPD-001-A. The corrective action program is implemented in accordance with the requirements of 10 CFR 50, Appendix B, Criterion XVI and Topical Report DUKE-QAPD-001-A. A single program is used regardless of the safety classification of the structure or component. Conditions adverse to quality, such as failures, malfunctions, deviations, defective material and equipment, and non-conformances, are promptly identified and corrected. Corrective actions are implemented through the initiation of a nuclear condition report for actual or potential problems, correction of an equipment deficiency, or the need for corrective maintenance. Site documents that implement AMPs ~~for LR~~ direct that a nuclear condition report be prepared in accordance with those procedures whenever non-conforming conditions are found (i.e., the acceptance criteria are not met). The corrective action procedures specify steps for promptly reporting, evaluating, and correcting conditions adverse to quality and significant conditions adverse to quality commensurate with the significance of the SSC or activity. Consistent with the significance of the identified condition, these steps include: (1) deficiency identification, (2) deficiency review, impact on operations and reportability determination, (3) nuclear condition report review, trending and classification (including appropriate cause determination along with any warranted extent of condition and extent of cause), (4) corrective action determinations, assignments, and implementation, (5) assessment of effectiveness of correction, and (6) nuclear condition report closure.

Request 2:

None

ENCLOSURE 1

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ATTACHMENT 25
RAI 3.5.2.2.2.6-1
[NON-PROPRIETARY]

Enclosure 1, Attachment 25
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 3.5.2.2.2.6-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
SLRA Section 4.2, “Reactor Vessel Neutron Embrittlement Analyses,” indicate that the RV for Unit 1 is built with bent rolled ASTM A302B modified steel plates, while those for Units 2 and 3 are made from forged ASTM A 508 Class 2 steel plates. The report ANP-3898NP, Revision 0 (page 9-6 of enclosure 4 of the SLRA) states that the Class 1 RV steel support skirt is made of a SA-516 Grade 70 carbon-manganese steel. Figure 9-1, “Reactor Pressure Vessel Support Assembly,” of ANP-3898NP, Revision 0, shows that the skirt is attached to the RV transition forging dutchman by a circumferential full penetration weld.

It is apparent that the method of construction of ONS Unit 1 RV differs with those of Units 2 and 3. Casting methods of parent material, microstructural heterogeneities produced during manufacturing (e.g., rolled steel plates have microstructural changes in one direction, while those that are forged have it in two) and welding processes used could affect the performance (including fracture toughness) of support skirt to dutchman circumferential weldments. The NRC by letter dated January 7, 2016 (ADAMS Accession No. ML16004A262), granted a relief to ONS from performing a 100 percent VT-3 visual examination required by ASME Code, Section XI, Subsection IWF (Examination Category F-A in Table IWF-2500-1) for the Class 1 RV steel support skirt assembly that excludes from inspection the circumferential welds attaching the skirt to the RV dutchman for weld numbers 1-RPV-WR36, 2-RPV-WR36, and 3-RPV-WR36 as noted in the ONS relief request (ADAMS Accession No. ML15201A573). The weldments are identified to be within the scope of license renewal and subject to AMR. In accordance with 10 CFR 54.21(a)(3) an applicant must demonstrate 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.

Literature indicates that there is no distinct advantage or disadvantage in the serviceability of fabricated components from rolled or forged steel plates provided these have nearly identical chemistry, casting methods, and similar microstructure. Literature also indicates that loss of material due to corrosion can also contribute to reduction in fracture toughness by allowing intrusion of corrosive agents (e.g., boric acid) to ingress into metals and welds and altering constrains conditions at the tip of preexisting flaws or cracks.

Issue:
The mechanical performance of fabricated steel components and their weldments depend on a variety of parameters such as mode of construction, preexisting flaws, inservice aggressive environments

(e.g., radiation exposure, boric acid exposure, pitting, crevice corrosion) that could affect their ductility, fatigue, and fracture toughness. It is not clear what steps were taken during manufacturing to ensure that no detectable flaws and/or residual stresses existed on the RV dutchman to skirt weldments (HAZs and welds) that could lead to their loss of structural integrity for each of the three ONS RV Units.
The staff notes that there are no SLRA Table 3.5.2 AMR line items specific to the RV skirt to dutchman welds to manage the effects of aging for the subsequent period of extended operation. The staff is not aware of any operating experience (OE) and ASME Code Section XI inspection results attesting their condition prior to entering the period of subsequent license operation. It is not clear when weldments of skirt to dutchman were last physically inspected to ASME Code Section XI. Since the aforementioned relief from the ASME Code Section XI mandated inspection of the circumferential weld of the skirt assembly to the RV dutchman was limited to the Fourth Ten-Year Inservice Inspection (ISI) Interval for all three of the ONS RV Units, it is not clear how the applicant plans to establish a baseline for the condition of the aforementioned welds prior to entering the subsequent period of extended operation and what steps it plans to take to ensure their intended function is maintained to the end of the subsequent period of extended operation and confirm the validity of the fracture toughness evaluation in SLRA Section 3.5.2.2.2.6.

Request:

1. Discuss what steps were taken to rule out detectable flaws and/or residual stresses on the dutchman to support skirt weldments (HAZs and welds) during manufacturing of ONS Unit 1, 2, and 3 RVs that could contribute to weldments' loss of strength, loss of fracture toughness due to aggressive environments in the RV annulus cavity.
2. Describe steps to be taken (e.g., ongoing OE review, implementation of maintenance rule) to establish a baseline reflective of the condition of the RV dutchman to steel support skirt weldments and confirm that their intended function is maintained for each of the ONS RV Units prior to entering the subsequent period of extended operation.
3. Outline how aging effects such as loss of material and irradiation on RV dutchman to steel support skirt weldments will be adequately managed such that their intended function will be maintained to the end of the subsequent period of extended operation and provide or point to the applicable Table 3.5.2 AMR line items.
4. Update applicable SLRA sections, as applicable and necessary, consistent with the responses to the requests above.

Response to RAI 3.5.2.2.2.6-1:

Request 1:

As described in ANP-3898P/NP, Revision 0, Section 9.3, circumferential weld WR-36 connects the transition forging (fabricated from ASTM A508-64 Class 2) to the RV support skirt (fabricated from two SA-516 Grade 70 rolled plates) for all three Oconee Units. **[**

] **b,c,d,e**. Non-destructive testing requirements specified **[**

] b,c,d,e

For all three Oconee units, weld WR-36 was fabricated in accordance with ASME Section III (1965 Edition through Summer 1967 Addenda) requirements and corresponding B&W shop specifications that complied with all ASME Section III fabrication requirements. Fabrication defects in weld WR-36 were removed and repaired as discussed above. Residual stresses within the weld were reduced and redistributed owing to final reactor vessel shell PWHT. There are no fabrication deficiencies for weld WR-36 that could contribute to reduction of strength or reduction of fracture toughness due to the environment in the RV annulus cavity. In addition, the baseline (pre-service) inspection of the Oconee RPV included a volumetric examination of weld WR-36 using UT. Any recordable indications were required to be re-examined using RT to obtain a reportable status. Based on available information from review of the three Oconee RV quality assurance data packages, there were no reportable defects in weld WR-36 for any of the Oconee units after completion of the baseline examination.

Request 2:

The baseline condition of the transition forging to RV support skirt weld (WR-36) relative to the ability to maintain intended function has been established through programmatic activities as described below:

a. The baseline condition of the RV support assembly, which includes weld WR-36, has been established in part through the periodic examination of accessible portions of the IWF-2500 examination boundary for Oconee Units 1, 2, and 3. As described in ANP-3898NP, Revision 0, Section 9.4.1, visual examinations of the Units 1, 2, and 3 RV support assembly were performed in 2012, 2013, and 2014, respectively. These IWF examinations are performed once per each ISI inspection interval. For each unit, a VT-3 visual examination was performed in accordance with ASME Section XI, Subsection IWF, under Examination Category F-A, Item F1.40, on accessible surfaces of the RV support assembly with a calculated coverage of approximately 66.5% of the surface areas within the examination boundary. Since the examination did not meet the 100% coverage requirement in IWF-2500, Duke Energy submitted Relief Request RR 15-ON-004, which was reviewed and approved by the NRC staff. RR 15-ON-004 described that a small portion of the accessible examination surface, which contains weld WR-36, is accessible through removable inspection panels. There were no unacceptable conditions or indications detected during these examinations.

   Based on the IWF examination results, it can be concluded that the accessible areas of the RV support assembly, including small portions of weld WR-36, are in good condition, which can be considered a measure of the baseline condition for confirming that intended function is maintained prior to entering the subsequent period of extended operation for Oconee Units 1, 2, and 3.

b. The baseline condition of weld WR-36 has been established in part through the absence of bounding degradation detected on the horizontal surface of the RV support assembly for Oconee Units 1, 2, and 3. Figure 3.5.2.2.2.6-1-1 shows that the IWF-2500 examination boundary for the RV support assembly consists of vertically-oriented and horizontally-oriented regions. Weld WR-36 is located in the upper part of the assembly vertical region. The assembly horizontal region consists of the support flange, the support skirt to support flange weld, and the anchor bolts. RR 15-ON-004 described that while a limited portion of the accessible examination surface is comprised of surface A-D, which includes weld WR-36, most of the accessible examination surface is comprised of surfaces D-E and G-C, which are situated at a lower elevation than weld WR-36, and include the horizontally-oriented skirt flange, support skirt to skirt flange weld, and the anchor bolts. It is reasonable to expect that the horizontally-oriented regions of the RV support flange would develop degradation (i.e., loss of material) more readily than the vertically-oriented regions of the support skirt, since the horizontal surfaces are more likely to be potentially exposed to standing water or accumulate boron deposits. Any degradation detected on the horizontal regions of the RV support assembly, where examination accessibility is greater, would precede, or bound, degradation that may develop on vertical regions of the support skirt including weld WR-36. In addition, ANP-3898NP Section 9.4.5 reports that since weld WR-36 is not impacted by irradiation embrittlement over an 80-year operating life, reduction in fracture toughness due to irradiation embrittlement is not a concern.

   It can be concluded that the absence of unacceptable conditions or indications detected during the most recent IWF examination of the accessible horizontal surfaces of the RV support assembly provides a measure of the baseline condition for confirming that intended function is maintained prior to entering the subsequent period of extended operation for Oconee Units 1, 2, and 3.

c. There is no known operating experience to indicate that degradation of pedestal-type RV supports has resulted in a failure to maintain intended function. Operating experience will continue to be reviewed and will inform and enhance the programs that manage the relevant aging effects on the RV support assembly. If operating experience is found to have a potential impact on intended function, the issue will be entered into the Corrective Action Program to drive resolution and maintain program effectiveness so that intended function is maintained prior to entering the subsequent period of extended operation for Oconee Units 1, 2, and 3.

Request 3:

Loss of Material due to the Air - Indoor Uncontrolled (External) environment will continue to be managed using the *ASME Section XI, Subsection IWF* AMP.

Loss of Material due to the Air with Borated Water Leakage (External) environment will continue to be managed using the *Boric Acid Corrosion* AMP.

Cumulative Fatigue Damage will continue to be managed as a TLAA using the *Fatigue Monitoring* AMP.

The applicable AMR line items for the support skirt can be found in SLRA Table 3.1.2-1, as revised in the response to RAI 3.5.2.2.2.6-2, Request 1 (Attachment 26).

Request 4:

The response to RAI 3.5.2.2.2.6-2, Request 1, revises the AMP from “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)” to “ASME Section XI, Subsection IWF (B2.1.30)” for the Aging Effect of Loss of Material for the Environment of Air - Indoor Uncontrolled (External). See the response to RAI 3.5.2.2.2.6-2, Request 1, for additional details for this change. No other updates to the SLRA are required as a resulting of the responses to Requests 1, 2, and 3, above.

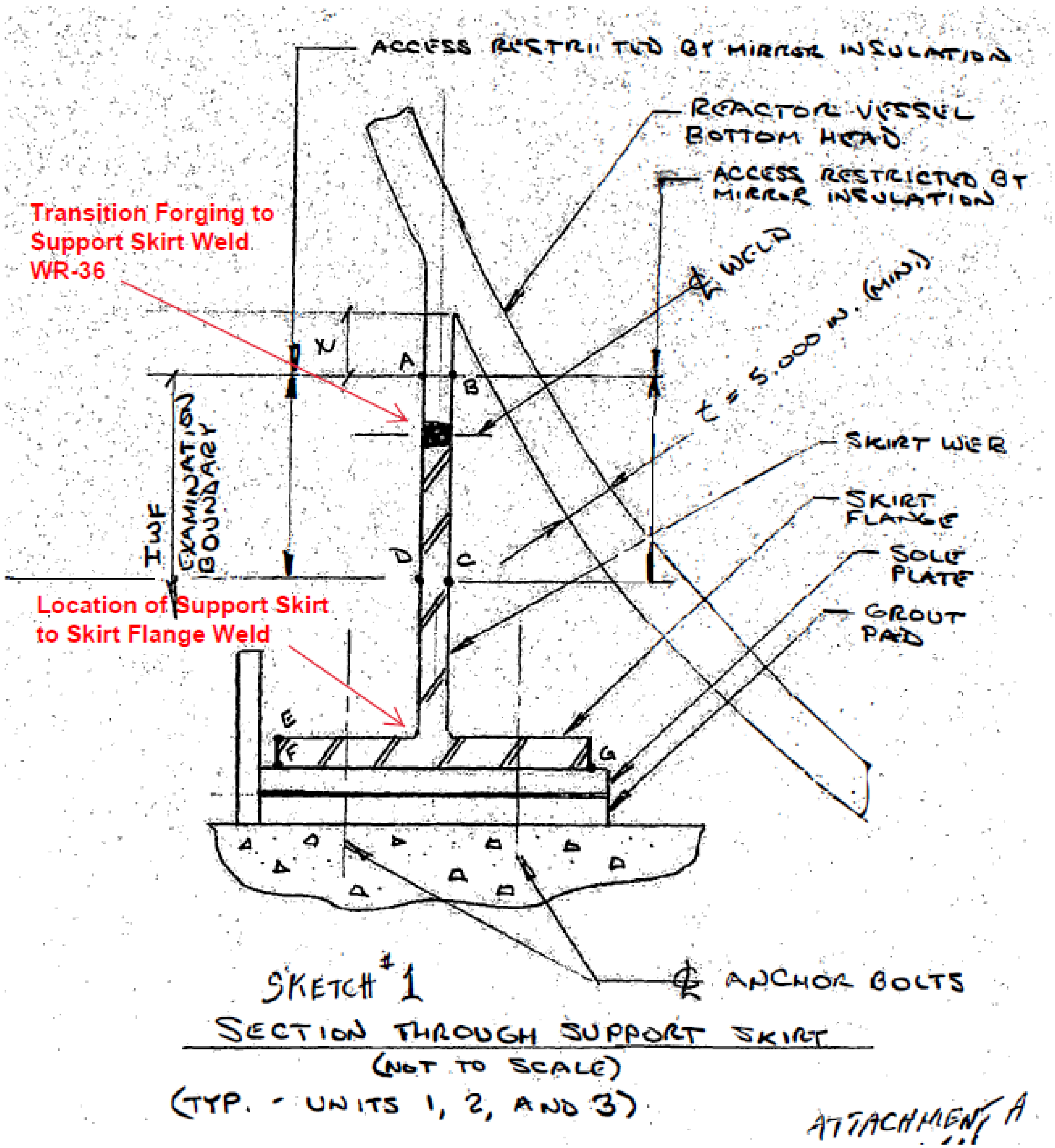

Figure 3.5.2.2.2.6-1-1

RV support assembly IWF-2500 examination boundary A-D-E-F-G-C-B. (Reference: Relief Request RR 15-ON-004)

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 26
RAI 3.5.2.2.2.6-2

Enclosure 1, Attachment 26

RAI 3.5.2.2.2.6-2:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
SLRA Section 3.5.2.2.2.6 defines the RV support assembly to be the RV steel "support skirt and the reactor vessel support flange, which were attached to the reactor vessel during fabrication of the reactor vessel." SLRA Table 3.1.2-1, "Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel - Aging Management Evaluation," includes an AMR line item for loss of material aging effect associated with the RV steel support skirt. The AMR cites SLRA AMP B2.1.1 "ASME Code Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD," with generic note J that states: "[n]either the component nor the material and environment combination is evaluated in NUREG-2191."

The SLRA Section B2.1.1 defers to "ASME [Code Section] XI, Subsection IWF" (SLRA AMP B2.1.30) to which the applicant claims consistency with enhancement and no exceptions to GALL-SLR XI.S3 for management of aging effects for loss of material, cracking, loss of preload, and loss of mechanical function for ASME Class 1, 2, and 3 and MC component supports. The RV skirt is classified as an ASME Class 1 support.

Issue:
ASME Code Section XI, Subsection IWB, Table IWB-"2500-1 (B-K) Examination Category B-K, Welded Attachments for Vessels, Piping, Pumps, and Valves," references Figure IWB-2500-14 which outlines the extent of the IWB boundary. From information provided in ONS relief request submittal (ADAMS Accession No. ML15201A573) it is apparent that the skirt to dutchman weldment are outside the ASME Code Section XI IWB inspection boundaries but included in ASME Code Section XI IWF inspections. It is not clear how ONS plans to use the SLRA AMP B2.1.1 to manage the effects of aging for loss of material through the SLRA Table 3.1.2-1 AMR line item when the jurisdiction of ASME Section IWB is well above the RV steel support skirt. It is also not clear how generic note J is justified for the ASME Class 1 RV steel support skirt component that is required to be managed for aging consistent with the guidance provided in AMP XI.S3 of NUREG-2191.

Request:

1. Justify the conclusion made in SLRA Table 3.1.2-1 AMR line item, that for the ASME Class 1 RV steel support skirt "[n]either the component nor the material and environment combination is evaluated in NUREG-2191."
2. Discuss how all applicable aging effects for the ASME Class 1 RV steel support skirt will be managed during the subsequent period of extended operation. If SLRA AMP B2.1.30 will be used, state so and provide the relevant SLRA Table 3.5.2 AMR line items.

Response to RAI 3.5.2.2.2.6-2:

Request 1:

The conclusion in SLRA Table 3.1.2-1 requires revision. The ONS SLRA Table 3.1.2-1, Reactor Vessel, Reactor Internals, and Reactor Coolant System – Reactor Vessel – Aging Management Evaluation, currently aligns to the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program for loss of material with an Industry Standard Note J.

In accordance with IWB-2500, the only examination area relevant to the RPV support skirt is described under Examination Category B-K, "Welded Attachments for Vessels, Piping, Pumps, and Valves." However, the weld between the transition forging and the support skirt is exempt from Table IWB-2500-1, Category B-K, Item B10.10 because the weld is outside of the IWB boundary shown on Figure IWB-2500-14 in Section XI of the ASME Boiler and Pressure Vessel Code. Per Figure IWB-2500-14, this weld is outside of the "t" (vessel thickness) dimension. Therefore, this weld is exempt from this examination.

The RPV support skirt is examined under Category F-A, Item F1.40. This is part of the ASME Section XI, Subsection IWF AMP. Therefore, the AMR line item for loss of material is revised to line item III.B1.1.T-24 (SRP Item 3.5.1-091) with an Industry Standard Note A in SLRA Table 3.1.2-1. A markup of the change is provided below.

Request 2:

The two applicable aging effects for the ONS 1, 2, and 3 RPV support skirts are cumulative fatigue damage and loss of material. Cumulative fatigue damage will be managed by the Fatigue Monitoring AMP and is evaluated in Section 4.3 of the ONS SLRA. The aging effect of loss of material will be managed by the Boric Acid Corrosion and ASME Section XI, Subsection IWF AMPs. The RPV support skirt was evaluated and is not susceptible to irradiation embrittlement as described in ANP-3898P/NP, Section 9.0. The applicable AMR line items for the support skirt can be found in SLRA Table 3.1.2-1, as revised in the response to request 1 above.

SLRA Revisions:

SLRA Table 3.1.2-1 (page 3-103) is revised as follows:

Table 3.1.2-1 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel - Aging Management Evaluation

Component Type	Intended	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Support Skirt	Structural Support	Steel	Air – Indoor Uncontrolled (External)	Cumulative Fatigue Damage	TLAA	IV.A2.R-70	3.1.1- 004	A
				Loss of Material	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~ **ASME Section XI, Subsection IWF (B2.1.30)**	~~None~~ **III.B1.1.T-24**	~~None~~ **3.5.1-091**	~~J~~ **A**
				None	None	None	None	J, 1
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B2.1.4)	IV.A2.R-17	3.1.1- 049	A

ENCLOSURE 1

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ATTACHMENT 27
RAI 3.5.2.2.2.2-3
[NON-PROPRIETARY]

Enclosure 1, Attachment 27
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 3.5.2.2.2.2-3:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
SLRA Section 3.5.2.2.2.2 states that the operating experience (OE) "identified no issues related to elevated temperatures affecting concrete structures. Additionally, analysis performed to determine the maximum concrete temperature of the primary shield wall [PSW] illustrates that the concrete will not exceed 200 °F for local loads." The SLRA concludes that a "plant-specific aging management program to manage the effects of reduction of strength and modulus due to elevated temperature is not required."

Section 3.8.3.5 of the UFSAR states that the "maximum allowable concrete temperature at penetrations in the Primary Shield Wall [(PSW)] shall not exceed 400 °F."

Section 3.5.2.2.2.2 of the SRP-SLR states concrete temperatures under normal operation or any other long-term period to 66 °C (150 °F) except for local areas, which are allowed to have increased temperatures not to exceed 93 °C (200°F) and recommends a further evaluation and a plant-specific program if any portion of the safety-related and other concrete structures exceeds the specified temperature limits. It concludes that higher temperatures may be allowed if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.

Issue:
SLRA Section 3.5.2.2.2.2 states that based on analysis performed the maximum concrete temperature of the PSW concrete will not exceed 200 °F for local loads. UFSAR Section 3.8.3.5 states that maximum allowable concrete temperature at penetrations in PSW shall not exceed 400 °F. It is not clear what analysis was performed that the SLRA references that led to the conclusion that the maximum PSW concrete temperature is 200 °F instead of the 400 °F.

Request:

1. Describe the SLRA Section 3.5.2.2.2.2 referenced analysis for the PSW concrete temperature.

2. Summarize the analysis results that justify the conclusion that the maximum PSW concrete temperature will not exceed 200 °F for local loads.

Response to RAI 3.5.2.2.2.2-3:

Two analyses were performed to calculate primary shield wall concrete temperatures to support the following statement in Section 3.5.2.2.2.2 of the ONS SLRA (Reference 5-1): "Additionally, analysis performed to determine the maximum concrete temperature of the primary shield wall illustrates that the concrete will not exceed 200 °F for local loads."

1. Oconee Reactor Vessel Cavity Concrete Temperature

2. Concrete Temperature Near the Hot Leg Nozzle

As reported in Section 3.5.2.2.2.2 of the ONS SLRA, maximum concrete temperatures are less than 200°F; these analyses are summarized below.

1. Analytical Methodology for the Oconee Reactor Cavity Concrete Temperature

A 2-dimensional model of the lower RV, RV skirt, and anchoring and supporting structure was prepared. **[**

]b,c,d,e

Summary and Conclusions

The temperature contours for the anchor bolts location are shown in Figure 3.5.2.2.2.2-3-1. Contours for the shear pin location are show in in Figure 3.5.2.2.2.2-3-2. Areas outside the range are colored in gray. The temperature units are Fahrenheit (°F). All primary shield wall concrete temperatures are less than 200° F; the highest concrete temperature in the primary shield wall adjacent to the core midplane is **[ ]b,d,e** (Figures 3.5.2.2.2.2-3-1 and 3.5.2.2.2.2-3-2). The concrete below the sole plate is predicted to reach a peak temperature of **[ ]**b,d,e for the shear pin model. The peak concrete temperature for the anchor bolt model equals **[ ]** b,d,e , which is lower than the shear pin model owing to conduction through the anchor bolts to the basemat. **[ ] b,c,d,e**, which is below 200°F.

[ ]b,d,e

Figure 3.5.2.2.2.2-3-1
Anchor Bolts Location Model, Reduced Temperature Scale

b,d,e

Figure 3.5.2.2.2.2-3-2
Shear Pin Location Model, Reduced Temperature Scale

2. Analytical Methodology for the Concrete Temperature Near the Hot Leg Nozzle

A manual heat transfer calculation was performed to estimate the concrete surface temperature at the location where the reactor vessel (RV) outlet nozzle/hot leg piping penetrates the Primary Shield Wall (PSW) for the Oconee (ONS) units.

The heat flow through the hot leg wall and insulation is primarily in the radial direction toward the concrete, so a one-dimensional (1-D) cylindrical heat transfer model may be employed.

Summary and Conclusions

From a one-dimensional steady-state approach, considering a combination of free convection and radiation around the PSW hot leg penetrations, the heat transfer coefficients and thermal resistances through the different material layers were calculated to estimate the local concrete temperature at the PSW penetrations near the hot leg regions. Air within the reactor cavity was set to ambient temperature, and because not fully known, the air flow within the penetrations was considered nearly stagnant.

With these conservative considerations, the results demonstrate that the PSW hot leg penetrations and local concrete temperatures (e.g., around sleeve for example) remain below 200 °F without exceeding it. Temperature results are provided in Figure 3.5.2.2.2.2-3-3.

Figure 3.5.2.2.2.2-3-3
PSW Temperatures Near Hot Leg Regions (Not to Scale)

[ ] b,d,e

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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SET 2

ATTACHMENT 28
RAI 3.5.2.2.2.6-4
[**NON-PROPRIETARY]**

Enclosure 1, Attachment 28
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Duke Energy and Framatome, Inc and will be denoted by 'D' or 'F', respectively.

RAI 3.5.2.2.2.6-4:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
Section 9.4.4.1 of ANP-3898NP/P Revision 0, "Framatome Reactor Vessel and RCP TLAA and Aging Management Review Input to the ONS SLRA," (Enclosure 4, Attachment 1 to the SLRA, ADAMS Accession No. ML21158A200 for the non-proprietary version and Enclosure 5 to the SLRA for the proprietary version) states that:

> "Neutron fluence and gamma dose at 80-years (72 EFPY) are calculated using source terms that bound all Oconee units to provide bounding estimates for RPV [reactor pressure vessel] support and the biological shield wall."

Issue:
Although the description provided in the ANP-3898 would lead to conservative neutron fluence and gamma dose estimates at 80-years (72 EFPY), the degree of conservatism in the calculations that results from employing the bounding source is not clear. In order for the NRC staff to evaluate the degree of conservatism in the source term and determine whether those source terms provide adequately bounding fluence and gamma dose estimates, comparative information relative to the actual plant operating condition is necessary. Such a determination will enable the staff to determine whether the downstream aging effects are based on valid estimates of radiation exposure.

Request:
Provide the radial Relative Power Distribution (RPD) of a recent cycle that is representative of typical operating conditions at ONS Units 1, 2, and 3.

Response to RAI 3.5.2.2.2.6-4:

In order to bound all potential future fuel cycle designs, a bounding source term for Oconee was used to determine the maximum E > 1.0 MeV and E > 0.1 MeV neutron flux and fluence, as well as the gamma dose at the inside surface of the biological shield wall. **[**

]Da,b,c; Fb,c,e Figure 3.5.2.2.2.6-4-1 shows the power distribution used in the ANP-3898P, Revision 0 fluence analysis.

Oconee has used low-leakage core loading patterns since the 1980s. In low-leakage cores, fresh fuel is loaded in the core interior and burned fuel is loaded on the core periphery. Figures 3.5.2.2.2.6-4-2, 3.5.2.2.2.6-4-3 and 3.5.2.2.2.6-4-4 show RPD maps for Oconee Unit 1 Cycle 29, Unit 2 Cycle 28, and Unit 3 Cycle 28. These core designs are typical of the low leakage cores that Oconee has utilized over the past 30 years.

In order to precisely quantify and evaluate the degree of conservatism in the source term, an adjoint calculation is required to determine the fraction of neutrons originating from each assembly. Since an adjoint calculation is not available, the degree of conservatism can be estimated using engineering judgement. It is well established that the peripheral assemblies are the most important with regard to RV fluence and leakage of neutrons outside the reactor vessel. In particular, assemblies N14, O13, and P12 contribute the most to the peripheral flux, which is used to calculate 72 EFPY fluence at the RV and the RV transition forging to RV skirt weld WR-36. The other assemblies that will also be important include L15 and R10. As shown in Figures 3.5.2.2.2.6-4-1, 3.5.2.2.2.6-4-2, 3.5.2.2.2.6-4-3 and 3.5.2.2.2.6-4-4, the bounding source term used in the ANP-3898P, Revision 0 fluence analysis will produce fluence results that are approximately 2-3 times higher than the actual core power distributions.

[ ] Da,b,c; Fb,d,e

Figure 3.5.2.2.2.6-4-1

Bounding source used to calculate conservative neutron fluence and gamma dose estimates at 80 years (72 EFPY)

[ ]Da,b,c; Fb,c,e

Figure 3.5.2.2.2.6-4-2

Oconee Unit 1 Cycle 29 relative power distribution

[ ] Da,b,c; Fb,c,e

Figure 3.5.2.2.2.6-4-3

Oconee Unit 2 Cycle 28 relative power distribution

[ ] Da,b,c; Fb,c,e

Figure 3.5.2.2.2.6-4-1

Oconee Unit 3 Cycle 28 relative power distribution

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit and Enclosure 2, Attachment 2 for the Duke Energy Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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SET 2

ATTACHMENT 29
RAI 3.5.2.2.2.6-5
[NON-PROPRIETARY]

Enclosure 1, Attachment 29
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 3.5.2.2.2.6-5:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
In Section 9.4.4.1 of ANP-3898NP/P, Revision 0 (Enclosure 4, Attachment 1 to the SLRA, ADAMS Accession No. ML21158A200), the applicant stated that the exposure level of 5.53E-04 displacements per atom (dpa) is conservatively assumed to be applicable to all the components of the RPV steel support assembly of each ONS unit.

Issue:
Even though the applicant stated that the exposure level of 5.53E-04 dpa is conservatively assumed to be applicable to all the components of the RPV steel support assembly, the staff is not clear how conservative the value of 5.53E-04 dpa is compared to the actual exposure level at the location where the RPV steel support assembly is anchored to the concrete pedestal embedment, i.e., at the components of the RPV steel support assembly at the RPV steel support flange elevation and below. The staff needs this clarification in order to evaluate the margin in exposure level at the RPV steel support flange elevation and below since the applicant determined that the RPV steel support flanges of the three ONS units (and the associated welds at ONS Units 1 and 2) are potentially susceptible to reduction of fracture toughness by irradiation embrittlement (Section 9.4.5 of ANP-3898NP/P, Revision 0). The staff needs this information as part of its overall evaluation of reasonable assurance that the RPV steel support assembly of each ONS unit will perform its intended function through the subsequent period of operation.

Request:
Describe the conservatism included in using the value of 5.53E-04 dpa at the RPV steel support flange elevation and below.

Response to RAI 3.5.2.2.2.6-5:

As discussed in Section 9.4.4.1 of ANP-3898P/NP, Revision 0, the DORT computer code was used to estimate the neutron fluence at the transition forging to RV skirt weld (WR-36). The location of weld WR-36 is approximately **[ ]$_{b,d,e}$** above the RV support assembly and embedment materials, as shown in Figure 3.5.2.2.2.6-5-1. Since the embedment location is significantly below the area of the vessel expected to reach an end-of-life fluence of $1x10^{17}$ n/cm^2, computer models of this region of the vessel were not prepared. Due to the complex geometry of the skirt region, high-fidelity three-dimensional (3D) transport solutions are required. Since calculational data is not available, measurement data can be used to infer the level conservatism introduced by using a location that is approximately **[** **]**b,d,e above the embedment location. In this case, chain data from Davis Besse cycle 6 will be used. Four 41-foot-long beaded stainless steel chains were placed in the cavity region quadrants at symmetric azimuthal positions. After irradiation, the chains were segmented and sent to the BWXT laboratory in Lynchburg, VA for testing. The results for the four chains were averaged together. This data was previously presented to the NRC in ANP-10348-P and is shown in Table 3.5.2.2.2.6-5-1.

[ ]**b,d,e**

Figure 3.5.2.2.2.6-5-1

Illustration of actual RPV steel support location compared to the calculated flux location

Table 3.5.2.2.2.6-5-1

Chain activity as a function of location below the vessel mating surface

[ ]b,d,e

The skirt weld and embedment are located at **[ ]b,d,e**, respectively, below the flange mating surface. To determine the expected reduction in flux, a curve fit of the measurement data was made, as shown in Figure 3.5.2.2.2.6-5-2.

Table 3.5.2.2.2.6-5-2 shows the interpolated results at the skirt weld and embedment locations. It can be inferred from the chain data that the value of 5.53E-04 dpa is conservative by a factor of 2 due to the location selected.

[ ]b,d,e

Figure 3.5.2.2.2.6-5-2

Comparison of calculated dpa location to the actual concrete pedestal embedment location

Table 3.5.2.2.2.6-5-2

Interpolated activities at the skirt weld and Concrete pedestal embedment locations

b,d,e

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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SET 2

ATTACHMENT 30
RAI 3.5.2.2.2.6-6
[**NON-PROPRIETARY]**

Enclosure 1, Attachment 30
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 3.5.2.2.2.6-6:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:

Section 9.4.1 of ANP-3898NP/P, Revision 0, discusses the applicant's assessment of the current condition of the RPV steel support skirt assemblies of ONS Units 1, 2, and 3. The staff noted the discussion of the visual examinations performed on the RPV steel support skirt assemblies during inservice inspections, but also noted that there is no information on preservice examinations.

Issue:

Since visual examinations are performed to look for evidence of gross deformation or misalignment that could result from cracks/indications rather than to detect the cracks/indications themselves, the staff needs information on the preservice examinations performed on the RPV steel support assemblies of ONS Units 1, 2, and 3 in order to clarify that any unacceptable cracks/indications in the RPV steel support assemblies would have been detected (through surface or volumetric examinations), and that, therefore, the specific component of the assemblies would have been replaced or repaired, or have not been used if preservice examinations in the material procurement specifications did not allow materials with preservice cracks/indications.

Request:
Describe either the preservice examinations (include all results) performed for the RPV steel support skirt assemblies of ONS Units 1, 2, and 3 or the material procurement specifications for the components of the RPV steel support skirt assemblies with regard to prohibiting preservice cracks/indications in the components used in the assemblies.

Response to RAI 3.5.2.2.2.6-6:

The fabrication steps and nondestructive testing (magnetic particle) for the RV skirt to support skirt flange circumferential weld (WR-37) is consistent with that described for the transition forging to RV skirt circumferential weld (WR-36), as described in the response to RAI 3.5.2.2.2.6-1 for Oconee Units 1, 2, and 3. RV skirt axial welds (WR-40) and RV support skirt flange welds (WR-50) for Oconee Units 1 and 2 were made **[**

][b,d,e] All nondestructive examinations were performed in accordance with the existing ASME Section III requirements. Welds WR-37, WR-40, and WR-50 for Oconee Units 1, 2, and 3 were fabricated in accordance with ASME Section III (1965 Edition through Summer 1967 Addenda) requirements and corresponding B&W shop specifications that complied with all ASME III fabrication requirements. Fabrication defects in these welds were removed and repaired as discussed in the response to RAI 3.5.2.2.2.6-1 for weld WR-36. Residual stresses within the weld were reduced and redistributed owing to final reactor vessel shell PWHT. There are no fabrication defects for welds WR-37, WR-40, or WR-50 that could contribute to reduction of strength or reduction of fracture toughness due to the environment in the RV annulus cavity.

The RV skirt was fabricated from SA-516 Grade 70 plate and the RV support skirt flange was fabricated from SA-515 Grade 70 plate material. These plates received volumetric inspection (UT) prior to rolling and welding. Based on review of the RV quality assurance data packages for all three Oconee Units, there are no fabrication defects for the RV skirt plate and RV support skirt flange that could contribute to reduction of strength or reduction of fracture toughness due to the environment in the RV annulus cavity.

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 31
RAI 3.5.2.2.2.6-7
[**NON-PROPRIETARY]**

Enclosure 1, Attachment 31
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 3.5.2.2.2.6-7:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
Section 9.3 of ANP-3898NP/P, Revision 0 states that the vertical bearing plate and nelson studs "do not support the RPV [steel] support assembly intended function" and that the intended function of the RPV steel support assembly is to provide structural support for the RPV.

Issue:
Based on its audit and the review of SLRA Section 3.5.2.2.2.6 and Sections 9.3 and 9.4.5 of ANP-3898NP/P, Revision 0, the staff is not clear how the vertical bearing plate and nelson studs do not have an RPV steel support assembly intended function of providing structural support for the RPV of ONS Units 1, 2, and 3.

If the vertical bearing plate and nelson studs have an intended function of providing structural support for the RPV, these two components need to be evaluated for loss of fracture toughness due to irradiation embrittlement consistent with the evaluations in Section 9.4 of ANP-3898NP/P, Revision 0. Also, if the vertical bearing plate and nelson studs do have the aforementioned intended function, the staff would need clarification whether the calculation of the stress intensities in Section 9.3 of ANP-3898NP/P, Revision 0 and the stresses in the evaluation of the RPV support flange and associated welds discussed in Section 9.4.5 of ANP-3898NP/P, Revision 0 included the effect of the vertical bearing plate and nelson studs.

Request:

1. Clarify whether the vertical bearing plate and nelson studs described in Section 9.3 of ANP-3898NP/P, Revision 0 have an intended function of providing structural support for the RPV of ONS Units 1, 2, and 3.

2. If the vertical bearing plate and nelson studs have an intended function of providing structural support for the RPV, evaluate the vertical bearing plate and nelson studs for loss of fracture toughness due to irradiation embrittlement consistent with the evaluations in Section 9.4 of ANP-3898NP/P, Revision 0. In this evaluation, include the following aspects of the vertical bearing plate and nelson studs: material specifications, initial NDT and associated margins, cited source documents of initial NDT and associated margins, and preservice examinations (including results) or material procurement specifications with regard to prohibiting preservice cracks/indications.

3. Clarify whether the calculation of the stresses in Section 9.4.2 of ANP-3898NP/P, Revision 0 and in the evaluation of the RPV support flange and associated welds discussed in Section 9.4.5 of ANP-3898NP/P, Revision 0, included the effect of the vertical bearing plate and nelson studs. If the effect of the vertical bearing plate and nelson studs was not included, explain how the stresses determined in Sections 9.4.2 and 9.4.5 of ANP-3898NP/P, Revision 0 are conservative with respect to stresses had the effect of the vertical bearing plate and nelson studs been included.

Response to RAI 3.5.2.2.2.6-7:

Request 1:

Based on the evaluation reported in BAW-1621, the sole plate, vertical bearing plate, and associated nelson studs do support the RV support assembly intended function and must be evaluated for susceptibility to reduction of fracture toughness by irradiation embrittlement. In accordance with BAW-1621, Section 6.2, the reactor vessel support skirt assembly and associated embedment were evaluated for structural integrity in response to NRC concerns regarding asymmetric loads following a Loss of Coolant Accident (LOCA) as required by Generic Letter 78-02, Design of the Reactor Pressure Vessel Support System. Responses to NRC RAIs regarding BAW-1621 were provided to the NRC through BAW-1621, Supplement 1. NRC acceptance of BAW- 1621 and BAW-1621, Supplement 1 for the Oconee Nuclear Station is reported in NRC safety evaluation report dated May 2, 1983 (ADAMS Accession Number ML15238A802). The BAW-1621 evaluation resulted in an assessment of the impact of asymmetric LOCA loads on the RPV supports that were not defined in the original ASME Section III RPV design report and are part of the CLB for the Oconee Units.

As described in BAW-1621, Section 6.2.1.1, finite element models were developed to determine the reactions of the embedment to LOCA loading. The sole plate and vertical bearing plate are fabricated from ASTM A36 or SAE 1030 steel; A36 and SAE 1030 are assumed to be identical relative to initial NDT. The material of construction for the nelson studs is not specified and the nelson studs are designated as H4L, which are headed concrete anchors and may be assumed to be low carbon mild steel since they are welded to the vertical bearing plate.

Request 2:

The sole plate, vertical bearing plate, and nelson studs were evaluated for susceptibility to reduction of fracture toughness consistent with the methodology reported in ANP-3898P/NP, Revision 0, Section 9.4.4. Consistent with ANP-3898P/NP, Revision 0, the 72 EFPY dpa is conservatively assumed at 5.53E-04 for all embedment items. Per the guidance in Section 4.3.4.2 of NUREG-1509, the margin term is determined using ASME Boiler and Pressure Vessel Code Section III, Appendix R, Figure R-1200-1 (2013 edition) to account for uncertainties in the NDT determination. Per Figure R-1200-1, a margin of 30°F (17°C) **[**

]$_{b,d,e}$ The results of the lowest service temperature (LST) evaluation for the embedment items (sole plate, vertical bearing plate, and nelson studs) are reported in Tables 3.5.2.2.2.6-7-1 and 3.5.2.2.2.6-7-2.

Table 3.5.2.2.2.6-7-1
Calculated ΔNDTT Values for the RPV Support Skirt Embedment Base Metal Materials

[ ]b,d,e

Table 3.5.2.2.2.6-7-2
Comparison of the ΔNDTT dpa at ONS RPV Support Skirt Embedment Base Metal LST to Projected 72 EFPY dpa

[ ]b,d,e

The only steel embedment items that may be susceptible to irradiation embrittlement are the nelson studs. Nelson studs are used to ensure the vertical bearing plate is tightly locked to the concrete behind the vertical bearing plate during construction. **[** **]b,c,e** may stress the nelson studs; however, with leak-before-break (LBB), the loads on the nelson studs for this scenario are expected to be insignificant and have no impact on the RV support intended function.

Request 3:

The discussion of RV support stresses reported in Section 9.4.2 of ANP-3898NP/P only addressed the RV support skirt steel members as identified in the Oconee ASME Section III RV design reports (i.e., anchor bolts, shear pins, and associated nuts and washers). In addition to the RV design reports, the RV support skirt assembly and associated embedment were evaluated for structural integrity in response to NRC concerns regarding asymmetric loads following a LOCA as required by Generic Letter

78-02, Design of the Reactor Pressure Vessel Support System. The BAW-1621 evaluation resulted in an assessment of the impact of asymmetric LOCA loads on the RV supports that were not defined in the original design reports and are part of the CLB for the Oconee Units.

As described in BAW-1621, Section 6.2.1.1, finite element models were developed to determine the reactions of the embedment to LOCA loading. The embedment system evaluated in BAW-1621 includes the sole plate, vertical bearing plate, and nelson studs. Therefore, based on the evaluation reported in BAW-1621, the vertical bearing plate, and associated nelson studs do support the RPV support assembly intended function and are evaluated for susceptibility to reduction of fracture toughness by irradiation embrittlement herein (see Response 2, above).

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 32
RAI 3.5.2.2.2.6-8
[NON-PROPRIETARY]

Enclosure 1, Attachment 32
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI 3.5.2.2.2.6-8:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(1) requires that a license renewal application contains an integrated plant assessment (IPA) that identifies and lists those structures and components that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires that the applicant demonstrates that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed such that their intended function(s) are maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Background:
In Section 9.4.5 of ANP-3898NP/P, Revision 0, the applicant evaluated the RPV support flanges of ONS, Units 1, 2, and 3, and the RPV support flange welds of ONS Units 1 and 2 because they were determined to be potentially susceptible to reduction of fracture toughness by irradiation embrittlement at 72 EFPY of operation. In this evaluation, the applicant stated that:

> "the RPV support flange was not explicitly addressed for faulted loads as the [anchor] bolts were determined to be the weak link and the [anchor] bolts were shown acceptable with margin."

In SLRA Section 3.5.2.2.2.6, the applicant also stated that this evaluation concluded that:

> "…the potential for irradiation embrittlement of the support flange is acceptable considering: (1) configuration of the reactor vessel support skirt welded to the flange plate and the numerous bolts connecting the flange plate to the concrete, and (2) stresses in the flange plate are bounded by the stresses in the RPV [steel] support skirt, which is considered the most vulnerable part of the fracture-critical member."

Issue:

SLRA Section 3.5.2.2.2.6 and ANP-3898NP/P, Revision 0 do not appear to provide sufficient details of the evaluation performed for reduction of fracture toughness by irradiation embrittlement for the RPV support flanges of ONS, Units 1, 2, and 3 and the RPV support flange welds of ONS Units 1 and 2. Specifically, there is insufficient detail of (a) how it was determined that for faulted loads, the weakest link were the 96 bolts that anchor the RPV steel support skirt assembly onto the concrete pedestal (48 bolts equally spaced outside of the RPV support flange and 48 bolts equally spaced inside) as depicted in Figure 9-2 of ANP-3898NP/P, Revision 0; (b) what the margins are in the anchor bolts for faulted loads; and (c) how it was determined that the RPV steel support skirt was considered the most vulnerable part of the fracture-critical member.

Request:

Provide details on:

1. The determination that for faulted loads, the weakest link were the 96 bolts that anchor the RPV steel support skirt assembly onto the concrete pedestal.

2. The margins in the anchor bolts for faulted loads.
3. The determination that the RPV steel support skirt was considered the most vulnerable part of the fracture-critical member.

Details should include methodology, major assumptions and conservatisms, and type of current licensing basis documents, such as stress reports or analyses of records from which the details are obtained.

Response to RAI 3.5.2.2.2.6-8:

The discussion of RV support stresses reported in Section 9.4.2 of ANP-3898NP/P only addressed the RV support skirt steel members as identified in the Oconee ASME Section III RV design reports. In addition to the RV design reports, the RV support skirt assembly and associated embedment were evaluated for structural integrity in response to NRC concerns regarding asymmetric loads following a Loss of Coolant Accident (LOCA) as required by Generic Letter 78-02, Design of the Reactor Pressure Vessel Support System in BAW-1621. Responses to NRC RAIs regarding BAW-1621 were provided to the NRC through BAW-1621, Supplement 1. NRC acceptance of BAW-1621 and BAW-1621, Supplement 1 for the Oconee Nuclear Station is reported in NRC safety evaluation report dated May 2, 1983 (ADAMS Accession Numbers ML19320B058, ML20009B628, and ML15238A802). The BAW-1621 evaluation resulted in an assessment of the impact of asymmetric LOCA loads on the RV supports that were not defined in the original RV design report and are part of the CLB for the Oconee Units.

The postulated LOCAs in BAW-1621 are from hot leg and cold leg guillotine breaks. The hydraulic forces were applied to the RV isolated model resulting in loads throughout the RV, RV supports, and RV internals. The RV support loads were used in the embedment structural models. **[**

]b,c,d,e

[

]b,c,d,e

BAW-1621 and BAW-1621 Supplement 1 were reviewed by the NRC. The NRC safety evaluation report, Sections 2.4.1 and 2.4.2, reported the results of the evaluation of the RV support assembly and RV embedments for the effects of asymmetric LOCA Loads. The NRC SER findings for the RV supports and embedment for Oconee are summarized below.

[

]b,c,d,e

Since the submittal and acceptance of BAW-1621, Leak-Before-Break (LBB) has successfully been applied (BAW-1847, Revision 1) which eliminates hot and cold leg piping breaks from the design basis. As a result, smaller attached line breaks are the design breaks for the RV and RV internals. These postulated breaks are breaks in the 14 inch core flood line, the 10 inch surge line and the 12 inch decay heat line. Further, the core flood nozzles have flow restrictors installed to minimize the blowdown due to a postulated core flood line break. These reduced flow areas considerably reduce the loadings in the RV support and embedment as can be ascertained from the reduced loads due to LBB noted in Section 9.4.2.1 of ANP-3898 NP/P. The evaluations reported in BAW-1621 and BAW-1621, Supplement 1, were not updated to credit the significant reduction in loads due to LBB.

In summary, the RV support and embedment critical stressed locations are: **[**

] b,d,e These locations exceeded the allowables but were shown acceptable. Loads in these components would be considerably reduced as a result of the elimination of the large bore primary piping breaks due to LBB. These critical items are addressed for reduction of fracture toughness in the response to RAI 3.5.2.2.2.6-7.

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 33
RAI B2.1.8-1

Enclosure 1, Attachment 33

RAI B2.1.8-1:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Subsequent License Renewal Application (SLRA) Supplement 1, dated October 28, 2021 (ADAMS Accession No. ML21302A208), revised SLRA Tables 3.3.2-48 and 3.3.2-49 to add Aging Management Review (AMR) item 3.3.1-126 to manage wall thinning due to erosion for the steel piping and piping components exposed internally to raw water in the condenser circulating water (CCW) system and the low pressure service (LPS) water system, respectively.

SLRA Supplement 1 also added plant specific notes to SLRA Tables 3.4.2-1, 3.4.2-3, 3.4.2-7, 3.4.2-10 that state, “Steel ‘Piping and Piping Components’ that are susceptible to wall thinning (due to erosion or flow accelerated corrosion) includes components constructed of carbon steel and gray cast iron.”

Issue:
SLRA Table 3.3.2-48 includes gray cast iron pump casings, strainer bodies, and valve bodies exposed internally to raw water. In addition, SLRA Table 3.3.2-48 includes ductile iron valve bodies exposed internally to raw water.

SLRA Table 3.3.2-49 includes gray cast iron filter bodies and pump casings and ductile iron valve bodies exposed internally to raw water. In addition, SLRA Table 3.3.2-49 includes copper alloy and copper alloy greater than 15 percent zinc components that are exposed internally to raw water.

A plant-specific note was not added to SLRA Tables 3.3.2-48 and 3.3.2-49, like the plant-specific note added in SLRA Supplement 1 stated above, to indicate whether “Steel Piping and Piping Components” includes components constructed of materials other than steel in the CCW or LPS systems.

Request:
Discuss whether “Steel Piping and Piping Components” includes the gray cast iron, ductile iron, copper alloy, and copper alloy greater than 15 percent zinc components in the CCW and LPS systems. If it does, then revise SLRA Tables 3.3.2-48 and 3.3.2-49 to include a plant specific note that clearly describes what materials are represented by “Steel Piping and Piping Components.”

Response to RAI B2.1.8-1:

LR-ISG-2012-01, “Wall Thinning Due to Erosion Mechanisms” provides the basis for the portions of GALL-SLR Report AMP XI.M17, “Flow Accelerated Corrosion” that address erosion. LR-ISG-2012-01 establishes that locations susceptible to erosion are determined based on plant-specific or industry operating experience, including extent of condition reviews performed in accordance with the station's corrective action process in response to said operating experience. Further, LR-ISG-2012-01 acknowledges that degradation due to erosion mechanisms is often due to improper operation or design and, as such, aging management of locations determined to be susceptible based on operating experience may not be required provided the source of the degradation is eliminated through design or operating parameter changes. Consistent with 10 CFR 54.21(a)(1), locations determined to be susceptible based on operating experience for which a time-based replacement strategy is implemented would not be subject to AMR. The scope of components for which the erosion-related guidance contained in GALL-SLR AMP XI.M17 applies is limited to locations identified as susceptible based on operating experience for which the station has "chosen to periodically monitor wall thickness as its basis for ensuring that the intended function(s) will be maintained in the period of extended operation."

Consistent with the expanded guidance provided in LR-ISG-2012-01, GALL-SLR Report AMP XI.M17, "Flow Accelerated Corrosion" (Element 4, Detection of Aging Effects) states: "For erosion mechanisms, the program includes the identification of susceptible locations based on the extent-of-condition reviews from corrective actions in response to plant-specific and industry OE." The addition of wall-thinning due to erosion as an aging effect in the CCW System (SLRA Table 3.3.2-48) and the LPS System (SLRA Table 3.3.2-4) is based on plant operating experience (OE). For both systems, the OE identifying erosion was limited to components constructed of carbon steel. Although these systems include components constructed of materials other than steel that are exposed to the same general service environment (i.e., raw water), degradation due to erosion is driven by local factors including flow conditions (e.g., cavitation damage downstream of a throttled valve) and geometric design (e.g., impingement damage at elbows). The Oconee OE indicating that erosion is occurring at certain locations in steel piping does not indicate that similar degradation is occurring in gray cast iron, ductile iron, or copper alloy components in other parts of the system. Therefore, wall thinning due to erosion is only an aging effect requiring management for steel piping in these systems and no revision to SLRA Tables 3.3.2-48 and 3.3.2-49 is required.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 34
RAI B2.1.8-2

Enclosure 1, Attachment 34

RAI B2.1.8-2:

<u>Regulatory Basis:</u>
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

<u>Background:</u>
The Aging Management Program XI.M17, "Flow-Accelerated Corrosion," "detection of aging effects" program element discussion in Volume 2 of NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," states that identification of erosion susceptible locations is based on corrective actions in response to plant-specific operating experience and that the associated components can be treated similar to other "susceptible-not-modeled" lines. Section 4.4 of SLR-ONS-AMPR-XI.M17, "Flow-Accelerated Corrosion AMP Evaluation Report," states, "An erosion susceptibility analysis has been performed to identify locations within FAC susceptible systems where the potential for erosion damage may exist." In addition, Section 4 of Technical Report BP-2017-0041-TR-01, "Duke Energy - Oconee Nuclear Station Unit 1 Erosion Susceptibility Analysis (ESA)," states, "The scope of the Erosion Susceptibility Analysis is all plant piping within the Unit 1 FAC [flow-accelerated corrosion] Susceptible Systems." During the audit of the FAC program, the applicant clarified that the ESAs are limited to FAC susceptible systems (only evaluates systems susceptible to FAC) and identification of erosion in non-FAC susceptible systems is only based on operating experience.

The NRC staff notes that, based on additional operating experience reviews performed during the FAC program audit, SLRA Supplement 1, dated October 28, 2021, and SLRA Supplement 2, dated November 11, 2021 (ADAMS Accession Nos. ML21302A208 and ML21315A012, respectively), added wall thinning due to erosion as an aging effect for several non-FAC susceptible systems (e.g., CCW, LPS, and recirculating cooling water system). In addition, Section 4.7 of SLR-ONS-AMPR-XI.M17 states, "Replacement with an alternate material does not remove the component from the list of erosion-susceptible locations since a material that is completely resistant to erosion is not available." However, the NRC staff notes that neither the SLRA nor documents reviewed during the FAC program audit include discussion about adding locations to the FAC program for non-FAC susceptible systems based on operating experience.

Operating experience example No. 2 in SLRA Section B2.1.8 relates to Action Request (AR) 01822156 and AR 01823513 for cavitation damage downstream of a valve in the 3B motor driven emergency feedwater (EFW) pump test recirculation line. The NRC staff notes that the ESA may exclude the downstream portion of the EFW recirculation lines (LB-130), using the infrequent-operation exclusion criteria with a statement "Interfacing system usage represent less than 2% of plant operating time." The staff also notes that AR 01822156 includes a search of other potentially applicable operating experience that identified PIP 09-3675, relating to leaks in the turbine driven EFW recirculation lines.

The associated discussion states that the PIP added locations along the turbine driven EFW recirculation lines to the FAC program.

Issue:
Based on the above discussion, the following issues are not clear:

1. Based on the discussion in AR 1822156, additional locations for non-FAC susceptible systems have been added to the FAC program based on operating experience. However, the SLR-ONS-AMPR-XI.M17 does not provide any discussion regarding this aspect. The guidance in GALL-SLR AMP XI.M17 states that components in this category may be treated similar to other "susceptible-not-modeled" lines. Although CSD-FAC-ALL-1610.005, Revision 0, "Susceptible Non-Modeled Program," states FAC susceptible systems are susceptible to non-FAC mechanisms (i.e., cavitation, impingement, or flashing), the procedure does not address the addition of non-FAC susceptible systems to the program. The portion of the "detection of aging effects" program element in SLR-ONS-AMPR-XI.M17, pertaining to the identification of erosion locations, only addresses FAC susceptible systems. Although Revision 5 of AD-EG-ALL-1610, "Flow Accelerated Corrosion Implementation," states, "Erosion degradation mechanisms and how they are addressed are further discussed in CSD-FAC-ALL-1610.007," the cited document states that it only applies to FAC susceptible systems. In addition, the portion of the "corrective actions" program element discussion in SLR-ONS-AMPR-XI.M17 states there is a list of erosion-susceptible locations, but there is no discussion about adding non-FAC susceptible components to the list based on operating experience. It is unclear what drives the inspections that manage wall thinning due to erosion for non-FAC susceptible locations.

2. Section 4.6 of SLR-ONS-AMPR-XI.M17 states, "Additionally, a safety factor of 2 is used when calculating the next scheduled inspection for components suspected to be wearing from an erosive mechanism." Sections 5.5 and 5.8 of AD-EG-ALL-1610 were referenced, however, additional information regarding where the requirement to use a safety-factor of 2 by manually changing the FAC Manager software for erosion mechanism evaluation on non-FAC susceptible components is not specified.

3. The ESAs use an infrequent-operation exclusion criteria, based on less than 2 percent of plant operating time for the EFW recirculation system. However, operating experience for upstream components in the motor driven and turbine driven EFW recirculation lines demonstrate that lines meeting infrequent-operation exclusion criteria have experienced cavitation erosion. The NRC staff notes that the comparable infrequent-operation exclusion criteria is also used in the FAC system susceptibility analysis, but the SLRA includes an enhancement to reassess this FAC exclusion to ensure adequate bases exist to justify this exclusion. The staff also notes that alternate industry guidance related to cavitation erosion (*see* Section 4.6 of EPRI 112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure") uses a value significantly less than 2 percent of operating time for susceptibility to this aging mechanism. Therefore, the validity of the infrequent-operation exclusion criteria used in the ESA for erosion mechanisms is unclear.

4. CSD-FAC-ALL-1610.001, "Outage Inspection Planning," notes that if significant wear is not found on components with trace chromium (measured greater than 0.10 percent), then no re-inspection is needed, unless the components are subject to erosion mechanisms. SLR-ONS-AMPR-XI.M17 does not discuss how the results of the ESA for FAC susceptible components

will be integrated into the existing Unit Specific Databases to ensure that components subject to erosion mechanisms will continue to be re-inspected if the measured chromium content is greater than 0.10 percent and significant wear has not been found.

Request:

1. Discuss how and where erosion susceptible locations for non-FAC susceptible systems are documented, and what drives the inspections that manage wall thinning due to erosion.

2. Show where the safety-factor of 2 (which requires manually changing the FAC Manager software) is specified for erosion mechanism evaluations on non-FAC susceptible components.

3. Discuss the basis for using an infrequent-operation exclusion criteria based on less than 2 percent of plant operating time given the operating experience discussed above. Include a discussion about the current enhancement to verify this exclusion criteria for FAC, but not needing to verify this exclusion criteria for erosion mechanisms.

4. Discuss how results from the ESA for FAC susceptible systems are accounted for in components meeting the trace chromium exemption within the program.

Response to RAI B2.1.8-2:

Request 1:

LR-ISG-2012-01, "Wall Thinning Due to Erosion Mechanisms" provides the basis for the portions of GALL-SLR Report AMP XI.M17, "Flow Accelerated Corrosion" that address erosion. LR-ISG-2012-01 establishes that locations susceptible to erosion are determined based on plant-specific or industry operating experience, including extent of condition reviews performed in accordance with the station's corrective action process in response to said operating experience. Further, LR-ISG-2012-01 acknowledges that degradation due to erosion mechanisms is often due to improper operation or design and, as such, aging management of locations determined to be susceptible based on operating experience may not be required provided the source of the degradation is eliminated through design or operating parameter changes. Consistent with 10 CFR 54.21(a)(1), locations determined to be susceptible based on operating experience for which a time-based replacement strategy is implemented would not be subject to aging management review. The scope of components for which the erosion-related guidance contained in GALL-SLR AMP XI.M17 applies is limited to locations identified as susceptible based on operating experience for which the station has "chosen to periodically monitor wall thickness as its basis for ensuring that the intended function(s) will be maintained in the period of extended operation." NRC recommendations provided in GALL-SLR AMP XI.M17 are consistent with the expanded guidance provided in LR-ISG-2012-01.

Consistent with the NRC recommendations provided in GALL-SLR and LR-ISG-2012-01, ONS relies on the corrective action program (CAP) to identify locations where erosion is occurring and ensure appropriate corrective actions are taken. When component degradation (such as erosion in non-FAC susceptible systems) is discovered in a plant system at ONS, the condition is documented in the CAP database. If identified, erosion conditions are evaluated in accordance with the CAP process to determine appropriate corrective actions. Corrective actions may include permanent repair to eliminate the cause of erosion, a time-based replacement strategy, or recurring inspections to monitor the condition of the component and drive a condition-based replacement strategy. Where ONS elects to rely on ongoing monitoring to manage wall thinning due to erosion, the recurring inspections are documented in preventive maintenance work orders. Inspections are implemented in accordance with ONS work management processes.

Based on a review of operating experience, the following non-FAC susceptible systems contain locations susceptible to wall thinning due to erosion: Condenser Circulating Water System, Low Pressure Service Water System, Recirculating Cooling Water System, and High Pressure Injection System. GALL-SLR AMP XI.M17 provides NRC recommendations for management of wall thinning due to erosion mechanisms “that are not being managed by another program.” Evaluation of the locations identified as susceptible to wall thinning due to erosion in non-FAC susceptible systems has determined that the applicable erosion mechanisms are appropriately managed by other ONS AMPs such that there is reasonable assurance that intended functions will be maintained consistent with the current licensing basis during the subsequent period of extended operation. The basis for this conclusion and identification of the AMP that will manage the applicable erosion mechanisms for each system is provided below. Wall thinning due to erosion in non-FAC susceptible systems will not be managed by the Flow-Accelerated Corrosion AMP at ONS. Revisions to the SLRA are provided below. The CAP process, ongoing review of operating experience, and periodic AMP effectiveness reviews ensure that if new locations susceptible to erosion are identified, then these locations will be added as inspection points to the appropriate existing AMP.

Condenser Circulating Water System/Low Pressure Service Water System:

The results of inspections performed on piping in these systems indicate that wall thinning due to erosion may be occurring in certain locations. The erosion mechanism of concern in these systems is solid particle impingement. Flashing and liquid droplet impingement erosion mechanisms are not possible in these systems. Cavitation is known to occur under certain circumstances in raw water systems if a large enough flow restriction is present such that local pressure passes below the vapor pressure at the liquid temperature. System design and operating conditions make flow induced cavitation unlikely and a review of operating experience has not identified any locations in which cavitation damage to system components has occurred. Cavitation can also occur in raw water system pumps if inadequate net positive suction head is available. However, cavitation damage in raw water system pumps is exclusively caused by either improper design, improper operation, or flow blockage upstream of the pump and would not be managed through long-term ongoing monitoring of the pump for wall thinning.

Wall thinning due to solid particle impingement of components in these systems is managed by the Open-Cycle Cooling System Water AMP. As noted in LR-ISG-2012-01, erosion in raw water systems is addressed through Open-Cycle Cooling Water System AMP inspections provided that erosion mechanisms other than solid particle impingement are not occurring. The ONS Open-Cycle Cooling Water System AMP provides for monitoring and trending of the material equipment in these systems through periodic inspections and testing such that there is reasonable assurance that intended functions will be maintained consistent with the current licensing basis in the subsequent period of extended operation. Inspection results are evaluated by engineering and trended in a database to determine the appropriate reinspection interval for each inspection point and ensure repair or replacement of affected components is performed prior to reaching minimum acceptable wall thickness.

Recirculating Cooling Water System:

The results of eddy current testing of the Recirculating Cooling Water System heat exchangers tubes identified indications of outer diameter pitting type indications. Subsequent analysis of tubes removed from the heat exchangers concluded that the outer diameter degradation consisted of scoring or gouging caused by solid particle impingement erosion. No other components in the system have been identified as susceptible to solid particle impingement and no other erosion mechanisms have been detected in the system. The ONS Flow-Accelerated Corrosion AMP is not designed to manage erosion of heat exchanger tubes. Eddy current testing of heat exchangers has been shown to be effective in monitoring the condition of heat exchanger tubes and identifying both outer diameter and inner diameter degradation such that appropriate corrective actions are taken. Existing periodic eddy current testing of the Recirculating Cooling Water System heat exchangers performed in accordance with the ONS Open-Cycle Cooling Water System AMP provide for effective management of wall thinning due to solid particle impingement of the heat exchanger tubes such that there is reasonable assurance that intended functions are maintained consistent with the current licensing basis in the subsequent period of extended operation. Eddy current testing of the Recirculating Cooling Water System heat exchanger tubes is performed on a four-year frequency. Eddy current testing results are evaluated by engineering and used to ensure appropriate corrective actions are taken. Corrective actions may include retubing of the heat exchanger, tube plugging, or changes to the frequency of testing.

High Pressure Injection System:

During analyses used to support the root cause evaluation following a High Pressure Injection System safe end-to-pipe leak in the fall of 2013, the potential for flow cavitation was identified. The flow conditions for cavitation to occur are present due to high flow rates through high pressure injection line restricting orifices during full flow testing performed at the end of each refueling outage. Since the conditions exist for cavitation to occur in these lines during full flow testing, periodic ultrasonic testing of elbows located downstream of the injection line restricting orifices have been established to determine if cavitation damage is occurring and ensure elbows experiencing cavitation are replaced prior to reaching minimum acceptable wall thickness. These periodic tests are implemented as augmented inservice inspection examinations through the ONS ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP. The ONS ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP provides for appropriate inspection techniques, acceptance criteria, evaluation standards, and repair/replacement strategies such that there is reasonable assurance that intended functions are maintained consistent with the current licensing basis in the subsequent period of extended operation. Ultrasonic testing of the High Pressure Injection System elbows is performed on a four-year frequency. Ultrasonic testing results are evaluated by engineering and any results indicating wall thickness less than mill tolerance (i.e., 87.5% of nominal wall thickness) are entered in CAP.

Request 2:

As discussed in the response to Request 1, wall thinning due to erosion in non-FAC susceptible systems will not be managed by the ONS Flow-Accelerated Corrosion AMP. NRC guidance in GALL-SLR AMP XI.M17 states that “the minimum safety factor for acceptable wall thickness and remaining service life should not be less than 1.1.” The ONS Flow-Accelerated Corrosion AMP relies on a more conservative safety factor of 2 for wall thinning due to erosion in FAC-susceptible systems due to the non-linear nature of degradation due to erosion mechanisms other than solid particle impingement. Erosion in non-FAC susceptible systems is not evaluated using the FAC software.

The erosion mechanism of concern for components managed by ONS Open-Cycle Cooling Water System AMP is solid particle impingement. As discussed in EPRI 3002005530, Recommendations for an Effective Program Against Erosive Attack, solid particle impingement exhibits a linear wear rate as opposed to other erosion mechanisms which do not exhibit linear wear rates. As such, the conservative safety factor of 2 used for evaluations of erosion in FAC-susceptible systems does not directly apply. Regardless, the ONS Open-Cycle Cooling Water AMP provides for appropriate conservatism to ensure that corrective actions are taken prior to loss of intended function. In accordance with existing program requirements, evaluations of susceptible locations in the Condenser Circulating Water System and Low Pressure Service Water System rely on a safety factor of 2 if the corrosion rate is determined based on the results of a single inspection. If multiple inspections have been performed on the component, then a safety-factor of 1.5 may be used. A pre-established conservative tube plugging criteria is used for the Recirculating Cooling Water System heat exchanger tubes to determine when corrective actions (i.e., plugging of affected tubes) is required. This approach is consistent with industry best practice as described in EPRI 1022980, Guidance for an Effective Heat Exchanger Program and has been shown to be effective in preventing through wall leakage of heat exchanger tubes regardless of the specific degradation mechanism.

Similarly, the ONS ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP ensures appropriate conservatism is used in evaluations of erosion in the High Pressure Injection System elbows located downstream of the injection line restricting orifices. Augmented inservice inspection examinations of these elbows are performed to determine if cavitation damage is occurring due to flow conditions during full flow testing and, if so, establish a wear rate to ensure repair/ replacement is performed prior to reaching minimum required wall thickness. These inspections are performed on a four-year frequency and rely on a conservative acceptance criteria of 87.5% of nominal wall thickness which ensures that degraded conditions are appropriately identified and evaluated prior to approaching minimum required wall thickness. If unacceptable conditions are identified during inspections, then evaluations are performed in accordance with the ONS CAP process and ASME Code requirements.

Request 3:

As discussed in the response to Request 1, NRC recommendations provided in GALL-SLR AMP XI.M17 state that operating experience should be used to determine locations susceptible to wall thinning due to erosion. In addition to identifying susceptible locations through operating experience as recommended in GALL-SLR, ONS has chosen to conservatively include locations determined to be potentially susceptible to erosion based on evaluation of flow conditions in individual piping lines of FAC-susceptible systems. As such, susceptible locations in the motor driven and turbine driven emergency feedwater recirculation lines are included in the Flow-Accelerated Corrosion AMP. Further, flow induced cavitation damage is a localized phenomenon that occurs directly downstream of inline flow restrictions (e.g., valves) such that cavitation damage of the downstream portion of the emergency feedwater recirculation lines would not be expected. Regardless, Enhancement 1 to the Flow-Accelerated Corrosion AMP is revised as shown below to require reassessment of the infrequent-operation exclusion criterion used in analyses performed to determine susceptibility to both FAC and erosion mechanisms.

Request 4:

ONS has chosen to conservatively identify locations determined to be potentially susceptible to erosion based on evaluation of FAC-susceptible system flow conditions in the SLRA. The locations identified by evaluation as potentially susceptible are in addition to locations determined to be susceptible based on operating experience as recommended by the NRC in GALL-SLR AMP XI.M17. These evaluations have no exclusion criteria based on material type (including chromium content). The intent of the evaluations used to identify piping lines potentially susceptible to wall thinning due to erosion is to provide a risk-ranking of piping lines in FAC-susceptible systems. The risk ranking is based on erosion susceptibility and consequence of failure and used to aid engineering in the selection of new inspection locations where operating experience has not already determined that erosion damage is occurring. Evaluations used to identify potentially susceptible piping in FAC-susceptible systems assessed the system design and operating conditions to exclude piping lines where erosive flow conditions would not be possible. The results of these evaluations do not indicate that erosive flow conditions are actually occurring during system operation. Further, even if erosion causing phenomena are occurring, that does not mean that wall thinning due to erosion will occur. Erosion damage is typically highly localized and will only occur if both erosive flow conditions exist and the local component geometric design is such that the forces applied to the component surface by the local flow regime are large enough to result in damage.

As noted in the RAI, existing program procedures allow components with measured chromium greater than 0.10% to be excluded from further inspections if inspection results do not identify significant wear. However, existing procedures specify that this exclusion does not apply to components that are subject to damage due to erosion mechanisms. This approach is consistent with industry guidance contained in NSAC-202L, Revision 4. In accordance with the existing program procedures, components could only be excluded from further inspections using this criterion if inspection results confirm that only FAC-related wear is present and no wall thinning due to erosion has occurred. Determination of the reinspection interval for components where wall thinning due to erosion is identified does not consider chromium content. As discussed above, evaluations performed to identify piping lines that are potentially susceptible to wall thinning due to erosion are only intended to assess whether erosive conditions are possible and not to identify where erosion is actually occurring. If inspection results for a potentially susceptible location confirm that no wall thinning due to erosion is occurring, then exclusion of this location from further inspections is appropriate.

SLRA Revisions:

SLRA Section 3.2.2.1.3 (page 3-213) is revised as follows:

3.2.2.1.3 High Pressure Injection System

Aging Management Programs

The aging effects for components in the High Pressure Injection System are managed by the following AMPs:

- ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)
- Bolting Integrity (B2.1.9)
- Boric Acid Corrosion (B2.1.4)
- Closed Treated Water System (B2.1.12)
- External Surfaces Monitoring of Mechanical Components (B2.1.23)
- ~~Flow-Accelerated Corrosion (B2.1.8)~~
- One-Time Inspection (B2.1.20)
- TLAA
- Water Chemistry (B2.1.2)

SLRA Section 3.3.2.1.21 (page 3-359 as revised by Supplement 2 (ML21315A012)) is revised as follows:

3.3.2.1.21 Recirculating Cooling Water System

Aging Management Programs

The aging effects for components in the Recirculating Cooling Water System are managed by the following AMPs:

- Bolting Integrity (B2.1.9)
- Boric Acid Corrosion (B2.1.4)
- ~~Flow-Accelerated Corrosion (B2.1.8)~~
- Closed Treated Water System (B2.1.12)
- External Surfaces Monitoring of Mechanical Components (B2.1.23)
- One-Time Inspection (B2.1.20)
- Open-Cycle Cooling Water System (B2.1.11)
- Selective Leaching (B2.1.21)

SLRA Section 3.3.2.1.48 (page 3-411 as revised by SLRA Supplement 1 (ML21302A208)) is revised as follows:

3.3.2.1.48 Condenser Circulating Water System

Aging Management Programs

The aging effects for components in the Condenser Circulating Water System are managed by the following AMPs:

- Bolting Integrity (B2.1.9)
- Boric Acid Corrosion (B2.1.4)
- Buried and Underground Piping and Tanks (B2.1.26)
- External Surfaces Monitoring of Mechanical Components (B2.1.23)
- ~~Flow-Accelerated Corrosion (B2.1.8)~~
- Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.24)
- Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.27)
- One-Time Inspection (B2.1.20)
- Open-Cycle Cooling Water System (B2.1.11)
- Selective Leaching (B2.1.21)
- TLAA

SLRA Section 3.3.2.1.49 (page 3-413 as revised by SLRA Supplement 1 (ML21302A208)) is revised as follows:

3.3.2.1.49 Low Pressure Service Water System

Aging Management Programs

The aging effects for components in the Low Pressure Service Water System are managed by the following AMPs:

- Bolting Integrity (B2.1.9)
- Boric Acid Corrosion (B2.1.4)
- Compressed Air Monitoring (B2.1.14)
- External Surfaces Monitoring of Mechanical Components (B2.1.23)
- ~~Flow-Accelerated Corrosion (B2.1.8)~~
- Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.24)
- Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B2.1.27)
- One-Time Inspection (B2.1.20)
- Open-Cycle Cooling Water System (B2.1.11)
- Selective Leaching (B2.1.21)
- TLAA

SLRA Table 3.2.1 (page 3-244) is revised as follows:

Table 3.2.1 Summary of Aging Management Programs for Engineered Safety Features Evaluated in Chapter V of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.2.1-065	Metallic piping, piping components exposed to treated water, treated borated water	Wall thinning due to erosion	AMP XI.M17, "Flow-Accelerated Corrosion"	No	Consistent with NUREG-2191, **except that a different program is credited. The *ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* (B2.1.1) program is credited for managing wall thinning for piping and piping components in the High Pressure Injection System**.

SLRA Table 3.2.2-3 (page 3-290) is revised as follows:

Table 3.2.2-3 Engineered Safety Features - High Pressure Injection System - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Piping and Piping Components	Pressure Boundary	Stainless Steel	Treated Borated Water (Internal)	Wall Thinning	~~Flow-Accelerated Corrosion (B2.1.8)~~ **ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)**	V.D1.E-407	3.2.1- 065	~~A~~ **E**, 3
			Treated Borated water >60°C (>140°F) (Internal)	Cumulative Fatigue Damage	TLAA	V.D1.E-13	3.2.1- 001	A,3

SLRA Table 3.3.1 (page 3-475 as revised by SLRA Supplement 1 (ML21302A208)) is revised as follows:

Table 3.3.1 Summary of Aging Management Programs for Auxiliary Systems Evaluated in Chapter VII of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1-126	Metallic piping, piping components exposed to treated water, treated borated water, raw water	Wall thinning due to erosion	AMP XI.M17, "Flow-Accelerated Corrosion"	No	Consistent with NUREG-2191**, except that a different program is credited. The *Open-Cycle Cooling Water System* (B2.1.11) program is credited for managing wall thinning for piping and piping components in the Condenser Circulating Water and Low Pressure Service Water systems.**

SLRA Table 3.3.2-21 (page 3-688 as revised by SLRA Supplement 2 (ML21315A012)) is revised as follows:

Table 3.3.2-21 Auxiliary Systems - Recirculating Cooling Water System - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Piping and Piping Components	Pressure Boundary	Copper Alloy (>15% Zn)	Closed-Cycle Cooling Water (External)	Wall Thinning	~~Flow-Accelerated Corrosion (B2.1.8)~~ **Open-Cycle Cooling Water System (B2.1.11)**	None	None	H, 3, **4**

SLRA Table 3.3.2-21 (page 3-692 as revised by SLRA Supplement 2 (ML21315A012)) is revised as follows:

Plant Specific Notes:

1. The heat exchanger plate performs the same function as a heat exchanger tube. Therefore, the component type is equivalent.
2. Material of construction is inhibited Brass which is not susceptible to selective leaching.
3. The outer surfaces of the RCW Heat Exchanger (0RCW-HX-000A,B,C,D) tubes are susceptible to wall thinning from erosion based on site specific operating experience. GALL does not recognize erosion as an aging effect in closed-cycle cooling water environments.
4. **The Open-Cycle Cooling Water System program is credited for managing wall thinning on the closed-cycle cooling water (exterior) side of the RCW Heat Exchanger tubes by means of eddy current testing performed on the interior (raw water side) of the tubes.**

Table 3.3.2-48 (page 3-928 as revised by SLRA Supplement 1 (ML21302A208)) is revised as follows:

Table 3.3.2-48 Auxiliary Systems - Condenser Circulating Water System - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Piping and Piping Components	Pressure Boundary	Steel	Raw Water (Internal)	Wall Thinning	~~Flow-Accelerated Corrosion (B2.1.8)~~ **Open-Cycle Cooling Water System (B2.1.11)**	VII.C1.A-409	3.3.1- 126	~~A~~ **E**
	Structural Integrity	Steel	Raw Water (Internal)	Cumulative Fatigue Damage	TLAA	VII.E1.A-34	3.3.1- 002	A,3
				Wall Thinning	~~Flow-Accelerated Corrosion (B2.1.8)~~ **Open-Cycle Cooling Water System (B2.1.11)**	VII.C1.A-409	3.3.1- 126	~~A~~ **E**

SLRA Table 3.3.2-49 (page 3-949 as revised by SLRA Supplement 1 (ML21302A208)) is revised as follows:

Table 3.3.2-49 Auxiliary Systems - Low Pressure Service Water System - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Piping and Piping Components	Pressure Boundary	Stainless Steel	Raw Water (Internal)	Cumulative Fatigue Damage	TLAA	VII.E1.A-57	3.3.1- 002	A,2
		Steel	Raw Water (Internal)	Cumulative Fatigue Damage	TLAA	VII.E1.A-34	3.3.1- 002	A,2
				Wall Thinning	~~Flow-Accelerated Corrosion (B2.1.8)~~ **Open-Cycle Cooling Water System (B2.1.11)**	VII.C1.A-409	3.3.1- 126	~~A~~ **E**
	Structural Integrity	Steel	Raw Water (Internal)	Cumulative Fatigue Damage	TLAA	VII.E1.A-34	3.3.1- 002	A,2
				Wall Thinning	~~Flow-Accelerated Corrosion (B2.1.8)~~ **Open-Cycle Cooling Water System (B2.1.11)**	VII.C1.A-409	3.3.1- 126	~~A~~ **E**

SLRA Appendix A2.1 (page A-4) is revised as follows:

A2.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The *ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* AMP is an existing condition monitoring program that manages the aging effects of cracking, loss of fracture toughness, and loss of material. **Wall thinning is managed through periodic augmented inservice inspection examinations for locations where the aging effect has been identified as a concern.** The program consists of periodic volumetric, surface, and/or visual examinations and leakage tests of ASME Class 1, 2, and 3 pressure retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure retaining bolting. The program includes assessment, identification of signs of degradation and establishment of corrective actions.

SLRA Appendix A2.8 (page A-10) is revised as follows:

A2.8 Flow-Accelerated Corrosion

Enhancement

The *Flow-Accelerated Corrosion* AMP will be enhanced to:

1. Reassess infrequently used piping systems excluded from the scope of the program **with respect to evaluations performed to determine susceptibility to both wall thinning due to flow accelerated corrosion and wall thinning due to erosion mechanisms** to ensure adequate bases exist to justify this exclusion for the SPEO.

SLRA Table A6.0 (page A-74) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
8	*Flow-Accelerated Corrosion* program	The *Flow-Accelerated Corrosion* AMP is an existing program that will be enhanced to reassess infrequently used piping systems excluded from the scope of the program **with respect to evaluations performed to determine susceptibility to both wall thinning due to flow accelerated corrosion and wall thinning due to erosion mechanisms** to ensure adequate bases exist to justify this exclusion for the SPEO.	B2.1.8	Program enhancements for SLR will be implemented six months prior to the SPEO.

SLRA Appendix B2.1.1 (page B-21) is revised as follows:

B2.1.1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

Program Description

The *ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* AMP is an existing condition monitoring program that manages the aging effects of cracking, loss of fracture toughness, and loss of material. **Wall thinning is managed through periodic augmented inservice inspection examinations for locations where the aging effect has been identified as a concern.** The program consists of periodic volumetric, surface, and/or visual examination, and leakage tests of ASME Code, Section XI Class 1, 2, and 3 pressure retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure retaining bolting for assessment, identification of signs of degradation, and establishment of corrective actions.

SLRA Appendix B2.1.1 (pages B-25 and B-26) is revised as follows:

B2.1.1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

Operating Experience

3. In November 2013, a reactor coolant system pressure boundary leak was identified at a high pressure injection line nozzle during a Unit 1 reactor building walkdown. An entry was made into the corrective action program and shutdown to Mode 5 was initiated in accordance with Technical Specifications. The leak was from a circumferential crack in the safe end-to-pipe butt weld located between the high pressure injection nozzle and the reactor coolant system loop injection block valve.

   A twenty one day outage was required to complete the initial investigation and the weld repairs. A root cause performed determined the cause was mechanical vibration-induced high-cycle fatigue which caused a through-wall crack. Contributing causes were that the system design resulted in higher than desired system vibration on the high pressure injection line and inadequate administrative guidance for conduct of augmented examinations. During analyses used to support the root cause evaluation for the safe end-to-pipe leak in the fall of 2013, the potential for flow cavitation (and subsequent erosion of the high pressure injection piping elbows immediately downstream of the inline flow orifices) was identified. This cavitation degradation is believed to be the result of the high pressure injection full flow tests that are performed at the end of each refueling outage to verify the operation of the high pressure injection system and confirm it can deliver the required flow rates for various high pressure injection nozzle configurations. Cavitation is likely with flow rates above 100 gpm through the 0.78” orifices.

   Corrective actions were initiated to prevent recurrence including the following:

   - Leaking weld replaced
   - Increased frequency of high pressure injection piping ultrasonic inspections
   - Inspected all high pressure injection lines using diverse techniques

- Implemented a means, in program governance procedures, to ensure the critical elements of the inspection procedures used to implement augmented inspections are incorporated into superseding procedures
- Revised all program procedures to incorporate root cause lessons learned

As outlined above, significant program improvements were made including increased rigor and improved basis documentation information for all augmented ISI examinations performed at ONS. These AMP improvements were made to ensure aging effects are properly evaluated and future examinations are scheduled commensurate with the safety significance of the component. The following two augmented exams were added for ONS Unit 1, 2 and 3 to manage aging effects and prevent recurrence.

- High pressure injection nozzle volumetric inspections of safe end to pipe butt welds to identify cracking.
- High pressure injection system piping elbow ultrasonic testing wall thickness measurements for wall thinning due to cavitation ~~(NOTE: Since these examinations are required to manage the potential aging effect for cavitation erosion of the elbows just downstream of the high pressure injection flow orifices, these are included as part of the Flow Accelerated Corrosion AMP)~~.

OE example 3 provides objective evidence that the corrective action program was used effectively to identify the cause of the weld failure, identify the population of welds within the extent of condition, and implement appropriate comprehensive corrective actions, including additional augmented examinations and programmatic enhancements.

SLRA Appendix B2.1.8 (page B-82 as revised by Supplement 1 (ML21302A208)) is revised as follows:

B2.1.8 FLOW-ACCELERATED CORROSION

Program Description

The program also manages wall thinning caused by mechanisms other than flow-accelerated corrosion in copper alloy, ~~copper alloy (>15%Zn),~~ steel and stainless steel piping and piping components exposed to ~~closed-cycle cooling water, raw water,~~ treated water and treated borated water environments in situations where periodic monitoring is used in lieu of eliminating the cause of various erosion mechanisms.

SLRA Appendix B2.1.8 (page B-83) is revised as follows:

B2.1.8 FLOW-ACCELERATED CORROSION

Enhancements

The following enhancement will be implemented in the following program element: Detection of Aging Effects (Element 4)

1. Reassess infrequently used piping systems excluded from the scope of the program **with respect to evaluations performed to determine susceptibility to both wall thinning due to flow accelerated corrosion and wall thinning due to erosion mechanisms** to ensure adequate bases exist to justify this exclusion for the SPEO.

SLRA Appendix B2.1.11 (page B-99) is revised as follows:

B2.1.11 OPEN-CYCLE COOLING WATER SYSTEM

Program Description

The program manages piping, piping components, and heat exchanger components in safety related and non-safety related raw water systems that are exposed to a raw water environment for loss of material, cracking, reduction of heat transfer, and flow blockage. **Wall thinning of the exterior (closed-cycle cooling water) side of the RCW Heat Exchanger tubes is also managed by the *Open-Cycle Cooling Water System* program. Eddy current testing is credited for managing wall thinning on the closed-cycle cooling water (exterior) side of the RCW Heat Exchanger tubes.** The program also manages loss of coating integrity for certain components that do not perform a pressure boundary intended function and where loss of coating integrity would not impact the intended functions of downstream components. System and component testing, flushing, visual inspections, and nondestructive examination are conducted to ensure that identified aging effects are managed such that system and component intended functions and integrity are maintained.

SLRA Appendix B2.1.22 (pages B-162 and B-163) is revised as follows:

B2.1.22 ASME CODE CLASS 1 SMALL-BORE PIPING

Operating Experience

2. In November 2013, a reactor coolant system pressure boundary leak was identified at a high-pressure injection line nozzle during a Unit 1 reactor building walkdown. An entry was made into the corrective action program and shutdown was initiated in accordance with Technical Specifications.

   The leak was from a circumferential crack in the safe end to pipe butt weld located between the high-pressure injection nozzle and the reactor coolant system loop injection block valve. A root cause evaluation determined that mechanical vibration induced high cycle fatigue caused a through wall crack. A system design that resulted in higher than desired system vibration on the high-pressure injection line and inadequate administrative guidance for conduct of augmented examinations were identified as contributing causes. During analyses used to support the root cause evaluation for the safe end to pipe leak, the potential for flow cavitation and subsequent erosion of the high- pressure injection piping elbows immediately downstream of the in-line flow orifices was identified. This cavitation degradation is believed to be the result of the high pressure injection full flow tests that are performed at the end of each refueling outage to verify the operation of the high-pressure injection system and confirm it can deliver the required flow rates for various high-pressure injection nozzle configurations. Cavitation is likely with flow rates above 100 gpm through the 0.78" orifices.

Corrective actions were initiated to prevent recurrence including the following:

- Leaking weld replaced
- Increased frequency of high-pressure injection nozzle safe end-to pipe butt weld ultrasonic inspections for lines that have higher vibration potential
- Inspected all high-pressure injection lines using diverse techniques
- Implemented a means, in program governance procedures, to ensure the critical elements of the inspection procedures used to implement augmented inspections are incorporated into revised or new superseding procedures
- Revised all program procedures to incorporate root cause lessons learned

As outlined above, significant program improvements were made including increased rigor and improved basis documentation information for all augmented ISI examinations performed at ONS. These AMP improvements were made to ensure aging effects are properly evaluated and future examinations are scheduled commensurate with the safety significance of the component.

The following two augmented exams were added for Oconee Unit 1, 2 and 3 to manage aging effects and prevent recurrence.

- High pressure injection nozzle volumetric inspections of safe end to pipe butt welds to identify cracking
- High pressure injection system piping elbow ultrasonic testing wall thickness measurements for wall thinning due to cavitation (NOTE: Cavitation erosion of the elbows just downstream of the high-pressure injection flow orifices is managed by the ~~*Flow Accelerated Corrosion* (B2.1.8)~~ ***ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* AMP (B2.1.1)** AMP).

Although high-cycle fatigue is typically considered a design issue rather than aging, this OE is conservatively considered evidence that cracking of small-bore butt welds could occur at ONS Unit 1 and, therefore, periodic inspections will be performed in accordance with Category C, as defined in Table XI.M35-1 of NUREG-2191. This OE example also provides objective evidence that the corrective action program was used effectively to identify the cause of the weld failure, identify the population of welds within the extent of condition, and implement appropriate comprehensive corrective actions, including increased frequency of ISI augmented examinations and programmatic enhancements.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 35
RAI B2.1.16-1

Enclosure 1, Attachment 35

RAI B2.1.16-1:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Table XI.M27-1 of NUREG-2191, Volume 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report" (ADAMS Accession No. ML17187A204), recommends that main drain tests follow Section 13.2.5 of NFPA 25. Section 13.2.5 of NFPA 25 requires "main drain tests to be conducted annually at each water-based fire protection system riser to determine whether there has been a change in the condition of the water supply piping and control valves." It also states, "When there is a 10 percent reduction in full flow pressure when compared to the original acceptance test or previously performed tests, the cause of the reduction shall be identified and corrected if necessary." Subsequent License Renewal Application (SLRA) Supplement 2, dated November 11, 2021 (ADAMS Accession No. ML21315A012), revised Enhancement 9 in SLRA Sections A2.16 and B2.1.16 and SLRA Table A6.0-1 to state, in part, "When there is a ten percent reduction in full flow pressure when compared to an established baseline value, the cause of the reduction shall be identified and corrected if necessary."

Issue:
It is unclear whether "an established baseline value" is referring to an original acceptance test (or comparable test result). If it is not referring to an original acceptance test (or comparable test result), it is unclear how the baseline value will be established. The test-to-test pressure monitoring should be capable of identifying significant degradation of the fire water system supply over several years.

Request:
1. Clarify whether "an established baseline value" is referring to an original acceptance test (or comparable test result).
2. If "an established baseline value" is not referring to an original acceptance test (or comparable test result), then discuss how the baseline value will be established and how it will result in identifying significant degradation of the fire water system supply over several years.

Response to RAI B2.1.16-1:

Request 1:

Original acceptance testing (i.e., construction era pre-service testing) results would not provide a valid baseline for the main drain testing acceptance criteria due to numerous modifications of the system that have been performed since initial construction. The results of initial testing will be used to establish the baseline value referenced in Enhancement 9 of the Oconee Fire Water System AMP. When there is a ten percent reduction in full flow pressure when compared to this baseline value, the cause of the reduction shall be identified and corrected if necessary. This is considered comparable to the acceptance criteria defined in NFPA 25, 2011 Edition, Section 13.2.5.

Request 2:

As stated above, the baseline value referred to in Enhancement 9 will be established using the results of initial testing for each deluge system main drain which is considered comparable to the acceptance criteria defined in NFPA 25, 2011 Edition, Section 13.2.5. Relying on a baseline value to determine the acceptance criteria ensures that a cumulative reduction in full flow pressure of 10 percent over multiple performances of the test will be identified and appropriate corrective actions will be taken. This addresses the NRC's concern that significant degradation of the fire water system supply that occurs over several years could go undetected using the standard acceptance criteria in NFPA 25, which allows for comparison to the previously performed test results.

SLRA Revisions:

None

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
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ATTACHMENT 36
RAI B2.1.16-2

Enclosure 1, Attachment 36

RAI B2.1.16-2:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Enhancement 5 to the Fire Water System program in SLRA Sections A2.16 and B2.1.16 and SLRA Table A6.0-1 states, “Revise inspection procedures to provide additional inspection guidance regarding age-related degradation and to include inspection parameters for items such as lighting, distance, offset, presence of protective coatings, and cleaning processes.” Revision 1 of SLR-ONS-AMPR-XI.M27, “Fire Water System AMP [Aging Management Program] Evaluation Report,” notes that this enhancement affects the “detection of aging effects” program element and notes that the program will be enhanced to include periodic internal visual inspections of sprinkler and deluge system branch line piping for flow blockage due to fouling. In addition, follow-up volumetric wall thickness examinations will be performed when internal visual inspections identify surface irregularities indicative of unacceptable degradation.

Issue:
Although Revision 1 of SLR-ONS-AMPR-XI.M27 provides examples of inspection parameters that will be added as part of Enhancement 5, the SLRA does not provide any description of the additional inspection guidance for age-related degradation to be added to the inspection procedures. The vagueness of the procedure change described in the SLRA does not seem to provide for future verification activities by the NRC or confirmation that commitments have been completed. In addition, controls for future changes to the program through the 10 CFR 50.59 process may not be able to be assured without a sufficiently comprehensive description of the enhancement.

Request:
Clarify what additional inspection guidance for age-related degradation is being added to the Fire Water System inspection procedures.

Response to RAI B2.1.16-2:

The Fire Water System program includes existing internal and external inspections of station fire suppression equipment. However, the procedures for these inspections may not consistently provide guidance on the identification of aging degradation. The intent of the enhancement is to ensure that the inspection procedures for the Fire Water System program contain specific guidance related to inspecting for and documenting indications of age-related degradation, including corrosion and flow blockage due to fouling. The following changes shown below are being made to the SLRA.

SLRA Revisions:

SLRA Section A2.16 (page A-18) is revised as follows:

A2.16 Fire Water System

Enhancements

5. Revise inspection procedures to ~~provide additional inspection guidance regarding age-related degradation and to~~ include inspection parameters for items such as lighting, distance, offset, presence of protective coatings, and cleaning processes **and to provide specific guidance for the identification and documentation of indications of age-related degradation. For internal surfaces this includes indications of corrosion such as surface irregularities and signs of fouling which could lead to flow blockage. For external surfaces age-related degradation includes indications of corrosion beyond minor surface rusting and signs of current or past leakage**.

SLRA Section B2.1.16 (page B-127) is revised as follows:

B2.1.16 Fire Water System

Enhancements

5. Revise inspection procedures to ~~provide additional inspection guidance regarding age-related degradation and to~~ include inspection parameters for items such as lighting, distance, offset, presence of protective coatings, and cleaning processes **and to provide specific guidance for the identification and documentation of indications of age-related degradation. For internal surfaces this includes indications of corrosion such as surface irregularities and signs of fouling which could lead to flow blockage. For external surfaces age-related degradation includes indications of corrosion beyond minor surface rusting and signs of current or past leakage**.

SLRA Table A6.0-1 (page A-83) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
16	*Fire Water System* program	5. Revise inspection procedures to ~~provide additional inspection guidance regarding age-related degradation and to~~ include inspection parameters for items such as lighting, distance, offset, presence of protective coatings, and cleaning processes **and to provide specific guidance for the identification and documentation of indications of age-related degradation. For internal surfaces this includes indications of corrosion such as surface irregularities and signs of fouling which could lead to flow blockage. For external surfaces age-related degradation includes indications of corrosion beyond minor surface rusting and signs of current or past leakage**. 6. Perform flow testing of at least one hose station in each building every five years to demonstrate the capability to provide the design pressure at required flow. Flow testing will be performed at the hydraulically most remote hose station, or if an alternative hose station is tested, the acceptance criteria for the test will account for the additional head loss that would occur if the hydraulically most remote hose station were tested such that the results of the flow test are representative of the limiting location. If acceptance criteria are not met, at least two additional tests shall be performed within five years. If subsequent tests do not meet acceptance criteria, an extent of condition and extent of cause analysis is conducted to determine the further extent of tests. The additional tests include at least one test at one of the other units with the same material, environment, and aging effect combination. 7. Perform external visual inspections of the elevated water storage tank in accordance with Section 9.2.5.5 of NFPA 25, 2011 Edition at least once every two years. 8. Perform flushing of the mainline strainers following system actuation in accordance with Section 10.2.7 of NFPA 25, 2011 Edition.	B2.1.16	Program enhancements for SLR will be implemented 6 months prior to the SPEO. Inspections or tests that are to be completed prior to SPEO are completed 6 months prior to the SPEO or no later than the last refueling outage prior to the SPEO.

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ATTACHMENT 37
RAI B2.1.16-3

Enclosure 1, Attachment 37

RAI B2.1.16-3:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Table XI.M27-1 in Volume 2 of NUREG-2191 recommends internal inspections of piping consistent with Section 14.2 of NFPA 25, which requires piping and branch lines be inspected internally for foreign organic and inorganic material.

Exception 3 to the “detection of aging effects” program element in SLRA Section B2.1.16 states, “Periodic internal inspections of the sprinkler system branch lines within the scope of the Oconee Fire Water System AMP will not be performed.” The justification for the exception states that the only in scope sprinkler systems are the manually actuated dry pipe sprinkler systems for the cable room, cable shaft level 3, and cable shaft level 4 & 5; and that they are “not subject to significant internal corrosion or flow blockage due to fouling since they are maintained dry and are not subject to periodic wetting during testing.” In addition, the justification states that plant operating experience did not show instances where these systems were actuated.

Section 4.4 of SLR-ONS-AMPR-XI.M27 states, “The sprinkler systems for the cable room, cable shaft level 3, and cable shaft level 4 & 5 are not subject to internal corrosion nor flow blockage due to fouling since they are maintained dry and are not subject to periodic wetting during testing. Internal inspections of these systems will not be performed.”

Issue:
Given that the internal inspections in Section 14.2 of NFPA 25 apply to both the piping and branch lines it is unclear whether Exception 3 applies to both the piping and branch lines because the Exception states “sprinkler system branch lines” and the Fire Water System AMP Evaluation Report states “Internal inspections of these systems will not be performed.” In addition, if Exception 3 only applies to the branch lines, it is unclear, based on information in the SLRA and documents reviewed during the Fire Water System program audit if the program currently includes internal inspections of the piping.

Request:

1. Discuss whether Exception 3 applies to both the piping and branch lines of the manually actuated dry pipe sprinkler systems for the cable room, cable shaft level 3, and cable shaft level 4 & 5, or if it only applies to the branch lines.

2. If Exception 3 only applies to the branch lines, does the Fire Water System program currently include internal inspections of the piping of the manually actuated dry pipe sprinkler systems

for the cable room, cable shaft level 3, and cable shaft level 4 & 5, consistent with Section 14.2 of NFPA 25?

Response to RAI B2.1.16-3:

Request 1:

Exception 3 applies to all system piping that is maintained dry between the first block isolation valve and the sprinklers in the manually actuated dry sprinkler systems for the cable room, cable shaft level 3, and cable shaft level 4 & 5 rooms.

Request 2:

Exception 3 applies to all system piping that is maintained dry between the first block isolation valve and the sprinklers in the manually actuated dry sprinkler systems for the cable room, cable shaft level 3, and cable shaft level 4 & 5 rooms. As justified in Exception 3, the Fire Water System program does not currently include internal inspections of the piping of the manually actuated dry pipe sprinkler systems for the cable room, cable shaft level 3, and cable shaft level 4 & 5.

SLRA Revisions:

SLRA Appendix B2.1.16 (pages B-125 and B-126) is revised as follows:

Exception 3 to NUREG-2191

Program Element Affected: Detection of Aging Effects (Element 4)

3. NUREG 2191, Table XI.M27-1 recommends periodic internal inspections of sprinkler system ~~branch line~~ piping consistent with Section 14.2 of NFPA 25, 2011 Edition. Periodic internal inspections of the sprinkler system ~~branch lines~~ **piping** within the scope of the Oconee *Fire Water System* AMP will not be performed since this piping is maintained dry between the first block isolation valve and the sprinklers.

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ATTACHMENT 38
RAI B.2.3.2-1

Enclosure 1, Attachment 38

RAI B.2.3.2-1

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. To complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
SLRA Table 3.1.2-4 has the AMR items for steam generator (SG) components. During the in-office audit, the staff asked about the environments assigned to the SG tube-to-tubesheet welds. In response to the breakout questions, SLRA Supplement 1 (ML21302A208), Attachment 8, proposed SLRA revisions that changed the description of the environment for these welds. SLRA Supplement 3 (ML21349A005), Attachments 6 and 8, also made changes to AMR line items for the SG tube-to-tubesheet welds in SLRA Table 3.1.2-4. Following is a summary of the changes in Supplements 1 and 3:

Supplement 1, Attachment 8:
- Changed the environment from “Reactor Coolant (Internal)” to “Secondary Feedwater (Internal)”. This change applies to the line items for “Cracking,” “Cumulative Fatigue Damage,” and “Loss of Material.”
- Changed the environment from “Secondary Feedwater (External)” to “Reactor Coolant (External).” This change applies to the line items for “Cracking” and “Loss of Material.”

Supplement 3:
- In Attachment 8, a new AMR item was added for “Cumulative Fatigue Damage,” to be managed by TLAA, for “Secondary Feedwater (Internal).” This AMR line uses the SLRA Table 3.1.2-4 original environment rather than the environment as modified by Supplement 1, Attachment 8.
- In Attachment 6, four of the “Notes” were changed from “A” to “C” for “Cumulative Fatigue Damage” for the “Reactor Coolant (Internal)” environment, and for “Cracking” and “Loss of Material” for the “Secondary Feedwater (External)” environment. These AMR lines use the SLRA Table 3.1.2-4 original environments rather than the environment as modified by Supplement 1, Attachment 8.

Issue:
For the “Environment” changes proposed to SLRA Table 3.1.2-4 in Supplement 1, Attachment 8, corresponding changes were not made to other items in SLRA Table 3.1.2-4, creating what appears to be inconsistencies between those items and the revised environments proposed in the supplement. In addition, changes proposed to SLRA Table 3.1.2-4 in Supplement 3 do not appear to be fully consistent with the Supplement 1 changes. The specific issues are:

a. The changes to “Environment” entries in SLRA Table 3.1.2-4 for the SG “Tube-To-Tube Sheet Welds” in Supplement 1, Attachment 8, were not accompanied by corresponding changes to the “NUREG-2191 Item,” “NUREG-2192 Table 1,” and “Notes.” As a result, the proposed environment changes appear to be inconsistent with the NUREG-2191 and NUREG-2192 items, and the corresponding Notes A.
b. The revisions to SLRA Table 3.1.2-4 for the SG “Tube-to-Tube Sheet Welds,” in Supplement 1, Attachment 8, results in two separate items for cracking in “Reactor Coolant (External),” with different “Aging Management Program,” NUREG-2191, NUREG-2192, and “Notes” entries.
c. The “Environment” changes to SLRA Table 3.1.2-4 proposed for the SG “Tube-to-Tube Sheet Welds” in Supplement 1, Attachment 8, are not included in the changes to SLRA Table 3.1.2-4 proposed in Supplement 3, Attachments 6 and 8. This includes the following:

    - The AMR item added for “Cumulative Fatigue Damage” in Attachment 8 of Supplement 3
    - The change of four of the AMR “Notes” from “A” to “C” in Attachment 6 of Supplement 3

Request:
Provide a single markup showing all proposed revisions to the steam generator “Tube-to Tube Sheet Welds” AMR lines in SLRA Table 3.1.2-4, and a summary description of the aging management for these welds relative to the original SLRA Table 3.1.2-4.

Response to RAI B.2.3.2-1:

SLRA Table 3.1.2-4 is being revised and a single markup showing all the proposed revisions to the steam generator “Tube-to-Tube Sheet Welds” AMR lines (page 3-204) is shown below.

The outside portions of the steam generator Tube-to-Tube Sheet welds are in a reactor coolant environment and are subject to the aging effects of cracking, cumulative fatigue damage, and loss of material. The aging effects of cracking and loss of material will be managed by the *Water Chemistry* and *Steam Generators* programs. The aging effect of cumulative fatigue damage is addressed in SLRA Section 4.3.2.3, *Once Through Steam Generators*, and is managed by the *Fatigue Monitoring* program (TLAA program).

The inside portions of the steam generator Tube-to-Tube Sheet welds, between the tubes and tubesheets, is exposed to a secondary feedwater environment. The applicable aging effects are cracking, and loss of material and these aging effects are managed by the *Steam Generators* and *Water Chemistry* programs.

SLRA Revisions:

SLRA Table 3.1.2-4 revisions are shown in their entirety in response to RAI B2.1.10-2 (Enclosure 1, Attachment 19).

SLRA Table 3.1.2-4 (page 3-204) is revised as follows and repeated here for convenience of review:

Table 3.1.2-4 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Steam Generators - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Tubesheet	Pressure Boundary	Steel with Nickel Alloy Cladding	Reactor Coolant (Internal)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D2.RP-47	3.1.1-042	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.RP-47	3.1.1-042	~~A~~ C
				Cumulative Fatigue Damage	TLAA	IV.D2.R-222	3.1.1-008	C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.R-440	3.1.1-127	A
					Water Chemistry (B2.1.2)	IV.D2.R-440	3.1.1-127	A
Tube-to-Tube Sheet Welds	Structural Support	Nickel Alloy	Reactor Coolant (External)	Cracking	Water Chemistry (B2.1.2)	IV.D2.RP-185	3.1.1-025	A
			Reactor Coolant ~~(Internal)~~ **(External)**	Cracking	Steam Generators (B2.1.10)	IV.D2.RP-185	3.1.1-025	A
				Cumulative Fatigue Damage	TLAA	IV.D2.R-46	3.1.1-002	~~A~~ C
				Loss of Material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A
			Secondary Feedwater ~~(External)~~ **(Internal)**	Cracking	Steam Generators (B2.1.10)	IV.D2.R-47	3.1.1-069	~~A~~ C
					Water Chemistry (B2.1.2)	IV.D2.R-47	3.1.1-069	~~A~~ C
				Loss of Material	Steam Generators (B2.1.10)	IV.D2.RP-233	3.1.1-077	~~A~~ C

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ATTACHMENT 39
RAI 3.4.2.2.2-1

Enclosure 1, Attachment 39

RAI 3.4.2.2.2-1:

Regulatory Basis:
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the matters identified in 10 CFR 54.21(a)(1) and 10 CFR 54.21(a)(2), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:
Oconee SLRA Table 3.4.1 is the summary of aging management programs for steam and power conversion systems, and it includes GALL-SLR Items 3.4.1-002 and 3.4.1-003. Item 3.4.1-002 addresses cracking due to stress corrosion cracking (SCC) for stainless steel piping, piping components, and tanks exposed to air or condensation. Table 3.4.1 states that this item requires further evaluation in SLRA Section 3.4.2.2.2. The discussions in Table 3.4.1 and SLRA Section 3.4.2.2.2 state that this item is not applicable because Oconee has no stainless steel piping, piping components, or tanks exposed to air or condensation in the scope of SLR in steam and power conversion systems, and that components in the steam and power conversion systems are insulated and aligned to GALL-SLR Item 3.4.1-104. For Item 3.4.1-104, the SLRA proposes to use the One-Time Inspection program to confirm SCC of insulated stainless steel piping, piping components, and tanks exposed to air or condensation is not occurring.

Item 3.4.1-003 addresses loss of material due to pitting or crevice corrosion for stainless steel and nickel alloy piping, piping components, and tanks exposed to air or condensation. Table 3.4.1 states that this item requires further evaluation in SLRA Section 3.4.2.2.3. The discussions in Table 3.4.1 and SLRA Section 3.4.2.2.3 state that the One-Time Inspection program will be implemented to confirm pitting and crevice corrosion of stainless steel piping, piping components, and tanks exposed to air or condensation are not occurring.

Issue:
GALL-SLR Items 3.4.1-002 and 3.4.1-003 address two different aging mechanisms for stainless steel piping, piping components, and tanks exposed to air or condensation. The discussions for these items in SLRA Table 3.4.1, SLRA Section 3.4.2.2.2, and SLRA Section 3.4.2.2.3, appear to be inconsistent with respect to the applicability of these items to Oconee SLR. For cracking due to SCC, the SLRA states that all of the components are insulated and Item 3.4.1-002 is not applicable. For crevice and pitting corrosion, the SLRA states that aging management for these components is consistent with the GALL-SLR, suggesting not all components are insulated. Because of this apparent inconsistency, it is not clear to the staff that the SLR aging management evaluation addresses SCC for all stainless steel piping, piping components, and tanks exposed to air or condensation in the steam and power conversion systems.

Request:
Clarify the apparent discrepancy in the aging management discussions for GALL-SLR Items 3.4.1-002 and 3.4.1-003 with respect to whether Oconee has uninsulated stainless steel piping, piping components, or tanks in the scope of SLR exposed to air or condensation in steam and power conversion systems. In addition, revise SLRA Table 3.4.1, SLRA Sections 3.4.2.2.2 and 3.4.2.2.3, and SLRA Tables 3.4.2-1 through 3.4.2-12, as appropriate, for consistency in the use of GALL-SLR Items 3.4.1-002 and 3.4.1-003.

Response to RAI 3.4.2.2.2-1:

There are no uninsulated stainless steel or nickel alloy components in the Steam and Power Conversion Systems exposed to an external air or condensation environment. All piping and piping components within the Steam and Power Conversion Systems are insulated and were aligned to NUREG-2192 Table 1 items 3.4.1-103 or 3.4.1-104 except for the Low Pressure Turbine Extraction System nickel alloy expansion joints. These expansion joints were incorrectly aligned to NUREG-2192 Table 1 item number 3.4.1-003. SLRA Table 3.4.2-6 is amended to align the insulated component to NUREG-2192 Table 1 item 3.4.1-103. As a result, conforming changes to SLRA Section 3.4.2.2.3 and Table 3.4.1 are also made.

SLRA Revisions:

SLRA Section 3.4.2.2.3 (page 3-1072) is revised as follows:

[3.4.1-003] - ~~The *One-Time Inspection* (B2.1.20) program will be implemented to confirm pitting and crevice corrosion of stainless steel piping, piping components, and tanks exposed to air or condensation is not occurring.~~ **ONS has no uninsulated stainless steel or nickel alloy piping, piping components, or tanks exposed to air or condensation in the scope of SLR in steam and power conversion systems. Components in the steam and power conversion system are insulated and are aligned to item 3.4.1-103.**

[3.4.1-103] – The *One-Time Inspection* (B2.1.20) program will be implemented to confirm pitting and crevice corrosion of insulated stainless steel **and nickel alloy** piping, piping components, and tanks exposed to air or condensation is not occurring.

SLRA Table 3.4.1 (page 3-1082) is revised as follows:

Table 3.4.1 Summary of Aging Management Programs for Steam And Power Conversion System Evaluated in Chapter VIII of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.4.1-003	Stainless steel, nickel alloy piping, piping components, tanks exposed to air, condensation	Loss of material due to pitting, crevice corrosion	AMP XI.M32, "One-Time Inspection," AMP XI.M36, "External Surfaces Monitoring of Mechanical Components," AMP XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components," or AMP XI.M42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks"	Yes (SRP-SLR Section 3.4.2.2.3)	~~Consistent with NUREG-2191. The *One-Time Inspection* program (B2.1.20) will be implemented to confirm that loss of material is not occurring in stainless steel components exposed to air-indoor uncontrolled (external) environments. See further evaluation in Section 3.4.2.2.3.~~ **Not Applicable. ONS has no uninsulated stainless steel or nickel alloy piping, piping components, or tanks exposed to air or condensation in the scope of subsequent license renewal in the steam and power conversion systems. The associated NUREG-2191 aging items are not used. Components in the steam and power conversion system are insulated and are aligned to item 3.4.1-103.**

SLRA Table 3.4.2-6 (page 3-1199) is revised as follows:

Table 3.4.2-6 Steam and Power Conversion Systems – Low Pressure Turbine Extraction System – Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Expansion Joint	Pressure Boundary	Nickel Alloy	Air – Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B2.1.20)	~~VIII.E.SP-127a~~ **VIII.H.S-451b**	~~3.4.1- 003~~ **3.4.1- 103**	A
			Treated Water (Internal)	Loss of Material	One-Time Inspection (B2.1.20)	VIII.E.SP-87	3.4.1- 085	A
					Water Chemistry (B2.1.2)	VIII.E.SP-87	3.4.1- 085	A
		Stainless Steel	Air – Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B2.1.20)	VIII.H.S-452b	3.4.1- 104	A

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ATTACHMENT 40
RAI B2.1.7-1

Enclosure 1, Attachment 40

RAI B2.1.7-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires that the effects of aging for the SSCs identified therein will be adequately managed such that their intended functions are maintained consistent with the CLB for the subsequent period of extended operation. To complete its review and enable the staff to make a reasonable assurance finding in accordance with 10 CFR 54.29(a) on functionality of reviewed SSCs, the staff requires additional information for the subsequent period of extended operation be provided regarding the matters described below.

Background:
In the gap analysis evaluation of SLRA AMP B2.1.7, the applicant states that the primary-expansion relationship between the Primary-category upper core barrel (UCB), lower core barrel (LCB), and flow-distributor (FD) bolts and the linked Expansion-category upper thermal shield (UTS) and lower thermal shield (LTS) bolts/studs is for the aging mechanism of stress corrosion cracking (SCC) only. In the MRP-227, Rev. 1-A report (ADAMS Accession No. ML20175A112), the EPRI MRP's sampling-based methodology for increasing (i.e., expanding) a set of Primary-component inspections to a set of defined Expansion-category components is defined in Chapter 5 of the report. For expansion-based relationships between Primary-category UCB, LCB, and FD bolts and Expansion-category UTS and LTS bolts, the specific criteria for expanding the ultrasonic test (UT) inspections to the UTS and LTS bolts (i.e., expansion-link criteria for the UTS and LTS bolts) are defined in Line Items B7, B8 and B12 of Table 5-1 in the MRP-227, Rev. 1-A report. For the revised expansion-link basis of UTS and LTS bolts under the AMP's gap analysis methodology, the applicant would no longer need to expand the UT inspections to the UTS and LTS bolts if the cause of any crack-like indications in the Primary UCB, LCB, or FD bolting was confirmed to be caused (i.e., initiated or grown) by a fatigue mechanism.

Issue:
The proposed expansion-link basis for the Expansion-category LTS bolts and UTS bolts adds in a new expansion-link methodology criterion that is currently outside the scope of the current EPRI MRP methodology for component-specific examinations and expansions in Chapter 5 and Table 5-1 of MRP-227, Rev. 1-A. More specifically, the expansion bases in Line Items B7, B8 and B12 of Table 5-1 to the MRP-227, Rev. 1-A report only apply if noted conditions of cracking are detected in either the UCB bolts, LCB bolts or FD bolts and the number of UCB, LCB, or FD bolts with detected crack-like conditions (including previously failed/removed bolts) are determined to exceed 10% of the total population of the Primary bolting type; aging mechanism determinations are not specified or included in these Table 5-1 items as additional bases for establishing whether expansion activities for the LTS and UTS bolts are necessary.

In addition, the UT inspections applied to the Primary-category UCB, LCB, and FD bolts may not be capable of generating UT signals that can distinguish cracking induced by SCC from cracking induced by a fatigue mechanism. Thus, the change in the referenced expansion-link basis deviates from EPRI MRP assumptions and methodology for the MRP-227-based program.

The staff needs additional information to justify the proposed expansion-link basis for the LTS and UTS bolts.

Request:

1. Provide the justification for using the proposed expansion-link methodology for the LTS bolts and UTS bolts given that Line Items B7, B8, and B12 in Table 5-1 of MRP-227, Rev. 1-A (or in Table 5-1 of MRP-227, Rev. 2) do not establish the need for expansion for the LTS bolts and UTS bolts based on confirmation of a specific cause (i.e., mechanism) of crack-like conditions that may be detected in any of the referenced Primary bolting types. As part of this justification, discuss why the proposed change in the expansion-link criterion would not need to be identified as an exception to the “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program element criteria in GALL-SLR AMP XI.M16A, PWR Vessel Internals,” as updated in NRC Interim Staff Guidance No. SLR-ISG-2021-01-PWRVI (ADAMS Accession No. ML20217L203).

2. Clarify how the UT inspection methods applied to the Primary-category UCB, LCB, and FD bolts are capable of generating UT signal results that can distinguish cracking caused by an SCC mechanism from cracking caused by a metal fatigue or cyclical loading mechanism. Otherwise, if it is determined that the UT inspection methods cannot generate UT signal results that are capable of distinguishing SCC from fatigue-induced cracking, define the additional activities that would need to be performed under SLRA AMP B2.1.7 to confirm the cause of crack-like indications that may be detected in the UCB, LCB, or FD bolts, and justify why these activities would not need to be included as an enhancement of the “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the AMP.

Response to RAI B2.1.7-1:

Requests 1 and 2:

The pertinent age-related degradation mechanisms for the bolts and studs/nuts applicable to this RAI are listed in Tables 4-1 and 4-4 of MRP-227, Revision 1-A. In MRP-227, Revision 1-A Table 4-4, the only applicable age-related degradation mechanism for the Expansion upper thermal shield (UTS) bolts and lower thermal shield (LTS) studs/nuts is stress corrosion cracking (SCC), which is unchanged by the ONS SLRA.

Expansion to these bolts/studs/nuts is based on relevant indications in the associated Primary items. Per Table 5-1 of MRP-227, Revision 1-A, ultrasonic testing (UT) examination acceptance criteria for the Primary bolts are established as part of the examination technical justification. As noted in MRP-228, Revision 4, a relevant indication is defined as cracking or other significant degradation that could impair the ability of the component to perform its design function, which does not differentiate between specific age-related degradation mechanisms. Therefore, after meeting the expansion criteria in Table 5-1 of MRP-227, Revision 1-A, expansion from Primary bolts (UCB, LCB, FD) would occur to the Expansion bolts/studs/nuts (UTS and LTS) if the Expansion criteria are met.

As noted above, the relevant indication does not differentiate between specific age-related degradation mechanisms, which is conservative. In other words, expansion will occur whether the relevant indications are from SCC or fatigue. Therefore, this does not need to be identified as an exception to the “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program element criteria in GALL-SLR AMP XI.M16A, PWR Vessel Internals, as updated in NRC Interim Staff Guidance No. SLR-ISG-2021-01-PWRVI.

SLRA Revisions:

None

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ATTACHMENT 41
RAI B2.1.7-2

Enclosure 1, Attachment 41

RAI B2.1.7-2:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires that the effects of aging for the SSCs identified therein will be adequately managed such that their intended functions are maintained consistent with the CLB for the subsequent period of extended operation. To complete its review and enable the staff to make a reasonable assurance finding in accordance with 10 CFR 54.29(a) on functionality of reviewed SSCs, the staff requires additional information for the subsequent period of extended operation be provided regarding the matters described below.

Background:
SLRA AMP B2.1.7 maintains the lower thermal shield (LTS) bolts as “Expansion” category components that are linked to UT inspections that will be performed on the Primary-category upper core barrel (UCB) bolts, lower core barrel (LCB) bolts, and flow distributor (FD) bolts, which are made for A-286 stainless steel (SS) materials. During its audit of SLRA AMP B2.1.7, the staff noted that in 1981, the applicant reported past incidents of cracking in the original A-286 stainless steel grade LTS bolts of ONS Unit 1 where the cracking was confirmed to be predominantly initiated by stress corrosion cracking (SCC). One of the corrective actions that resulted from the applicant’s evaluation of this operating experience (OpE) was to replace the LTS bolts with LTS bolts fabricated from X-750 nickel-based alloy material.

Issue:
In MRP-227, Rev. 1-A, the EPRI Materials Reliability Program (MRP) identifies that LTS bolts made from either A-286 SS materials or X-750 Nickel-based alloy materials are susceptible to SCC. The staff noted that the risk-informed gap analysis basis for maintaining the LTS bolts as designated Expansion-category components presumes that any SCC conditions found in the A-286 SS materials of the UCB, LCB, and FD bolts going forward will still be representative and predictive of SCC conditions that may be postulated to occur in LTS bolts made from another SCC-susceptible material (i.e., X-750). Given the susceptibility of X-750 Nickel-based alloy to SCC, the staff needs the following additional information: (1) a justification for why the replacement LTS bolts made from X-750 would not be expected to develop SCC in the manner or to the degree or extent that the original LTS bolts made from A-286 SS had developed SCC; and (2) justification for why the UCB, LCB, and FD bolts made from the same A-286 SS materials would still be considered as lead Primary component indicators for any SCC that is postulated to occur in LTS bolts made from the different X-750 material (such that the replacement LTS bolts could still remain as Expansion-category components for the referenced Primary bolting types).

Request:
Justify why the replacement X-750 LTS bolts would not be expected to develop SCC in the manner or nearly to the extent that the original LTS bolts made from A-286 SS had developed SCC in the past or to the extent that the Primary UCB, LCB, and FD bolts made from A-286 SS (i.e., as the leading Primary components) might be expected or postulated to develop SCC in the future. Clarify how this information supports maintaining the replacement LTS bolts as the “Expansion” category components for SLRA AMP B2.1.7.

Response to RAI B2.1.7-2:

Examinations in the early 1980s of Babcock and Wilcox (B&W) designed units revealed failures in bolts fabricated from Alloy A-286, Condition A (ASME SA-453, Grade 660). The results of an extensive evaluation program revealed the failure mechanism of the Alloy A-286, Condition A bolts to be intergranular stress corrosion cracking (IGSCC). The Babcock and Wilcox Owner's Group (B&WOG) developed the internals bolting surveillance program (IBSP) to characterize both the potential for IGSCC and the extent of stress relaxation in high-strength reactor vessel (RV) internals bolting materials (Alloys A-286 and X-750) in a pressurized water reactor (PWR) environment. The IBSP was comprised of three phases, culminating in removal of the in-reactor capsules and examination of the bolts of similar design to high strength RV internals bolting used to replace failures seen in the early 1980s.

Failures during this testing were observed in some Alloy A-286, Condition A bolts. However, no failures occurred when the peak stress levels were kept below ~70-80% of the yield strength, which is consistent with the design recommendations by the B&WOG from the early 1980s and the recommended stress limits in the EPRI Materials Handbook. The Alloy X-750, HTH Condition bolts performed without failure under all conditions tested with calculated peak stresses up to 293 ksi (2019 MPa), which is well above the material's yield strength. Therefore, in terms of susceptibility, it is reasonable for the Alloy A-286, Condition A bolts to be considered a leading indicator for SCC in the Alloy X-750, HTH Condition bolts. This is supported by decades of operating experience with Alloy X-750, HTH Condition replacement bolts with no observed failures.

Note that the preliminary categorization and Primary/Expansion linkage is based on a combination of susceptibility (likelihood of the degradation mechanism) and severity of consequences. The main reason for the UCB, LCB, and FD bolts leading the UTS and LTS studs/nuts is that the UCB, LCB, and FD bolts have a higher severity of consequence.

The Primary/Expansion relationship for the bolts/studs/nuts is unchanged from MRP-227, Revision 1-A and the SLRA.

SLRA Revisions:

None

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ATTACHMENT 42
RAI B2.1.7-3
[**NON-PROPRIETARY]**

Enclosure 1, Attachment 42
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI B2.1.7-3:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires that the effects of aging for the SSCs identified therein will be adequately managed such that their intended functions are maintained consistent with the CLB for the subsequent period of extended operation. To complete its review and enable the staff to make a reasonable assurance finding in accordance with 10 CFR 54.29(a) on functionality of reviewed SSCs, the staff requires additional information for the subsequent period of extended operation be provided regarding the matters described below.

Background:
In SLRA Section B2.1.7, “PWR Vessel Internals,” the applicant states that it “removed visual VT-3 examination of high strength bolt locking devices.” During its audit, the staff noted that the removal of the VT-3 examinations of high strength bolt locking devices will result in downgrading these bolt locking devices into the “No Additional Measures” (NAM) category of the MRP-227-based AMP.

Issue:
The applicant does not identify in the SLRA which of the specific RVI bolt locking devices are made from high-strength materials, such that the referenced locking device types can be placed in the NAM category of the program and no longer be subject to VT-3 visual inspection criteria. In addition, the applicant’s basis for placing the high-strength RVI bolt locking devices in the NAM category is that the referenced high-strength bolt locking devices no longer screen in for irradiation embrittlement (IE) and that there are no other aging effect/mechanism combinations that would need to be managed during the subsequent period of extended operation. However, in the “*operating experience*” element of SLRA AMP B2.1.7, the applicant identified ONS-specific operating experience (OpE) with fully or partially missing bolt locking device welds in at least one of the high-strength RVI bolt locking device types. Thus, the staff needs further justification regarding the relevant OpE and its impact to the high-strength bolt locking device types being placed into the NAM category.

Request:
Identify which of the RVI high-strength bolt locking devices will no longer receive a VT-3 examination commensurate with the general statement made to remove these inspections. Discuss how the ONS OpE associated with the fully and partially missing bolt locking device welds (as applicable to one of the referenced high-strength bolt locking device types) does not impact the applicant’s conclusion that there are no aging effect and mechanism combinations that need to be managed during the subsequent period of extended operation and that the referenced high-strength bolt locking devices can be placed into the NAM category of the program.

Response to RAI B2.1.7-3:

This RAI pertains to the removal of the VT-3 examination requirement for the locking devices associated with the following high strength bolts:

- B7.Upper Core Barrel (UCB) Bolts
- B8.Lower Core Barrel (LCB) Bolts
- B12.Flow Distributor (FD) Bolts
- B7.1.Upper Thermal Shield (UTS) Bolts
- B8.1.Lower Thermal Shield (LTS) Studs and Nuts

As noted in MRP-189, Revision 3, the locking devices for these bolts/studs/nuts are fabricated from austenitic stainless steel, such as Type 304L (see MRP-189, Revision 3, Table 3-1, pages 3-13, 3-16, and 3-20).

Page B-79 of the ONS SLRA discusses one LCB bolt locking device was found with a missing weld on one side and a small weld on the other side. This operating experience is likely not due to age-related degradation but was included for completeness.

[

]b,c,e Therefore, it is not necessary to continue performing a VT-3 examination of these Category A locking devices.

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

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ATTACHMENT 43
RAI B2.1.7-4
[**NON-PROPRIETARY]**

Enclosure 1, Attachment 43
[Non-Proprietary]

Note: Text that is within brackets is proprietary to Framatome, Inc.

RAI B2.1.7-4:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires that the effects of aging for the SSCs identified therein will be adequately managed such that their intended functions are maintained consistent with the CLB for the subsequent period of extended operation. To complete its review and enable the staff to make a reasonable assurance finding in accordance with 10 CFR 54.29(a) on functionality of reviewed SSCs, the staff requires additional information for the subsequent period of extended operation be provided regarding the matters described below.

Background:
In SLRA AMP B2.1.7 and its supporting documentation, the applicant has downgraded the following CB cylinder components into the “No Additional Measures” (NAM) category of the program: (1) Unit 1, 2, and 3 CB cylinder plates and flange, (2) circumferential seam welds associated with the Unit 1 and 3 CB cylinder top flange components, (3) CB upper cylinder-to-lower cylinder circumferential seam welds in Units 1 and 3, (4) CB cylinder axial seam welds in Units 1, 2, and 3. For this basis, applicant is using component-specific fabrication practices to justify: (1) that there are no longer any aging mechanisms that screen in for these CB cylinder components, and (2) that the specified CB cylinder components can be placed into the NAM category.

(Note: The RAI does not apply to the Unit 2 CB cylinder top flange circumferential seam welds and the Unit 2 CB upper cylinder-to-lower cylinder circumferential seam weld being elevated to “Primary” category status as of the third revision of the formal Framatome gap analysis (Ref. 4)).

Issue:
The staff noted that the applicant’s conclusion that “no aging effects/mechanisms of management” and placing these CB cylinder components into the NAM category is not consistent with the conclusions in MRP-189, Rev. 3. For example, the staff confirmed that the neutron fluence for 80 years of operation for the stainless steel CB cylinder and cylinder seam welds referenced in ANP-3899P, Revision 0, are projected to exceed the thresholds for IE in MRP-189, Revision 3. Thus, further justifications will be necessary for placing these referenced CB cylinder component into the NAM category of the program.

Request:
For those ONS CB cylinder top flange circumferential seam welds, CB cylinder axial seam welds, and CB upper cylinder-to-lower cylinder circumferential seam welds that are being placed into the NAM category of the program, justify how the fabrication practices support the placement of these welds into the NAM category, considering that these practices are associated with weld configurations in non-irradiated, pre-service component conditions.

As part of this clarification, the staff requests that the applicant identify the projected irradiation exposures (i.e., projected neutron fluence [E> 1.0 MeV] and disintegrations per atom (*dpa)* exposure) and applicable stress loadings for these CB cylinder weld components through 72 EFPY so that the staff can assess them to the corresponding susceptibility thresholds in Table 3-1 of MRP-189, Rev. 3 for the following aging mechanisms: (1) stress corrosion cracking (SCC); (2) irradiation-assisted stress corrosion cracking (IASCC); (3) fatigue, and (4) irradiation embrittlement (IE).

If the 72 EFPY projected irradiation values or loading values (or needed combination of the values) for the CB cylinder welds exceed the criteria in MRP-189, Rev. 3, justify that cited pre-service fabrication practice remains valid to support the conclusion that specific aging mechanism(s) do not need to be age-managed in the applicable ONS CB cylinder seam weld types that are being placed into the NAM category (i.e., for the version of the AMP that will be implemented during the subsequent period of extended operation).

<u>References for the RAIs (Reference 4 Was Only Provided in the Audit Portal)</u>

1. EPRI Non-Proprietary Report No. 3002017168, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A),” June 2020 (ADAMS Accession No. ML20175A112).
2. EPRI Proprietary Report No. 3002013218, “Materials Reliability Program: Screening, Categorization, and Ranking of Babcock & Wilcox-Designed Pressurized Water Reactor Internals Component Items and Welds (MRP-189, Revision 3),” December 2019 (ADAMS Accession No. ML20019K284).
3. Framatome Proprietary Report No. ANP-3899P, Revision 0, “Framatome Reactor Vessel Internals TLAA Input to the ONS SLRA,” May 2021. (ADAMS Accession No. ML21158A200).
4. Framatome Proprietary Report No. 51-9312330-003, “Oconee-Specific RV Internals Aging Management Strategy Development and Inspection Categorization for SLR,” November 11, 2021.

Response to RAI B2.1.7-4

The following subsections address each of the age-related degradation mechanisms for the welds of interest: irradiated-assisted stress corrosion cracking (IASCC), stress corrosion cracking (SCC), fatigue, and irradiation embrittlement (IE). The purpose of these subsections is to summarize the justification for the No Additional Measures categorization for the weld/age-related degradation mechanism combination.

<u>IASCC</u>

For the projected dose of **[ ]**$_{b,d,e}$ at 80 years for the core barrel cylinder top and bottom cylinder vertical seam welds and core barrel top cylinder-to-bottom cylinder circumferential seam weld, the two estimated values of residual stress do meet the IASCC stress criterion (Reference: MRP-189, Revision 3, Section 3.3, Item D). The projected dose of the top core barrel cylinder-to-top core barrel flange circumferential seam weld is less than 3 dpa; therefore, IASCC is not applicable for this weld. Following the FMECA, the core barrel cylinder top and bottom cylinder vertical seam welds and core barrel top cylinder-to-bottom cylinder circumferential seam weld are safety preliminary Category B and C for IASCC, respectively, as shown in Table 4-4 of MRP-189, Revision 3.

IASCC has been addressed for the core barrel cylinder top and bottom cylinder vertical seam welds, and it is justified and concluded these welds at all three Oconee Units are downgraded to No Additional Measures.

IASCC has been addressed for the core barrel top cylinder-to-bottom cylinder circumferential seam weld, and it is justified and concluded these welds at Oconee Units 1 and 3 are downgraded to No Additional Measures. This justification is not applicable for this weld at Unit 2; therefore, this weld is made Primary.

SCC

SCC has been addressed for the welds, and the welds are **[**

]b,c,e

Note that the Primary item area of concern associated with the top core barrel cylinder-to-top core barrel flange circumferential seam weld for Oconee Unit 2 is the weld heat-affected zone (HAZ), not the weld itself. Due to different record search results, the top core barrel cylinder-to-top core barrel flange circumferential seam weld HAZ for Oconee Units 1 and 3 are No Additional Measures. Details are provided in reference 4 above.

Fatigue

A new fatigue ranking process was used for the development of MRP-189, Revision 3, as described in Section 3.2 Item H and Section 3.3 Item F. The results of this ranking process, combined with safety consequence, resulted in the core barrel with a focus on the cylinder (including circumferential weld) and top flange (including circumferential weld to the top core barrel cylinder), being recommended as Primary for fatigue for SLR.

A cumulative usage factor (CUF) value for the Oconee Units 1, 2, and 3 core barrel was recently calculated. As a result of this calculation, fatigue is downgraded to No Additional Measures for the core barrel for all Oconee Units.

IE

IE has been addressed for the core barrel cylinder, and **[**

]b,c,e .

Therefore, for Oconee Units 1 and 3, IE for the core barrel cylinder and top flange is considered No Additional Measures (N).

SLRA Revisions:

None

Associated Enclosures:

See Enclosure 2, Attachment 1 for the Framatome Affidavit.

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ATTACHMENT 44
RAI B2.1.7-5

Enclosure 1, Attachment 44

RAI B2.1.7-5:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires that the effects of aging for the SSCs identified therein will be adequately managed such that their intended functions are maintained consistent with the CLB for the subsequent period of extended operation. To complete its review and enable the staff to make a reasonable assurance finding in accordance with 10 CFR 54.29(a) on functionality of reviewed SSCs, the staff requires additional information for the subsequent period of extended operation be provided regarding the matters described below.

Background:
During the audit breakout session of October 6, 2021, the staff discussed aging management review (AMR) information issues for reactor vessel internal (RVI) components with the applicant. The AMR issues are documented and summarized in Duke Energy document entitled "TRP 15, PWR Vessel Internals, Oconee SLRA, Response to NRC Breakout Questions 15 – 31 and Follow-up from 10-1-2021 Breakout," as placed on the audit portal for the SLRA.

Issue:
In Duke Energy document entitled "TRP 15, PWR Vessel Internals, Oconee SLRA, Response to NRC Breakout Questions 15 – 31 and Follow-up from 10-1-2021 Breakout," the applicant identifies those AMR item changes that they applicant would be making to the AMR items for RVI components in SLRA Table 3.1.1 or SLRA Table 3.1.2-2.

Request:
Indicate whether Duke Energy will be making any changes to the AMR items for RVI components in SLRA Table 3.1.1 or Table 3.1.2-2 consistent those issue response statements calling for AMR item adjustments in the Duke Energy document entitled "TRP 15, PWR Vessel Internals, Oconee SLRA, Response to NRC Breakout Questions 15 – 31 and Follow-up from 10-1-2021 Breakout." Justify the basis for deciding on whether changes to the AMR items will be made.

Responses to RAI B2.1.7-5:

SLRA Tables 3.1.1 and 3.1.2-2 are updated to include the changes described below and shown in the attached markups which are consistent with adjustments described during the audit breakout session on October 6, 2021.

SLRA Table 3.1.2-2, component "Baffle/Former Bolts and Screws (including dowels, baffle-to former bolts, locking pins, baffle-to-former shoulder screws, locking dowel, baffle-to-baffle bolts, locking rings, and barrel-to-former cap screws)" is revised to only include GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for "No Additional Measures" for the nickel alloy alignment dowels.

SLRA Table 3.1.2-2 plant specific Note 1 is revised as follows: "The only component item that was Expansion for 60 years and was changed to Primary for 80 years due to the increase in the severity of the age-related degradation mechanism is the lower grid rib section for IE." An additional plant specific note is added to discuss the core barrel. "The core barrel cylinder (including vertical and circumferential seam welds) was downgraded to No Additional Measures for Unit 1 and Unit 3 as justified in MRP-227-A Applicant/Licensee Action Item 6. The same downgrade is not applicable for Unit 2 because there were post stress relief weld repairs, which resulted in the top flange circumferential weld HAZ being Primary for SCC/IE and the middle circumferential weld region being Primary for IASCC/IE."

SLRA Table 3.1.2-2, component "Core Barrel Cylinder (including core barrel cylinders, top flange, and bottom flange)", is revised to include GALL-SLR Item IV.B4.R-424 (SRP Item 3.1.1-119) for loss of fracture toughness with an NEI note B. This requires an update to Table 3.1.1 for Item Number 3.1.1-119.

SLRA Table 3.1.2-2, component group "Core Barrel-to-Thermal Shield Bolts," is revised to include an alignment with GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for "No Additional Measures" and a plant specific note for the locking devices.

SLRA Table 3.1.2-2, component "Core Barrel-to-Thermal Shield Bolts", should be revised to align to GALL-SLR Item IV.B4.RP-246c (SRP Item 3.1.1-051b) for cracking due to SCC. Line items for cracking due to fatigue should be removed.

SLRA Table 3.1.2-2, component group "Core Support Shield-to-Core Barrel Bolts," is revised to include an alignment with GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for "No Additional Measures" and a plant specific note for the locking devices.

SLRA Table 3.1.2-2, component "Guide Blocks and Bolts" is updated to remove the alignment to GALL-SLR Item IV.B4.RP-260 (SRP Item 3.1.1-058b).

SLRA Table 3.1.2-2, component "Guide Blocks and Bolts" is updated to remove the alignment to GALL-SLR Item IV.B4.RP-246 (SRP Item 3.1.1-051b).

SLRA Table 3.1.2-2, component "Lower Grid Rib Section" is updated to align with GALLSLR Item IV.B4.R-424 (SRP Item 3.1.1-119) for change in dimension, loss of fracture toughness, and loss of material. NEI note B should be used. An update to Table 3.1.1 for Item Number 3.1.1-119 is provided below.

SLRA Table 3.1.2-2, component group "Lower Internals Assembly-to-Core Barrel Bolts," is revised to include an alignment with GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for "No Additional Measures" and a plant specific note for the locking devices.

SLRA Table 3.1.2-2, component group "Lower Internals Assembly-to-Thermal Shield Bolts," is revised to include an alignment with GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for "No Additional Measures" and a plant specific note for the locking devices.

SLRA Table 3.1.2-2, component "Lower Internals Assembly-to-Thermal Shield Bolts", is revised to include NEI note D for the alignment to GALL-SLR Item IV.B4.RP-260 (SRP Item 3.1.1-058b).

SLRA Table 3.1.2-2 is updated to include an additional line item for each of the stainless steel or nickel alloy "no additional measures" components aligning to Item 3.1.1-087 (GALL-SLR Item IV.B4.RP-24). This will address the aging management for loss of material due to pitting or crevice corrosion for the remaining stainless steel and nickel alloy reactor internals components using the *Water Chemistry* program.

SLRA Table 3.1.2-2, component group “Shell Forging-to-Flow Distributor Bolts,” is revised to include an alignment with GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for “No Additional Measures” and a plant specific note for the locking devices.

SLRA Table 3.1.2-2, component group “Vent Valve Body and Retaining Rings,” is revised to include an alignment with GALL-SLR Item IV.B4.RP-236 (SRP Item 3.1.1-055a) for “No Additional Measures” and a plant specific note for the vent valve bodies.

SLRA Revisions:

SLRA Table 3.1.1 (page 3-76) is revised as follows:

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-119	Stainless steel, nickel alloy, stellite PWR reactor vessel internal components or LRA/ SLRA specified reactor vessel internal component exposed to reactor coolant, neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement or thermal aging embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear	Plant-specific aging management program or AMP XI.M16A, “PWR Vessel Internals,” with adjusted site-specific or component-specific aging management basis for a given component	Yes (SRP-SLR Section 3.1.2.2.9)	~~Not applicable. Loss of fracture toughness, changes in dimensions, loss of preload, and loss of material of stainless steel and nickel alloy PWR reactor vessel internal components exposed to reactor coolant and neutron flux are addressed by items 3.1.1-058a and 3.1.1-058b. The associated NUREG-2191 aging items are not used. See further evaluation in Section 3.1.2.2.9.~~ **Consistent with NUREG-2191.**

SLRA Table 3.1.2-2 (page 3-110) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Baffle Plates and Formers (including baffle plates and former plates)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Change in Dimension	PWR Vessel Internals (B2.1.7)	IV.B4.RP-246b	3.1.1- 058b	D, 2
						IV.B4.RP-247b	3.1.1- 058a	D, 2
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-249	3.1.1- 058a	B
						IV.B4.RP-250	3.1.1- 058b	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				Reduction in Fracture Toughness	TLAA	IV.B4.RP-376	3.1.1- 015	A
Baffle/Former Bolts and Screws (including dowels, baffle-to-former bolts, locking pins, baffle-to-former shoulder screws, locking dowel, baffle-to-baffle bolts, locking rings, and barrel-to-former cap screws)	Core Support	Nickel Alloy	Reactor Coolant and Neutron Flux (External)	~~Cracking~~	~~PWR Vessel Internals (B2.1.7)~~	~~None~~	~~None~~	~~F~~

SLRA Table 3.1.2-2 (page 3-111) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Baffle/Former Bolts and Screws (including dowels, baffle-to-former bolts, locking pins, baffle-to-former shoulder screws, locking dowel, baffle-to-baffle bolts, locking rings, and barrel-to-former cap screws)	Core Support	Nickel Alloy	Reactor Coolant and Neutron Flux (External)	~~Cracking~~	~~Water Chemistry (B2.1.2)~~	~~None~~	~~None~~	~~F~~
				~~Cumulative Fatigue Damage~~	~~TLAA~~	~~IV.B4.R-53~~	~~3.1.1-003~~	~~A~~
				~~Loss of Fracture Toughness~~	~~PWR Vessel Internals (B2.1.7)~~	~~None~~	~~None~~	~~F~~
				~~Loss of Material~~	~~PWR Vessel Internals (B2.1.7)~~	~~None~~	~~None~~	~~F~~
					~~Water Chemistry (B2.1.2)~~	~~IV.B4.RP-24~~	~~3.1.1-087~~	~~A~~
				~~Loss of Preload~~	~~PWR Vessel Internals (B2.1.7)~~	~~None~~	~~None~~	~~F~~
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1-055a**	**A**

SLRA Table 3.1.2-2 (page 3-115) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Core Barrel Cylinder (including core barrel cylinders, top flange, and bottom flange)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cracking	PWR Vessel Internals (B2.1.7)	IV.B4.RP-241a	3.1.1- 051a	D, 3
					Water Chemistry (B2.1.2)	IV.B4.RP-241a	3.1.1- 051a	C
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	~~IV.B4.RP-249~~ **IV.B4.R-424**	~~3.1.1- 058a~~ **3.1.1-119**	~~D, 1~~ **B, 3**
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A

SLRA Table 3.1.2-2 (page 3-116) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Core Barrel-to-Thermal Shield Bolts (including thermal shield-to-core barrel bolts, thermal shield-to-core barrel bolts locking clips, and thermal shield cap screws)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cracking	~~PWR Vessel Internals (B2.1.7)~~	~~IV.B4.RP-246d~~	~~3.1.1- 051b~~	~~B~~
						~~IV.B4.RP-248~~	~~3.1.1- 051a~~	~~B~~
						~~IV.B4.RP-248a~~	~~3.1.1- 051a~~	~~B~~
					Water Chemistry (B2.1.2)	IV.B4.RP-246c	3.1.1- 051b	A
						~~IV.B4.RP-248~~	~~3.1.1- 051a~~	~~A~~
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Material	PWR Vessel Internals (B2.1.7)	IV.B4.RP-246e	3.1.1- 058b	B
						IV.B4.RP-248b	3.1.1- 058a	B
					Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1- 055a**	**B, 4**

SLRA Table 3.1.2-2 (page 3-118) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Core Support Shield-to-Core Barrel Bolts (includes core support shield-to-core barrel bolts and core support shield-to-core barrel bolts locking cups and tie plates)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Material	PWR Vessel Internals (B2.1.7)	IV.B4.RP-248b	3.1.1- 058a	B
					Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1- 055a**	**B, 4**
				Reduction in Fracture Toughness	TLAA	IV.B4.RP-376	3.1.1- 015	A

SLRA Table 3.1.2-2 (page 3-121) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Guide Blocks and Bolts (including guide blocks, guide block bolts, guide block washers, and dowels)	Core Support	Nickel Alloy	Reactor Coolant and Neutron Flux (External)	~~Loss of Fracture Toughness~~	~~PWR Vessel Internals (B2.1.7)~~	~~IV.B4.RP-260~~	~~3.1.1- 058b~~	~~B~~
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
		Stainless Steel	Reactor Coolant and Neutron Flux (External)	~~Cracking~~	~~PWR Vessel Internals (B2.1.7)~~	~~IV.B4.RP-246~~	~~3.1.1- 051b~~	~~B~~
					~~Water Chemistry (B2.1.2)~~	~~IV.B4.RP-246~~	~~3.1.1- 051b~~	~~A~~
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-260	3.1.1- 058b	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
Incore Guide Support Plate	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B
Incore Guide Tube Components (including IMI guide tubes, gussets, guide tube nuts, guide tube washers, and locking clips)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A

SLRA Table 3.1.2-2 (page 3-122) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Incore Guide Tube Components (including IMI guide tubes, gussets, guide tube nuts, guide tube washers, and locking clips)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-259	3.1.1- 058a	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
Incore Guide Tube Spider Castings	Core Support	Cast Austenitic Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-258	3.1.1- 058a	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
Lower Grid and Shell Forgings (including lower grid forging and lower shell forging)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B
Lower Grid Flow Distributor Plate	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B
Lower Grid Rib Section	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Change in Dimension	PWR Vessel Internals (B2.1.7)	~~IV.B4.RP-246b~~ **IV.B4.R-424**	~~3.1.1- 058b~~ **3.1.1-119**	~~D~~ **B**
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A

SLRA Table 3.1.2-2 (page 3-123) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Lower Grid Rib Section	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	~~IV.B4.RP-249~~	~~3.1.1- 058a~~	~~D, 1~~
						IV.B4.R-424	**3.1.1-119**	**B, 1**
						~~IV.B4.RP-259~~	~~3.1.1- 058a~~	~~B~~
				Loss of Material	PWR Vessel Internals (B2.1.7)	~~IV.B4.RP-246b~~	~~3.1.1- 058b~~	~~D~~
						IV.B4.R-424	**3.1.1-119**	**B**
					Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
Lower Grid Rib-to-Shell Forging Screws (including rib-to-shell forging cap screws, and rib- to-shell forging cap screw locking pins)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B
Lower Internals Assembly-to-Core Barrel Bolts (including lower grid assembly-to-core barrel bolts and lower grid assembly- to-core barrel bolts locking cups)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Change in Dimension	PWR Vessel Internals (B2.1.7)	IV.B4.RP-247b	3.1.1- 058a	B
				Cracking	PWR Vessel Internals (B2.1.7)	IV.B4.RP-247	3.1.1- 051a	B
						IV.B4.RP-247a	3.1.1- 051a	B

SLRA Table 3.1.2-2 (page 3-124) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Lower Internals Assembly-to-Core Barrel Bolts (including lower grid assembly-to-core barrel bolts and lower grid assembly-to-core barrel bolts locking cups)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cracking	Water Chemistry (B2.1.2)	IV.B4.RP-247	3.1.1- 051a	A
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-260	3.1.1- 058b	D
				Loss of Material	PWR Vessel Internals (B2.1.7)	IV.B4.RP-247b	3.1.1- 058a	B
						IV.B4.RP-247c	3.1.1- 058a	B
					Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				Loss of Preload	PWR Vessel Internals (B2.1.7)	IV.B4.RP-247c	3.1.1- 058a	B
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1- 055a**	**B, 4**

SLRA Table 3.1.2-2 (page 3-126) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Lower Internals Assembly-to- Thermal Shield Bolts (includes lower grid assembly-to-thermal shield studs/nuts and lower grid assembly- to-thermal shield bolts/studs/nuts/ locking cups and tie plates)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cracking	PWR Vessel Internals (B2.1.7)	IV.B4.RP-246	3.1.1- 051b	B
						IV.B4.RP-246a	3.1.1- 051b	B
					Water Chemistry (B2.1.2)	IV.B4.RP-246	3.1.1- 051b	A
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-260	3.1.1- 058b	~~B~~ D
				Loss of Material	PWR Vessel Internals (B2.1.7)	IV.B4.RP-246b	3.1.1- 058b	B
					Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B, 4
No Additional Measures Components	Core Support	Nickel Alloy	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-24	3.1.1- 087	A, 6
		Stainless Steel		None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-24	3.1.1- 087	A, 6

SLRA Table 3.1.2-2 (page 3-128) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Plenum Rib Pads	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Loss of Preload	PWR Vessel Internals (B2.1.7)	IV.B4.RP-251a	3.1.1- 058a	B
Reinforcing Plates	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B
Rib-to-Ring Screws (includes rib-to-ring cap screws and locking pins)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	None	PWR Vessel Internals (B2.1.7)	IV.B4.RP-236	3.1.1- 055a	B
Shell Forging-to-Flow Distributor Bolts (includes flow distributor-to-shell forging bolts and flow distributor-to-shell forging bolts locking clip)	Core Support	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Change in Dimension	PWR Vessel Internals (B2.1.7)	IV.B4.RP-256b	3.1.1- 058a	B
				Cracking	PWR Vessel Internals (B2.1.7)	IV.B4.RP-256	3.1.1- 051a	B
						IV.B4.RP-256a	3.1.1- 051a	B
					Water Chemistry (B2.1.2)	IV.B4.RP-256	3.1.1- 051a	A
				Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Material	PWR Vessel Internals (B2.1.7)	IV.B4.RP-256b	3.1.1- 058a	B
					Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1- 055a**	**B, 4**

SLRA Table 3.1.2-2 (page 3-132) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Vent Valve Body and Retaining Rings (includes vent valve guide blocks, vent valve body, vent valve retaining rings, and vent valve jack	Core Support	Cast Austenitic Stainless Steel	Reactor Coolant and Neutron Flux (External)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-252	3.1.1- 058a	B
						IV.B4.RP-252a	3.1.1- 058b	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1- 055a**	**B, 5**
		Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-252	3.1.1- 058a	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
	Flow Restriction	Cast Austenitic Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A
				Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-252	3.1.1- 058a	B
						IV.B4.RP-252a	3.1.1- 058b	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A
				None	**PWR Vessel Internals (B2.1.7)**	**IV.B4.RP-236**	**3.1.1- 055a**	**B, 5**
		Stainless Steel	Reactor Coolant and Neutron Flux (External)	Cumulative Fatigue Damage	TLAA	IV.B4.R-53	3.1.1- 003	A

SLRA Table 3.1.2-2 (page 3-133) is revised as follows:

Table 3.1.2-2 Reactor Vessel, Reactor Internals, and Reactor Coolant System - Reactor Vessel Internals - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Vent Valve Body and Retaining Rings (includes vent valve guide blocks, vent valve body, vent valve retaining rings, and vent valve jack	Flow Restriction	Stainless Steel	Reactor Coolant and Neutron Flux (External)	Loss of Fracture Toughness	PWR Vessel Internals (B2.1.7)	IV.B4.RP-252	3.1.1- 058a	B
				Loss of Material	Water Chemistry (B2.1.2)	IV.B4.RP-24	3.1.1- 087	A

Plant Specific Notes:

1. The only component item~~s~~ that was ~~were~~ Expansion for 60 years and was ~~were~~ changed to Primary for 80 years due to the increase in the severity of the age-related degradation mechanism ~~are the core barrel cylinder (including the center circumferential weld) for IE and~~ is the lower grid rib section for IE.
2. The aging effect of loss of material due to wear is not applicable to this component.
3. **The core barrel cylinder (including vertical and circumferential seam welds) was downgraded to No Additional Measures for Unit 1 and 3 as justified in MRP-227-A Applicant/License Action Item 6. The same downgrade is not applicable for Unit 2 because there were post stress relief weld repairs, which resulted in the top flange circumferential weld HAZ being Primary for SCC/IE and the middle circumferential weld region being Primary for IASCC/IE.**
4. **The alignment of No Additional Measures applies only to the locking devices associated with the component type.**
5. **The alignment of No Additional Measures applies only to the Vent Valve bodies.**
6. **This line addresses all No Additional Measures components in Table 3.1.2-2, Reactor Vessel, Reactor Internals, and Reactor Coolant System – Reactor Vessel Internals – Aging Management Evaluation, for loss of material due to pitting and crevice corrosion.**

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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ATTACHMENT 45
RAI B2.1.33-1

Enclosure 1, Attachment 45

RAI B2.1.33-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section B2.1.33, "Structures Monitoring," states that the Oconee Structures Monitoring AMP, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALL-SLR Report AMP XI.S6, "Structures Monitoring." For the "scope of program" program element, the GALL-SLR Report AMP states that the program should include all SCs, component supports, and structural commodities in the scope of license renewal that are not covered by other structural aging management programs. Additionally, for the "parameters monitored or inspected" program element, the GALL-SLR Report states that steel bracing and edge supports associated with masonry walls should be inspected for deflection or distortion, loose bolts, and loss of material due to corrosion.

Issue:
SLRA Section 2.4.8 lists several commodities and components that were identified as within the scope of Structures Monitoring Program. However, based on the review of the audited documents and the SLRA enhancements, it is unclear why some of these commodities or components (e.g., piping supports, fuel transfer tube, fire barriers – penetration seals, penetration sleeves, louvers, line supports, aluminum components, liners, sump, drains/curbs, piles, unit vent, lead shield supports, etc.) were not included in the scope of the Structures Monitoring Program to demonstrate that the effects of aging will be adequately managed during the subsequent period of extended operation. The staff noted it was not clear how the procedure identified these components within the scope of the program and their inspection criteria.

Furthermore, SLRA enhancement no. 2 adds the inspection of supports and bracings components associated with masonry walls to the scope of the Structures Monitoring Program. However, the existing program does not seem to include the parameters to be monitored or inspected for these components (e.g., deflection or distortion). Therefore, it is not clear how the Structures Monitoring Program will be consistent the GALL-SLR Report for the parameters to be monitored or inspected for the steel bracing and edge supports associated with masonry walls.

Request:

1. For the commodities and components identified as being managed by the Structures Monitoring Program in SLRA Section 2.4.8, clarify why some of these commodities and components (as listed, in part, above) were not included/added to the scope of the program to demonstrate that they will be adequate managed. Update the SLRA as necessary.

2. Clarify how the Structures Monitoring Program will be consistent with the parameters to be monitored or inspected for the steel bracing and edge supports components associated with masonry walls.

Response to RAI B2.1.33-1:

Request 1:

The Oconee Structures Monitoring program procedure identifies a list of the types of components that are inspected as part of the Structures Monitoring program. All of the components identified in Section 2.4.8 are not identified specifically in the current station procedure but are considered to be bound by the categories identified in the Structures Monitoring procedure. For example, structural steel and equipment supports are identified in the station procedures. These component types include piping supports, line supports, liners, unit vents and piles. Concrete elements include trenches, sumps, and drain/curbs. While the Structures Monitoring procedure is considered inclusive of the components discussed in SLRA Section 2.4.8, a revision to Enhancement 2 is made to ensure the list of components identified in SLRA Section 2.4.8 are explicitly included in the scope of the Structures Monitoring program, except as noted below.

Supplement 2 dated November 11, 2021 (ADAMS Accession No. ML21315A012), Attachment 12, clarified that fire barriers – penetration seals are not managed by the *Structures Monitoring* program. Supplement 2 revised Table 3.2.1-13, Containments, Structures, and Component Supports – Miscellaneous Structural Commodities – Aging Management Evaluation, to show proper alignment of the component Fire Barriers – penetrations seals to the *Fire Protection* (B2.1.30) program. The *ASME XI, Subsection IWF* (B2.1.30) program also performs inspections in Class 1, 2, and 3 piping supports.

SLRA Section 2.4.2, System Evaluation Boundaries states that “The fuel transfer tubes are evaluated within the spent fuel cooling system in Section 2.3.3.” and are identified in Table 3.3.2-17 Auxiliary Systems – Spent Fuel Cooling System – Aging Management Evaluation on page 3-362 of the SLRA. The stainless steel fuel transfer tube is managed by the One-Time Inspection (B2.1.20) program.

Request 2:

For masonry walls, the site implementation of the *Structures Monitoring* program monitors parameters that include separation from supports that could be due to deflection or distortion, which is addressed separately under the ONS Masonry Wall AMP. Steel structures, which include structural steel bracing and edge supports associated with masonry walls, are inspected for twisting, loose bolts, and loss of material due to corrosion as part of the ONS Structures Monitoring AMP. Deflection and distortion are not specifically identified in the current station procedures although implied by the parameters identified above. An enhancement is added to the *Structures Monitoring* program to include structural steel bracing and edge supports associated with masonry walls are inspected for deflection or distortion, loose bolts, and loss of material due to corrosion.

SLRA Revisions:

Note that the SLRA Revisions for RAIs B2.1.33-1 (Enclosure 1, Attachment 45), B2.1.33-2 (Enclosure 1, Attachment 46), B2.1.33-3 (Enclosure 1, Attachment 47), B2.1.33-4 (Enclosure 1, Attachment 48), and B2.1.33-5 (Enclosure 1, Attachment 49) are provided in the SLRA Revisions below with the exception of changes resulting to Appendix A2.32 and B2.1.32 in RAI B2.1.33-4 (Enclosure 1, Attachment 48). The SLRA Revisions for Appendix A2.32 and B2.1.32 are provided in the RAI B2.1.33-4 (Enclosure 1, Attachment 48) response.

SLRA Table 3.5.2-1 (page 3-1358 as revised by Supplement 3 (ML21349A005)) is revised as follows:

Table 3.5.2-1 Containments, Structures, and Component Supports - Auxiliary Building - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Fiber Reinforced Polymer	SS	Fiber Reinforced Polymer	Air – Outdoor (External)	Hardening or Loss of Strength, Loss of Material, Cracking or Blistering	Structures Monitoring (B2.1.33)	None	None	J
				Loss of Adhesion	**Structures Monitoring (B2.1.33)**	**None**	**None**	**J**

SLRA Revisions:

SLRA Appendix A2.33 (pages A-34, A-35, and A-38 as revised by Supplement 3 (ML21349A005)) is revised as follows:

A2.33 Structures Monitoring

Program Description

The *Structures Monitoring* AMP is an existing condition monitoring program that consists of periodic visual inspection and monitoring of the condition of concrete and steel structures, structural components, component supports, and structural commodities to ensure that aging degradation (such as those described in ACI 349.3R, ACI 201.1R, SEI/ASCE 11, and other documents) will be detected, the extent of degradation determined and evaluated, and corrective actions taken prior to loss of intended functions. Structures are monitored on an interval of a nominal five years.

Inspections also include seismic joint fillers, elastomeric materials; and reinforcement of masonry walls, and periodic evaluation of groundwater chemistry and opportunistic inspections for the condition of below grade concrete. **Some concrete walls are covered by metal siding and are considered inaccessible. Inspections of the accessible concrete structures are used as leading indicators for the condition of the concrete under the metal siding. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).**

Additionally, a fiber reinforced polymer has been applied to portions of the masonry walls. The inspection of the fiber reinforced polymer is every third outage (six years) per unit. The inspection includes visual inspection of the fiber reinforced polymer, visual inspection of mortar joints along the bottom edge of the fiber reinforced polymer strengthened masonry walls, and adhesion pull-off testing of test panels.

Enhancements

2. Procedures will be revised to specify that structural components inspected include **piping supports, line supports,** structural bolting, anchor bolts and embedments, supports and bracings associated with masonry walls, pipe whip restraints and jet impingement shields, transmission towers, panels and other enclosures, racks, sliding surfaces, sump and pool liners, wear plates, electrical cable trays and conduits, tube tracks, electrical duct banks, manholes, **louvers, aluminum components, liners, sump, drains/curbs, piles, unit vent, lead shield supports,** doors, penetration seals, and other elastomeric materials.

20. Remove statements in the current station procedures that allow for potential to extend inspections to a 10-year frequency based on adequate justification.

21. Revise station Structures Monitoring procedures to explicitly state that more frequent inspections may be needed based on evaluation of observed degradation.

22. Revise program procedures to specify that evaluation of inspection results includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such

inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding).

23. **Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).**

24. **Enhance procedures to include structural steel bracing and edge supports associated with masonry walls are inspected for deflection or distortion, loose bolts, and loss of material due to corrosion.**

SLRA Table A6.0-1 (pages A-98 and A-104 previously revised in Supplement 3 (ML21349A005)) is revised as follows

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
33	*Structures Monitoring* program	2. Procedures will be revised to specify that structural components inspected include **piping supports, line supports,** structural bolting, anchor bolts and embedments, supports and bracings associated with masonry walls, pipe whip restraints and jet impingement shields, transmission towers, panels and other enclosures, racks, sliding surfaces, sump and pool liners, wear plates, electrical cable trays and conduits, tube tracks, electrical duct banks, manholes, **louvers, aluminum components, liners, sump, drains/curbs, piles, unit vent, lead shield supports,** doors, penetration seals, and other elastomeric materials. **20. Remove statements in the current station procedures that allow for potential to extend inspections to a 10-year frequency based on adequate justification.** **21. Revise station Structures Monitoring procedures to explicitly state that more frequent inspections may be needed based on evaluation of observed degradation.** **22. Revise program procedures to specify that evaluation of inspection results includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding).** **23. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).** **24. Enhance procedures to include structural steel bracing and edge supports associated with masonry walls are inspected for deflection or distortion, loose bolts, and loss of material due to corrosion.**	B2.1.33	Program enhancements for SLR will be implemented no later than 6 months prior to the SPEO.

SLRA Appendix B2.1.33 (page B-222 as revised in Supplement 3 (ML21349A005)) is revised as follows:

B2.1.33 STRUCTURES MONITORING

Program Description

The *Structures Monitoring* AMP is an existing condition monitoring program that consists of periodic visual inspection and monitoring the condition of concrete and steel structures, structural components, component supports, and structural commodities to ensure that aging degradation (such as those described in ACI 349.3R, ACI 201.1R, SEI/ASCE 11, and other documents) will be detected, the extent of degradation determined and evaluated, and corrective actions taken prior to loss of intended functions. Quantitative results (measurements) and qualitative information from periodic inspections are trended with sufficient detail, such as photographs and surveys for the type, severity, extent, and progression of degradation, to ensure that corrective actions can be taken prior to a loss of intended function. The acceptance criteria are derived from applicable consensus codes and standards. For concrete structures, the program includes personnel qualifications and quantitative evaluation criteria of ACI 349.3R. Structures are monitored on an interval of a nominal five years. There are provisions for more frequent inspections when conditions are observed that have a potential for impacting an intended function. Unacceptable conditions, when found, are evaluated or corrected in accordance with the corrective action program. The monitoring methods are effective in detecting the applicable aging effects and the frequency of monitoring is adequate to prevent significant age related degradation to ensure there is no loss of intended function.

The *Structures Monitoring* AMP was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93-01, *"Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"* and Regulatory Guide 1.160, *"Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."* The program includes elements of the *Masonry Walls* (B2.1.32) program and *Inspection of Water-Control Structures Associated with Nuclear Power Plants* (B2.1.34) program.

Concrete structures are inspected for indications of deterioration and distress including evidence of leaching, loss of material, cracking, and a loss of bond, as defined in ACI 201.1R. Inspections of the accessible concrete structures are used as leading indicators for the condition of the concrete under the metal siding. Steel and stainless steel components are inspected for cracking due to stress corrosion cracking and loss of material due to corrosion. Wear plates are additionally inspected for loss of material due to wear. Inspections also include seismic joint fillers, elastomeric materials; and fiber reinforced polymers and steel bracings associated with masonry walls. **The inspection of the fiber reinforced polymer is every third outage (six years) per unit. The inspection includes visual inspection of the fiber reinforced polymer, visual inspection of mortar joints along the bottom edge of the fiber reinforced polymer strengthened masonry walls, and adhesion pull-off testing of test panels.**

The program also includes provisions for periodic testing and assessment of groundwater chemistry and opportunistic inspections of accessible below grade concrete structures. **Some concrete walls are covered by metal siding and are considered inaccessible. Inspections of the accessible concrete structures are used as leading indicators for the condition of the concrete under the metal siding. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).**

Applicable components within the scope of this program include, but are not limited to: bolting, concrete anchors and embedments, concrete components, decking and siding, doors and door seals, ductbanks, external surfaces of bus enclosures (metallic ducting) and bus enclosure structural supports, expansion and seismic joints, foundations, hatches, hazard barriers, metal components such as louvers, miscellaneous steel, penetrations seals and sleeves, piles, pipe whip restraints and jet impingement shields, shielding, steel components, wear plates, steel liners, supports, panels, racks, cabinets, enclosures, cable trays, conduits, wire way gutters, and tubing track.

SLRA Appendix B2.1.33 (page B-223 and B-226 as revised in Supplement 3 (ML21349A005)) is revised as follows:

NUREG-2191 Consistency

The *Structures Monitoring* AMP is an enhanced program that will be consistent with the recommendations provided in Section XI.S6, *Structures Monitoring* program of NUREG-2191 with the enhancements **and exceptions as** described ~~in~~ below.

Enhancements

2. Procedures will be revised to specify that structural components inspected include **piping supports, line supports,** structural bolting, anchor bolts and embedments, supports and bracings associated with masonry walls, pipe whip restraints and jet impingement shields, transmission towers, panels and other enclosures, racks, sliding surfaces, sump and pool liners, wear plates, electrical cable trays and conduits, tube tracks, electrical duct banks, manholes, **louvers, aluminum components, liners, sump, drains/curbs, piles, unit vent, lead shield supports,** doors, penetration seals, and other elastomeric materials. (Element 1)

20. Remove statements in the current station procedures that allow for potential to extend inspections to a 10-year frequency based on adequate justification. (Element 4)

21. Revise station Structures Monitoring procedures to explicitly state that more frequent inspections may be needed based on evaluation of observed degradation. (Element 4)

22. Revise program procedures to specify that evaluation of inspection results includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding). (Element 4)

23. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc). (Elements 3 and 4)

24. Enhance procedures to include structural steel bracing and edge supports associated with masonry walls are inspected for deflection or distortion, loose bolts, and loss of material due to corrosion. (Element 3)

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 46
RAI B2.1.33-2

Enclosure 1, Attachment 46

RAI B2.1.33-2:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section B2.1.33, “Structures Monitoring,” states that the Oconee Structures Monitoring AMP, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALL-SLR Report AMP XI.S6, “Structures Monitoring.” For the “detection of aging effects” program element, the GALL-SLR Report AMP states that the program includes a provision for more frequent inspections based on an evaluation of the observed degradations, and also states that, in general, that all structures be monitored on an interval not to exceed 5 years. SLRA Supplement No. 3, dated December 15, 2021 (ADAMS Accession No. ML21349A005), revised the program’s description in SLRA Sections A2.33 and B2.1.33 to eliminate the previous discussion related to increasing the inspection frequency from 5 years to 10 years. The SLRA Supplement further stated that the program aligns with the GALL-SLR Report for a 5 year inspection frequency.

Issue:
During the audit, the staff reviewed the documents and procedure associated with the Structures Monitoring program. During the review of Section 5.1.2 of Procedure No. AD-EG-ONS-1214, it was noted that the current procedure includes a provision that allows for an increased inspection frequency of up to a 10 year interval. Therefore, it is not clear how Oconee’s Structures Monitoring AMP will still be consistent with the “detection of aging effects” program element of the GALL-SLR Report AMP XI.S6 if the existing program includes a provision that allows for an increased inspection frequency of up to 10 years.

During the review of Section 5.1.2 of Procedure No. AD-EG-ONS-1214, it was also noted that the procedure requires more frequent inspection when an unusual event, such as flooding or seismic activity, occurs. However, no similar provision was identified for when an evaluation of the observed degradations warrants an increased inspection frequency. Therefore, it is unclear if Section 5.1 is consistent with the GALL-SLR Report to ensure that provision exist for more frequent inspections based on an evaluation of the observed degradations.

Request:
1. Clarify how the Structures Monitoring Program will be consistent with the GALL-SLR Report, as claimed in the SLRA, for ensuring that all structures and components be monitored on an interval not to exceed 5 years.

2. Clarify how the Structures Monitoring Program will be consistent with the GALL-SLR Report, as claimed in the SLRA, for ensuring that provision exist for more frequent inspections based on an evaluation of the observed degradations.

3. Update the SLRA as necessary consistent with the responses to Requests 1 and 2 above.

Response to RAI B2.1.33-2:

Request 1:

As identified in Supplement 3, ONS intends to comply with SLR-GALL for the *Structures Monitoring* program by ensuring that all structures and components are monitored on an interval not to exceed 5 years. An enhancement is added to the *Structures Monitoring* program (B2.1.33) to explicitly remove statements in the current station procedures that allow for potential to extend inspections to a 10-year frequency based on adequate justification for components in the scope of SLR.

Request 2:

An enhancement is added to the *Structures Monitoring* program (B2.1.33) procedure to explicitly state that more frequent inspections may be needed based on evaluation of observed degradation.

Request 3:

The SLRA has been revised through the preparation of a supplement to identify these two new enhancements to the *Structures Monitoring* program.

SLRA Revisions:

SLRA Appendices A2.33, B2.1.33, and Table A6.0-1 are revised to state the following:

20. Remove statements in the current station procedures that allow for potential to extend inspections to a 10-year frequency based on adequate justification.
21. Revise station Structures Monitoring procedures to explicitly state that more frequent inspections may be needed based on evaluation of observed degradation.

The revisions to these SLRA sections are provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 47
RAI B2.1.33-3

Enclosure 1, Attachment 47

RAI B2.1.33-3:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Section B2.1.33, “Structures Monitoring,” states that the Oconee Structures Monitoring AMP, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALL-SLR Report AMP XI.S6, “Structures Monitoring.” For the “detection of aging effects” program element, the GALL-SLR Report AMP states that, for sites with nonaggressive groundwater/soil (pH > 5.5, chlorides < 500 ppm, and sulfates < 1,500 ppm), the program needs to (a) evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas, and (b) examine representative samples of the exposed portions of the below grade concrete, when excavated for any reason.

Issue:
During the review of Procedure No. AD-EG-ONS-1214, it was noted that, in Section 5.2.3, the current plant procedure includes a provision to ensure that visual inspections are performed in normally inaccessible areas when areas are made accessible by excavation or by other means (effectively addressing item (b) from the GALL-SLR Report for underground structures). However, it is not clear what provision exists in the current procedure to ensure that inaccessible areas are evaluated for acceptability when conditions exist (has been identified) in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (i.e., to effectively address item (a) from the GALL-SLR Report). Therefore, it is not clear how the Structures Monitoring Program will be consistent with the GALL-SLR Report for adequately managing the aging effects of inaccessible structural elements when exposed to a nonaggressive groundwater/soil environment. It is noted that a similar criterion might be applicable to other inaccessible structural elements (e.g., structural components covered by metal siding) located above ground.

Request:
Clarify how the Structures Monitoring Program is consistent with the GALL-SLR Report for evaluating the acceptability of inaccessible areas (including inaccessible areas aboveground) when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. Update the SLRA as necessary.

Response to RAI B2.1.33-3:

An enhancement shall be added to the *Structures Monitoring* program to revise program procedures to specify that evaluation of inspection results includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. (e.g. exposed to groundwater/soil environment) and above grade (e.g. structural components covered by metal siding).

A supplement is needed to add the enhancement/commitment identified above.

SLRA Revisions:

SLRA Appendices A2.33, B2.1.33, and Table A6.0-1 revisions are shown in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45. The SLRA sections are being revised to include:

22. Revise program procedures to specify that evaluation of inspection results includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding).

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 48
RAI B2.1.33-4

Enclosure 1, Attachment 48

RAI B2.1.33-4:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SLRA Sections B2.1.32, "Masonry Walls," and B2.1.33, "Structures Monitoring," state that the Oconee programs, with the enhancements provided in the SLRA, will be consistent with the ten program elements of GALL-SLR Report AMPs XI.S5, "Masonry Walls," and XI.S6, "Structures Monitoring." The SRPSLR states that as part of the development of the SLRA, the applicant should assess the AMPs in the GALL-SLR Report. It is incumbent on the applicant to ensure that the conditions and operating experience (OE) at the plant are bounded by the conditions and OE for which the GALL-SLR Report program was evaluated. If these bounding conditions are not met, the applicant should address the additional effects of aging and augment the AMP(s) in the GALL-SLR Report in the SLRA, as appropriate.

During the on-site audit, the staff noted that most of the structures above ground were covered with metal siding on the exterior, which makes the exterior of the structural components inaccessible for inspection. This plant-specific condition is not generically bounded by the general conditions for which the GALL-SLR Report programs XI.S5 and XI.S6 were evaluated. To address this plant-specific condition, the applicant revised SLRA Sections B2.1.32 and B2.1.33 in Attachment 15 of SLRA Supplement 3, dated December 15, 2021 (ADAMS Accession No. ML21349A005), to describe the actions that will be taken to manage the structural components that are inaccessible for inspection by the metal siding.

Issue:
Although the changes provided in SLRA Supplement 3 describes, in part, how the programs will manage the inaccessible areas covered by metal siding, additional clarification is necessary for the following:

1. It is not clear how plant procedures ensure that engineering inspections are performed on structural elements covered by metal siding when portions are exposed for any reason (e.g., maintenance activities, modification, etc.). It is noted that Section 5.2 of procedure no. ADEGONS1214 provides a similar provision for inaccessible areas requiring an excavation, but it is not clear if a similar provision exist for inaccessible areas covered by metal siding.

2. SLRA Section A2.33, "Structures Monitoring," describes the "opportunistic inspections for the condition of below grade concrete." SLRA Section A2.32, "Masonry Walls," describes the general inspections associated with the monitoring of masonry walls. However, it is not clear why the programs' descriptions do not describe the program actions associated this plant specific condition (i.e., the type of evaluation and/or opportunistic inspections that will be used to monitor the condition for inaccessible areas covered by metal siding).

3. The SLRA, as revised by SLRA Supplement 3, does not (a) state if plant specific operating experience from the inspection of accessible concrete structures considered to be leading indicators has resulted in indication of the presence of, or result in, degradation to such inaccessible areas, and (b) provide a justification to demonstrate that the proposed monitoring actions will be adequate to manage the aging effects for inaccessible areas covered by metal siding so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Request:

1. Clarify how the existing Structures Monitoring Program ensures that engineering inspections are performed on structural elements covered by metal siding when portions are exposed for any reason (e.g., maintenance activities, modification, etc.). Update the SLRA as necessary.

2. Clarify why the program descriptions in SLRA Sections A2.33 and A2.32 do not describe the program actions associated with the monitoring of inaccessible areas covered by metal siding. Update the SLRA as necessary.

3. State if plant-specific operating experience from the inspection of accessible concrete areas considered to be leading indicators of the above grade inaccessible structural elements has resulted in indication of the presence of, or result in, degradation to such inaccessible areas. Also, provide a justification to demonstrate that the proposed monitoring actions will be adequate to manage the aging effects for inaccessible areas covered by metal siding.

Response to RAI B2.1.33-4:

Request 1:

An enhancement is added to the *Structures Monitoring* program procedures to that opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc). This enhancement is applicable to the *Masonry Walls* (B2.1.32) program and the *Structures Monitoring* (B2.1.33) program.

Request 2:

A revision to SLRA Section A3.32 *Masonry Walls* program and A2.33 *Structures Monitoring* program to include descriptions of the programmatic actions associated with monitoring of inaccessible areas covered by metal siding. SLRA Sections B2.1.32 *Masonry Walls* program and B2.1.33 *Structures Monitoring* program will also be revised to discuss monitoring of inaccessible areas covered by metal siding.

Request 3:

An OE search was performed to identify if OE is present in which the accessible concrete areas have identified degradation which results in degradation to inaccessible areas. The OE search revealed that there have been no examples of inspections of accessible concrete areas that are considered leading indicators of the above grade inaccessible structural elements that have resulted in the indication of the presence of, or resulted in, degradation to inaccessible areas.

SLRA Revisions:

SLRA Appendix A2.32 (page A-34 previously revised in Supplement 3 (ML21349A005)) is revised as follows:

A2.32 Masonry Walls

Program Description

The *Masonry Walls AMP* is an existing condition monitoring program that is implemented as part of the *Structures Monitoring* (A2.33) AMP and manages cracking, loss of material, and loss of material (spalling and scaling) that could impact the intended function of the masonry walls. The *Masonry Walls* AMP consists of inspections, consistent with Inspection and Enforcement Bulletin 80-11, and plant-specific monitoring, proposed by Information Notice 87-67, for managing shrinkage, separation, gaps, loss of material and cracking of masonry walls such that the evaluation basis is not invalidated and intended functions are maintained.

Some masonry walls are covered by metal siding and are considered inaccessible. Visual inspections will be performed on a representative sample of masonry walls that are covered by metal siding to look for loss of material, cracking, or other signs of degradation. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).

SLRA Appendix A2.33 (page A-34 and A-35 previously revised in Supplement 3 (ML21349A005)) is revised to add the below to the program description. This change is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

A2.33 Structures Monitoring

Program Description

Some concrete walls are covered by metal siding and are considered inaccessible. Inspections of the accessible concrete structures are used as leading indicators for the condition of the concrete under the metal siding. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).

SLRA Appendix A2.33 (page A-38 previously revised in Supplement 3 (ML21349A005)) and SLRA Table A6.0-1 (page A-104 previously revised in Supplement 3 (ML21349A005)) is revised to add enhancement 23. This change is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

Enhancements

23. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
33	*Structures Monitoring* program	**23. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).**	B2.1.33	Program enhancements for SLR will be implemented no later than 6 months prior to the SPEO.

SLRA Appendix B2.1.32 (page B-218 as revised in Supplement 3 (ML21349A005)) is revised as follows:

B2.1.32 MASONRY WALLS

Program Description

The *Masonry Walls* AMP is an existing program implemented as part of the current *Structures Monitoring* (B2.1.33) AMP. It is based on the guidance provided in IE Bulletin 80-11, *"Masonry Wall Design"*, and NRC Information Notice 87-67, *"Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11"*, and is implemented through station procedures.

The *Masonry Walls* AMP manages inspections of masonry walls for cracks in joints, unsealed penetrations, missing or broken blocks, or separation from supports that could impact the intended function or potentially invalidate its evaluation basis. The program relies on periodic visual inspections, conducted at a nominal five year frequency to monitor and maintain the condition of masonry walls within the scope of SLR so that the established evaluation basis for each masonry wall remains valid during the SPEO. Observed aging effects that could impact masonry wall intended function or potentially invalidate its evaluation basis are entered into the corrective action process for further analysis, repair, or replacement. Masonry walls that are considered fire barriers are also managed by the *Fire Protection* (B2.1.15) program.

Some masonry walls are covered by metal siding and are considered inaccessible. Visual inspections will be performed on a representative sample of masonry walls that are covered by metal siding to look for loss of material, cracking, or other signs of degradation. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).

Aging effects of masonry wall structural steel support elements that provide technical basis for boundary conditions used in seismic analysis are managed by the *Structures Monitoring* (B2.1.33) AMP.

SLRA Appendix B2.1.33 (page B-222 as revised in Supplement 3 (ML21349A005)) is revised to add the below to the program description. This change is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

B2.1.33 STRUCTURES MONITORING

Program Description

The program also includes provisions for periodic testing and assessment of groundwater chemistry and opportunistic inspections of accessible below grade concrete structures. **Some concrete walls are covered by metal siding and are considered inaccessible. Inspections of the accessible concrete structures are used as leading indicators for the condition of the concrete under the metal siding. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc).**

Applicable components within the scope of this program include, but are not limited to: bolting, concrete anchors and embedments, concrete components, decking and siding, doors and door seals, ductbanks, external surfaces of bus enclosures (metallic ducting) and bus enclosure structural supports, expansion and seismic joints, foundations, hatches, hazard barriers, metal components such as louvers, miscellaneous steel, penetrations seals and sleeves, piles, pipe whip restraints and jet impingement shields, shielding, steel components, wear plates, steel liners, supports, panels, racks, cabinets, enclosures, cable trays, conduits, wire way gutters, and tubing track.

SLRA Appendix B2.1.33 (page B-226 as revised in Supplement 3 (ML21349A005)) is revised to add the below to the enhancements. This change is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

Enhancements

23. Opportunistic engineering inspections should be performed on structural elements covered by metal siding when portions are exposed for any reason (e.g. maintenance activities, modification, etc). (Elements 3 and 4)

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 49
RAI B2.1.33-5

Enclosure 1, Attachment 49

RAI B2.1.33-5:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
The GALL-SLR Report contains an AMR evaluation of a large number of structures and components (SCs) that may be in the scope of a typical SLRA and may need to be the subject of an AMR in accordance with requirements in 10 CFR 54.21(a)(1). However, the AMR line items in the SRP-SLR and GALL-SLR Report may not provide a comprehensive list of all structures and components that need to be within the subject of an AMR or a comprehensive list of all potential aging effects that may be applicable to those structures or components as being the subject of an AMR. Therefore, plant-specific AMRs should be performed if additional aging effects or components (not referenced in the SRP-SLR and GALL-SLR Report) are applicable to GALL-SLR Report AMP should be described and justified.

SLRA Section B2.1.33, “Structures Monitoring,” stated that the installation of the fiber reinforced polymer (FRP) associated with masonry walls was allowed through a license amendment request that was approved by the NRC on June 27, 2011 (ADAMS Accession No. ML11164A257). The SLRA also stated that that the Structures Monitoring Program will be used to inspect and manage the aging effects for the FRP component. Since the GALL-SLR Report does not generically address FRP components, SLRA Table 3.5.2-1 performed a plant-specific AMR of this component by considering the aging effects of hardening or loss of strength, loss of material, cracking or blistering. SLRA Section B2.1.33, as revised by SLRA Supplement No. 3 in letter dated December 15, 2021 (ADAMS Accession No. ML21349A005), also provided a new exception to the “detection of aging effects” program element to allow for the inspection and testing of the FRP on a 6 year inspection frequency. In this exception, the SLRA described the environment for which the FRP is exposed and the additional testing that is performed (i.e., adhesion pull-off testing).

Issue:
Additional clarification is necessary for the following:

1. SLRA B2.1.33, as amended, states that the program inspection includes visual inspection of the FRP, visual inspection of mortar joints along the bottom edge of the FRP strengthened masonry walls, and an adhesion pull-off testing of control panels which constitutes a more thorough and robust inspection than the typical visual inspection of the Structures Monitoring program. Since the adhesion pull-off testing is intended to monitor any degradation associated with bonding failure between the masonry wall and the FRP material, it is not clear what AMR line item, in SLRA Table 3.5.2-1, was used to evaluate/credit the associated aging effect/mechanism (e.g., loss of adhesion) that will be monitored or inspected for the FRP.

2. SLRA B2.1.33, generally states that the inspection of the FRP is within the scope of the Structures Monitoring Program. Since the installation of the FRP was allowed through a

license amendment request that only considered the inspections until the end of the (current) license in July 2034, it is not clear if the intent for the Structures Monitoring Program is to manage the FRP, throughout the subsequent period of extended operations, using the same parameters, inspection criteria, and methods described in the approved license amendment request to ensure its intended functions are maintained during SPEO.

3. Considering that SLRA Supplement No. 3 added an exception to the Structures Monitoring Program, it is not clear why the "NUREG 2191 Consistency" Section in SLRA B2.1.33 only states that the Structures Monitoring AMP will be consistent with the enhancements, without crediting the exception.

4. It is not clear why the programs description, for the UFSAR Supplement in Section A2.33, does not describe, as required by 10 CFR 54.21(d), the program's plant specific activities for managing the effects of aging for FRP associated with the masonry walls (i.e., the type of inspection and testing, including the inspection frequency, that will be used to monitor the condition for this component). It is noted that the inspection of FRP is not generically addressed or described in the FSAR Supplement Summaries of the GALLSLR Report for the Structures Monitoring Program.

<u>Request:</u>

1. Clarify what line item in SLRA Table 2 addresses the plant-specific aging effect that is associated with the adhesion pull-off testing (e.g., loss of adhesion).

2. Clarify if the Structures Monitoring Program will manage the degradations associated with the FRP, throughout the subsequent period of extended operations, using the same inspection parameters, inspection criteria, and methods described in the license amendment request that was authorized by the NRC on June 27, 2011, to ensure its intended functions are maintained during SPEO.

3. Clarify why the "NUREG 2191 Consistency" Section in SLRA B2.1.33 did not credit the exception in its statement.

4. Clarify why the program description from the UFSAR Supplement in Section A2.33 does not describe the program's plant specific activities for managing the effects of aging of the FRP associated with the masonry walls (including the type of inspection, testing, and inspection frequency), as required by 10 CFR 54.21(d).

Response to RAI 2.1.33-5:

Request 1:

The SLRA currently does not contain an AMR line item for the plant-specific aging effect that is associated with the adhesion pull-off testing. The SLRA is revised to add an AMR line item for loss of adhesion for the Fiber Reinforced Polymer. Note that this change is also included in Enclosure 1, Attachment 45, page 3-1358.

Request 2:

The Structures Monitoring Program manages the degradations associated with the fiber reinforced polymer (FRP), throughout the subsequent period of extended operations, using the same inspection parameters, inspection criteria, and methods described in the license amendment request that was authorized by the NRC on June 27, 2011, to ensure its intended functions are maintained during SPEO.

Request 3:

SLRA Supplement No. 3 in letter dated December 15, 2021 (ADAMS Accession No. ML21349A005) identified an exception to the frequency of inspection for the fiber reinforced polymer. SLRA Section B2.1.33 Exception to NUREG-2191 statement was revised, but the corresponding NUREG-2191 Consistency statement was not revised. NUREG-2191 Consistency section should state "The *Structures Monitoring* AMP is an enhanced program that will be consistent with the recommendations provided in Section XI.S6, *Structures Monitoring* program of NUREG-2191 with the enhancements and exceptions described below." The SLRA is updated appropriately.

Request 4:

The SLRA UFSAR Supplement in Section A2.33 is revised to include the Fiber Reinforced Polymer plant specific activities for aging management (including the type of inspection, testing, and inspection frequency), as required by 10 CFR 54.21(d).

SLRA Revisions:

SLRA Table 3.5.2-1 (page 3-1358 as revised by Supplement 3 (ML21349A005)) is revised to include loss of adhesion as an aging effect. This change is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

Table 3.5.2-1 Containments, Structures, and Component Supports - Auxiliary Building - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Fiber Reinforced Polymer	SS	Fiber Reinforced Polymer	Air – Outdoor (External)	Hardening or Loss of Strength, Loss of Material, Cracking or Blistering	Structures Monitoring (B2.1.33)	None	None	J
				Loss of Adhesion	Structures Monitoring (B2.1.33)	None	None	J

SLRA Appendix A2.33 (page A-35 as revised by Supplement 3 (ML21349A005)) is revised to add information pertaining to fiber reinforced polymer to the program description. This change is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

A2.33 Structures Monitoring

Program Description

Additionally, a fiber reinforced polymer has been applied to portions of the masonry walls. The inspection of the fiber reinforced polymer is every third outage (six years) per unit. The inspection includes visual inspection of the fiber reinforced polymer, visual inspection of mortar joints along the bottom edge of the fiber reinforced polymer strengthened masonry walls, and adhesion pull-off testing of test panels.

SLRA Appendix B2.1.33 (page B-222 as revised by Supplement 3 (ML21349A005)) is revised as follows and is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

B2.1.33 STRUCTURES MONITORING

Program Description

Concrete structures are inspected for indications of deterioration and distress including evidence of leaching, loss of material, cracking, and a loss of bond, as defined in ACI 201.1R. Inspections of the accessible concrete structures are used as leading indicators for the condition of the concrete under the metal siding. Steel and stainless steel components are inspected for cracking due to stress corrosion cracking and loss of material due to corrosion. Wear plates are additionally inspected for loss of material due to wear. Inspections also include seismic joint fillers, elastomeric materials; and fiber reinforced polymers and steel bracings associated with masonry walls. **The inspection of the fiber reinforced polymer is every third outage (six years) per unit. The inspection includes visual inspection of the fiber reinforced polymer, visual inspection of mortar joints along the bottom edge of the fiber reinforced polymer strengthened masonry walls, and adhesion pull-off testing of test panels.** The program also includes provisions for periodic testing and assessment of groundwater chemistry and opportunistic inspections of accessible below grade concrete structures.

SLRA Appendix B2.1.33 (page B-223) is revised as follows and is provided in the response to RAI B2.1.33-1 in Enclosure 1, Attachment 45.

NUREG-2191 Consistency

The *Structures Monitoring* AMP is an enhanced program that will be consistent with the recommendations provided in Section XI.S6, *Structures Monitoring* program of NUREG-2191 with the enhancements **and exception as** described ~~in~~ below.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
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SET 2

ATTACHMENT 50
RAI 3.5.1-092-1

Enclosure 1, Attachment 50

RAI 3.5.1-092-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
The GALL-SLR Report contains an AMR evaluation of a large number of structures and components (SCs) that may be in the scope of a typical SLRA and may need to be the subject of an AMR in accordance with requirements in 10 CFR 54.21(a)(1). However, the AMR line items in the SRP-SLR and GALL-SLR Report may not provide a comprehensive list of all structures of components that need to be within the subject of an AMR or a comprehensive list of all potential aging effects that may be applicable to those structures or components as being the subject of an AMR. Therefore, plant-specific AMRs should be performed if additional aging effects (not referenced in the SRP-SLR and GALL-SLR Report) are applicable to a specific structure or component subject to an AMR. It is noted that wear plate components are not generically addressed by the GALL-SLR Report.

Issue:
During the review of the AMR items in SLRA Table 3.5.2-22, as revised by Attachment 3 of SLRA Supplement No. 3 dated December 15, 2021 (ADAMS Accession No. ML21349A005), the staff noted two instances where the table includes two separate AMR line items addressing the same material, environment, and aging effect combination for wear plates (each instance has a different material (steel, stainless steel) and environment combination). In the revised SLRA Table 3.5.2-22 (page 3-1449) in Attachment 3 of SLRA Supplement No. 3, for the instance where the steel wear plate is exposed to an “air-indoor uncontrolled (external)” environment, one of the two AMR line items is associated with the GALL-SLR Report AMR line item 3.5.1-092 and references generic note C to indicate that a different component was used.

The other AMR line item for steel wear plate is a plant-specific AMR that references generic note H to indicate that the aging effect is not in the GALL-SLR Report for this component, material, and environment combination. For the instance where the stainless steel wear plate is exposed to an “air (external)” environment, one of the two AMR line items is associated with the GALL-SLR Report AMR line item 3.5.1-100 and references generic note C. The other AMR line item for stainless steel wear plate is a plant-specific AMR that also references generic note H. Therefore, for each of steel and stainless steel wear plate, it is not clear which line item the SLRA intends to use to evaluate the wear plates and demonstrate that the associated aging effects will be adequately managed by the program.

During the review of AMR items in SLRA Table 3.5.2-22, as revised by SLRA Supplement No. 3, it was also noted that cracking is an aging effect that will be managed for stainless steel wear plates by the Structures Monitoring Program. However, SLRA enhancement no. 19 specifies only monitoring for loss of material for wear plates but does not include monitoring for cracking. Therefore, it is not clear how the Structures Monitoring Program will be adequate to manage all the applicable aging effects refenced in SLRA Table 3.5.2-22.

Request:

1. Clarify, with sufficient justification, which of the line items from SLRA Table 3.5.2-22, as revised in Attachment 3 of SLRA Supplement No. 3, will be used to evaluate the wear plates (for each instance describe above) and demonstrate that the associated aging effects of wear plates will be adequately managed during the period of extended operations. Update the SLRA as necessary.

2. Clarify how the Structures Monitoring Program will also be adequate to manage all the applicable aging effect of cracking for wear plates identified in SLRA Table 3.5.2-22, as revised in Attachment 3 of SLRA Supplement No. 3. Update the SLRA as necessary.

Response to RAI 3.5.1-092-1:

Request 1:

The steel wear plate exposed to an "air-indoor uncontrolled" environment contains two AMR line items. The AMR line item associated with SRP-SLR item 3.5.1-092 references generic note D (Note C was changed to Note D for this line item in Supplement 3 (ML21349A005)) to indicate that a different component was used, that is a "wear plate" instead of "support members; welds; bolted connections; support anchorage to building structure", to evaluate for loss of material due to general, pitting corrosion. This AMR line item evaluates loss of material for the steel wear plates for this material, environment, and aging effect combination. The plant specific AMR line item will be deleted as the SRP-SLR item 3.5.1-092 already evaluates the wear plates and demonstrate that the associated aging effects will be adequately managed by the program. The SLRA is amended to correct this discrepancy. Additionally, enhancement 2 was revised in Supplement 3 (ML21349A005) to include wear plates as included in the scope of the *Structures Monitoring* program. The *Structures Monitoring* program currently manages structural steel for loss of material, therefore the associated aging effects will be adequately managed by the program.

The stainless-steel wear plate exposed to an "air" environment contains two AMR line items. The line item associated with SRP-SLR item 3.5.1-100 references generic note D (Note C was changed to Note D for this line item in Supplement 3 (ML21349A005)) to indicate that a different component was used, that is a "wear plate" instead of "support members; welds; bolted connections; support anchorage to building structure", to evaluate for loss of material due to pitting and crevice corrosion and cracking due to SCC. This AMR line item evaluates loss of material, cracking for the stainless-steel wear plates for this material, environment, and aging effect combination. The plant specific AMR line item will be deleted as the SRP-SLR item 3.5.1-100 already evaluates the wear plates and demonstrate that the associated aging effects will be adequately managed by the program. Enhancement 19 is revised to state that stainless-steel components will be monitored for loss of material due to pitting and crevice corrosion and cracking due to SCC ensuring that the associated aging effects will be adequately managed by the program.

Request 2:

As stated in Response #1, the *Structures Monitoring* Program is revised to state that stainless-steel components will be monitored for loss of material due to pitting and crevice corrosion and cracking due to SCC. In addition, Supplement 3 (ML21349A005) revised enhancement 2 to explicitly include wear plates as in scope for the *Structures Monitoring* program. These enhancements ensure that stainless-steel wear plates will be adequately managed for cracking due to SCC by the *Structures Monitoring* program.

SLRA Revisions:

SLRA Table 3.5.2-22 (page 3-1449 as revised by Supplement 3 (ML21349A005)) is revised as follows:

Table 3.5.2-22 Containments, Structures, and Component Supports - Component Supports - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191 Item	NUREG-2192 Table 1	Notes
Wear Plate	SP	Stainless Steel	Air (External)	Loss of Material, Cracking	Structures Monitoring (B2.1.33)	III.B4.T-37b	3.5.1- 100	C
				~~Loss of Material~~	~~Structures Monitoring (B2.1.33)~~	~~None~~	~~None~~	~~H, 6~~
			Air with Borated Water Leakage (External)	None	None	III.B4.TP-4	3.5.1- 098	C, 5
		Steel	Air – Indoor Uncontrolled (External)	Loss of Material	Structures Monitoring (B2.1.33)	III.B4.TP-43	3.5.1- 092	C
						~~None~~	~~None~~	~~H, 6~~

SLRA Table 3.5.2-22 (page 3-1450 as revised by Supplement 3 (ML21349A005)) is revised as follows:

Table 3.5.2-22 Containments, Structures, and Component Supports - Component Supports - Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect	Aging Management Program	NUREG-2191Item	NUREG- 2192 Table 1	Notes

Plant Specific Notes:

1. Steel elements include support members, bearing plates, base plates, connections, instrument racks and structural frames.
2. Aluminum Elements include support members and cable trays.
3. Stainless Steel elements includes support members, bearing plates, base plates, connections, instrument racks and structural frames.
4. Subject component is the fuel transfer tube support.
5. Air with Borated Water Leakage Environment is for components in the Auxiliary Building, Reactor Building and Borated Water Storage Tank Superstructure.
6. ~~Loss of material due to wear will be managed for wear plates.~~

SLRA Appendix A2.33 (page A-38 as revised by Supplement 3 (ML21349A005)) is revised as follows:

Enhancements

19. Expand the program to monitor **stainless steel components** ~~wear plates~~ for indications of significant loss of material ~~due to wear~~ **and cracking through visual inspections**. Establish acceptance criteria for **stainless steel components** ~~wear plates~~ as no significant loss of material **or cracking** ~~due to wear~~ that could result in a loss of intended function, as required by design.

SLRA Table A6.0-1 (page A-104 as revised by Supplement 3 (ML21349A005)) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
33	*Structures Monitoring* program	19. Expand the program to monitor **stainless steel components** ~~wear plates~~ for indications of significant loss of material ~~due to wear~~ **and cracking through visual inspections**. Establish acceptance criteria for **stainless steel components** ~~wear plates~~ as no significant loss of material **or cracking** ~~due to wear~~ that could result in a loss of intended function, as required by design.	B2.1.33	Program enhancements for SLR will be implemented no later than 6 months prior to the SPEO.

SLRA Appendix B2.1.33 (page B-222 as revised by Supplement 3 (ML21349A005)) is revised as follows:

B2.1.33 STRUCTURES MONITORING

Program Description

The *Structures Monitoring* AMP is an existing condition monitoring program that consists of periodic visual inspection and monitoring the condition of concrete and steel structures, structural components, component supports, and structural commodities to ensure that aging degradation (such as those described in ACI 349.3R, ACI 201.1R, SEI/ASCE 11, and other documents) will be detected, the extent of degradation determined and evaluated, and corrective actions taken prior to loss of intended functions. Quantitative results (measurements) and qualitative information from periodic inspections are trended with sufficient detail, such as photographs and surveys for the type, severity, extent, and progression of degradation, to ensure that corrective actions can be taken prior to a loss of intended function. The acceptance criteria are derived from applicable consensus codes and standards. For concrete structures, the program includes personnel qualifications and quantitative evaluation criteria of ACI 349.3R. Structures are monitored on an interval of a nominal five years. There are provisions for more frequent inspections when conditions are observed that have a potential for impacting an intended function. Unacceptable conditions, when found, are evaluated or corrected in accordance with the corrective action program. The monitoring methods are effective in detecting the applicable aging effects and the frequency of monitoring is adequate to prevent significant age related degradation to ensure there is no loss of intended function.

The *Structures Monitoring* AMP was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93-01, *"Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"* and Regulatory Guide 1.160, *"Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."* The program includes elements of the *Masonry Walls* (B2.1.32) program and *Inspection of Water-Control Structures Associated with Nuclear Power Plants* (B2.1.34) program.

Concrete structures are inspected for indications of deterioration and distress including evidence of leaching, loss of material, cracking, and a loss of bond, as defined in ACI 201.1R. Steel and stainless steel components are inspected for cracking due to stress corrosion cracking and loss of material due to corrosion. ~~Wear plates are additionally inspected for loss of material due to wear.~~ Inspections also include seismic joint fillers, elastomeric materials; and fiber reinforced polymers and steel bracings associated with masonry walls. The program also includes provisions for periodic testing and assessment of groundwater chemistry and opportunistic inspections of accessible below grade concrete structures.

SLRA Appendix B.2.1.33 (page B-226 as revised by Supplement 3 (ML21349A005)) is revised as follows:

19. Expand the program to monitor **stainless steel components** ~~wear plates~~ for indications of significant loss of material ~~due to wear~~ **and cracking through visual inspections**. Establish acceptance criteria for **stainless steel components** ~~wear plates~~ as no significant loss of material **or cracking** ~~due to wear~~ that could result in a loss of intended function, as required by design. (Element 3 and 6)

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 51
RAI 3.5.1-093-1

Enclosure 1, Attachment 51

RAI 3.5.1-093-1:

Regulatory Basis:
Title 10 of the *Code of Federal Regulations* (CFR) Section 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:
SRP-SLR Table 3.5-1, item 093, states that the component "Galvanized steel support members; welds; bolted connections; support anchorage to building structure" for applicable structural components that are associated with cable trays, conduit, HVAC ducts, tube track, instrument tubing, non-ASME piping and components, emergency diesel generator supports, HVAC system components, and/or other miscellaneous mechanical equipment should be managed for loss of material due to pitting and crevice corrosion by the AMP XI.S6, "Structures Monitoring," program when the component is exposed to an air outdoor environment. For this material and component combination, the SRP-SLR Table 3.5.1, item 095, states that no aging effects needs to be managed or aging management program required for "galvanized steel support members; welds; bolted connections; support anchorage to building structure" when the component is exposed to an air indoor (uncontrolled) environment.

SLRA Table 3.5.1, AMR item 3.5.1-093, stated that this AMR line item is not applicable/used because galvanized steel components were evaluated using the GALL/SRP-SLR AMR line items associated with steel. Similarly, SLRA Table 3.5.1, AMR item 3.5.1-095, stated that this AMR line item is not applicable/used because galvanized steel components were evaluated using the GALL/SRP-SLR AMR line items associated with steel.

Issue:
After reviewing the SLRA AMR line items associated with steel, it is not clear how the galvanized steel supports, associated with the SRP-SLR Table 3.5.1, items 093 and 095 (for the structural components described above), were evaluated for ONS. The SLRA Table 3.5.1 does not address which alternate SRP-SLR line item is used for galvanized support members and/or which galvanized supports do not require aging management based on environment. Additional clarification is necessary to understand what galvanized steel supports requires evaluation and will be managed for the aging effects by the Structures Monitoring Program, and what galvanized steel supports will not require a program or aging effect to be managed due to the environment.

Request:
Clarify what alternate GALL/SRP-SLR AMR line items were used to evaluate the galvanized steel supports associated with AMR line items 3.5.1-093 and 3.5.1-095 and the applicability for ONS of the different galvanized steel supports/components described in the GALL-SLR Report for these AMR line items. Update the SLRA as necessary.

Response to RAI 3.5.1-093-1:

SRP-SLR Table 3.5-1, item 093, states that the component "Galvanized steel support members; welds; bolted connections; support anchorage to building structure" should be managed for loss of material due to pitting and crevice corrosion by the AMP XI.S6, “Structures Monitoring,” program when the component is exposed to an air outdoor environment. This material and environment combination is evaluated by SRP-SLR Table 3.5.1, item 092, for which components “Support members; welds; bolted connections; support anchorage to building structure” exposed to an air outdoor environment are managed for loss of material due to general, pitting corrosion by the AMP XI.S6, “Structures Monitoring,” program.

For “Galvanized steel support members; welds; bolted connections; support anchorage to building structure”, the SRP-SLR Table 3.5.1, item 095, states that no aging effects needs to be managed or AMP required when the component is exposed to an air indoor (uncontrolled) environment. This material and environment combination is evaluated by SRP-SLR Table 3.5.1, item 092, for which components “Support members; welds; bolted connections; support anchorage to building structure” exposed to an air indoor (uncontrolled) environment are managed for loss of material due to general, pitting corrosion by the AMP XI.S6, “Structures Monitoring,” program.

Table 3.5.1, Item 093 and 095 Discussion section is updated in the SLRA to include a reference denoting that Table 3.5.1, Item 092 is used to evaluate the component, material and environment combination for these AMR line items.

SLRA Revisions:

Table 3.5.1 Item 3.5.1-093 (page 3-1348) is revised as follows:

Table 3.5.1 Summary of Aging Management Programs for Containments, Structures and Component Supports Evaluated in Chapters II and III of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-093	Galvanized steel support members; welds; bolted connections; support anchorage to building structure	Loss of material due to pitting, crevice corrosion	AMP XI.S6, "Structures Monitoring"	No	Not ~~applicable~~ **used**. Galvanized steel components are evaluated using NUREG-2191 items for Steel. **Refer to Item Number 3.5.1-092.** The associated NUREG-2191 aging items are not used.

Table 3.5.1 Item 3.5.1-095 (page 3-1349) is revised as follows:

Table 3.5.1 Summary of Aging Management Programs for Containments, Structures and Component Supports Evaluated in Chapters II and III of the GALL-SLR Report

Item Number	Component	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-095	Galvanized steel support members; welds; bolted connections; support anchorage to building structure	None	None	No	Not ~~applicable~~ **used**. Galvanized steel components are evaluated using NUREG-2191 items for Steel. **Refer to Item Number 3.5.1-092.** The associated NUREG-2191 aging items are not used.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

ATTACHMENT 52
RAI B2.1.28-1

Enclosure 1, Attachment 52

RAI B2.1.28-1:

Regulatory Basis:
Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:

ASME Section XI, Subsection IWE
SLRA Section B2.1.28, as amended by ONS SLRA Supplement 3 dated December 15, 2021 (ADAMS Accession No. ML21349A005), states, in part, that the program will be enhanced to specify that for "high strength" structural bolting consisting of ASTM A325, ASTM F1852, ASTM F2280, and/or ASTM A490 bolts, the preventative actions for storage, lubrication, and stress corrosion cracking potential discussed in Section 2.0 of Research Council for Structural Connections publication "Specification for Structural Joints Using ASTM A325 or ASTM A490 Bolts," will be used.

SLRA Structures Monitoring
SLRA Section B2.1.33 states that the program will be enhanced to provide guidance for storage, lubricants, and the steps to minimize stress corrosion cracking potential discussed in Section 2 of Research Council for Structural Connections publication, "Specification for Structural Joints Using ASTM A325 or A490 Bolts" for structural bolting consisting of ASTM A325, ASTMF1852, and/or ASTM A490 bolts; and that program will be enhanced to provide guidance so that when replacement bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants will be in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339. SLRA Section B2.1.33 also states that the program will be enhanced to provide guidance for proper specification of new high strength bolting material and lubricant to prevent or mitigate degradation and failure of structural bolting in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339.

GALL-SLR report states that the preventive actions should emphasize proper selection of bolting material and lubricants and appropriate installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting. If the structural bolting consists of ASTM A325 and/or ASTM A490 bolts (including respective equivalent twist-off type ASTM F1852 and/or ASTM F2280 bolts), the preventive actions for storage, lubricant selection, and bolting and coating material selection discussed in Section 2 of Research Council for Structural Connections publication "Specification for Structural Joints Using High-Strength Bolts" need to be considered.

Issue:
It is unclear whether bolting and coating material selection discussed in Section 2 of the Research Council for Structural Connections publication will included in the enhancement to Element 2 in the ASME Section XI, Subsection IWE program.

It is unclear why preventive actions to ensure bolting integrity for "high strength" structural bolting consists of ASTM F2280 bolts were not addressed by the Structures Monitoring Program.

It is unclear whether coating material selection discussed in Section 2 of the Research Council for Structural Connections publication will included in the enhancement to Element 2 in the Structures Monitoring program.

Request:

1. Clarify whether bolting and coating material selection discussed in Section 2 of Research Council for Structural Connections publication will be included in the enhancement to Element 2 in the ASME Section XI, Subsection IWE program. Provide the justification if bolting and coating material selection is not included.
2. Clarify whether preventive action to ensure bolting integrity for “high strength” structural bolting consists of ASTM F2280 bolts will be included by the Structures Monitoring Program.
3. Clarify whether coating material selection discussed in Section 2 of the Research Council for Structural Connections publication will be included in the enhancement to Element 2 in the Structures Monitoring program. Provide the justification if coating material selection is not included.

Response to RAI B2.1.28-1:

Request 1:

The bolting and coating material selection discussed in Section 2 of Research Council for Structural Connections publication is included in the enhancement to Element 2 in the *ASME Section XI, Subsection IWE* program. Additionally, the bolting and coating material selection discussed in Section 2 of Research Council for Structural Connections publication is included in the enhancement to Element 2 in the *ASME Section XI, Subsection IWF* program.

Request 2:

The *Structures Monitoring* program preventive action to ensure bolting integrity for “high strength” structural bolting will include ASTM F2280 bolts.

Request 3:

The bolting and coating material selection discussed in Section 2 of Research Council for Structural Connections publication is included in the enhancement to Element 2 in the *Structures Monitoring* program.

SLRA Revisions:

SLRA Section A2.28 (page A-30 previously updated in Supplement 3 (ML21349A005)) is revised as follows:

Enhancements

The *ASME Section XI, Subsection IWE* AMP will be enhanced to:

Specify that for "high strength" structural bolting consisting of ASTM A325, ASTM F1852, ASTM F2280, and/or ASTM A490 bolts, the preventive actions for storage, lubrication, **bolting and coating material selection,** and stress corrosion cracking potential discussed in Section 2.0 of RCSC (Research Council for Structural Connections) publication *"Specification for Structural Joints Using ~~ASTM A325 or ASTM A490~~* ***High-Strength*** *Bolts,"* will be used. Procedures will be revised to specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, "*Degradation and Failure of Bolting in Nuclear Power Plants*," EPRI TR-104213, "*Bolted Joint Maintenance & Application Guide*," and the additional recommendations of NUREG-1339, "*Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants*".

SLRA Section A2.30 (page A-33, previously updated in Supplement 3 (ML21349A005)) is revised as follows:

Enhancements

2. Procedures will be revised to specify that for structural bolting consisting of ASTM A325, ASTMF1852, ASTM F2280, and/or ASTM A490, the preventive actions for storage, lubricants, **bolting and coating material selection,** and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication, "*Specification for Structural Joints Using ~~ASTM A325 or A490~~* ***High-Strength*** *Bolts*," will be used. (Element 2)

SLRA Section A2.33 (page A-36) is revised as follows:

Enhancements

4. For structural bolting consisting of ASTM A325, ASTMF1852, **ASTM F2280,** and/or ASTM A490, provide guidance for storage, lubricants, **bolting and coating material selection,** and the steps to minimize stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication, "*Specification for Structural Joints Using ~~ASTM A325 or A490~~* ***High-Strength*** *Bolts*".

SLRA Appendix Table A6.0-1 (pages A-94 previously updated in Supplement 3 (ML21349A005) and A-95) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
28	*ASME Section XI, Subsection IWE* program	The *ASME Section XI, Subsection IWE* AMP is an existing program that will be enhanced to: 1. The program will be enhanced to specify that for "high strength" structural bolting consisting of ASTM A325, ASTM F1852, ASTM F2280, and/or ASTM A490 bolts, the preventative actions for storage, lubrication, **bolting and coating material selection,** and stress corrosion cracking potential discussed in Section 2.0 of Research Council for Structural Connections publication *"Specification for Structural Joints Using ~~ASTM A325 or ASTM A490~~ **High-Strength** Bolts,"* will be used. Procedures will be revised to specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, "*Degradation and Failure of Bolting in Nuclear Power Plants*," EPRI TR-104213, "*Bolted Joint Maintenance & Application Guide*," and the additional recommendations of NUREG-1339, "*Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants*".	B2.1.28	Program enhancements for SLR will be implemented no later than 6 months prior to the SPEO.

SLRA Appendix Table A6.0-1 (pages A-96 previously updated in Supplement 3 (ML21349A005)) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
30	*ASME Section XI, Subsection IWF* program	2. Procedures will be revised to specify that for structural bolting consisting of ASTM A325, ASTMF1852, ASTM F2280, and/or ASTM A490, the preventive actions for storage, lubricants, **bolting and coating material selection,** and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication, "*Specification for Structural Joints Using ~~ASTM A325 or A490~~ **High-Strength** Bolts*," will be used. (Element 2)	B2.1.30	Program enhancements for SLR will be implemented and a one-time inspection of an additional 5% of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports is conducted within 5 years prior to the SPEO, and is to be completed

#	Program	Commitment	AMP	Implementation
				prior to the SPEO. Other enhancements are completed 6 months prior to the SPEO or no later than the last refueling outage prior to the SPEO.

SLRA Appendix Table A6.0-1 (pages A-100) is revised as follows:

Table A6.0-1: Subsequent License Renewal Commitments

#	Program	Commitment	AMP	Implementation
33	*Structures Monitoring* program	For structural bolting consisting of ASTM A325, ASTMF1852, **ASTM F2280,** and/or ASTM A490, provide guidance for storage, lubricants, **bolting and coating material selection,** and the steps to minimize stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication, "*Specification for Structural Joints Using ~~ASTM A325 or A490~~* ***High-Strength*** *Bolts*".	B2.1.33	Program enhancements for SLR will be implemented no later than 6 months prior to the SPEO.

SLRA Appendix B2.1.28 (page B-199 previously updated in Supplement 3 (ML21349A005)) is revised as follows:

Enhancements

The following enhancements will be implemented in the following program elements: Preventive Actions (Element 2), Parameters Monitored or Inspected (Element 3), and Detection of Aging Effects (Element 4).

1. The program will be enhanced to specify that for "high strength" structural bolting consisting of ASTM A325, ASTM F1852, ASTM F2280, and/or ASTM A490 bolts, the preventative actions for storage, lubrication, **bolting and coating material selection,** and stress corrosion cracking potential discussed in Section 2.0 of Research Council for Structural Connections publication *"Specification for Structural Joints Using ~~ASTM A325 or ASTM A490~~* ***High-Strength*** *Bolts,"* will be used. Procedures will be revised to specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, "*Degradation and Failure of Bolting in Nuclear Power Plants*," EPRI TR-104213, "*Bolted Joint Maintenance & Application Guide*," and the additional recommendations of NUREG-1339, "*Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants*". (Element 2)

SLRA Appendix B2.1.30 (page B-210 previously updated in Supplement 3 (ML21349A005)) is revised as follows:

Enhancements

2. Procedures will be revised to specify that for structural bolting consisting of ASTM A325, ASTMF1852, ASTM F2280, and/or ASTM A490, the preventive actions for storage, lubricants, **bolting and coating material selection,** and stress corrosion cracking potential discussed in Section 2 of Research Council for Structural Connections publication, "*Specification for Structural Joints Using ~~ASTM A325 or A490~~* ***High-Strength*** *Bolts*," will be used. (Element 2)

SLRA Appendix B2.1.33 (pages B-224) is revised as follows:

Enhancements

4. For structural bolting consisting of ASTM A325, ASTMF1852, **ASTM F2280,** and/or ASTM A490, provide guidance for storage, lubricants, **bolting and coating material selection,** and the steps to minimize stress corrosion cracking potential discussed in Section 2 of Research Council for Structural Connections publication, "*Specification for Structural Joints Using ~~ASTM A325 or A490~~* ***High-Strength*** *Bolts*". (Element 2)

ENCLOSURE 2

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 2

AFFIDAVITS

Attachment	Affidavit
1	Framatome Affidavit
2	Duke Energy Affidavit

AFFIDAVIT

1. My name is Philip A. Opsal. I am Manager, Product Licensing for Framatome Inc. (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Duke Energy Letter RA-22-0036, Subject: "Duke Energy Carolinas, LLC (Duke Energy), Oconee Nuclear Station (ONS), Units 1, 2, and 3, Docket Numbers 50-269, 50-270, 50-287, Renewed License Numbers DPR-38, DPR-47, DPR-55, Subsequent License Renewal Application, Responses to NRC Request for Additional Information Set 2," Enclosure 1, Attachments 14P, 15P, 25P, 27P, 28P, 29P, 30P, 31P, 32P, 42P and 43P (herein referred to as "this Document'). Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome’s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c), 6(d), and 6(e) above.

7. In accordance with Framatome’s policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on February 7, 2022.

Philip A. Opsal

AFFIDAVIT of Steven M. Snider

1. I am Site Vice President, Oconee, Duke Energy Carolinas, LLC ("Duke Energy") and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke Energy.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential. I am familiar with the Duke Energy information contained in Enclosure 1, Attachment 28, RAI 3.5.2.2.2.6-4 of the Oconee Subsequent License Renewal Application Responses to NRC Request for Additional Information Set 2 (Duke Energy letter dated February 14, 2022, Serial number RA-22-0036).

4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke Energy and has been held in confidence by Duke Energy and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke Energy. Information is held in confidence if it falls in one or more of the following categories.

      (a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by a vendor or consultant, without a license from Duke Energy, would constitute a competitive economic advantage to that vendor or consultant.

      (b) The information requested to be withheld consist of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage for example by requiring the vendor or consultant to perform test measurements, and process and analyze the measured test data.

      (c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation assurance of quality or licensing of a similar product.

      (d) The information requested to be withheld reveals cost or price information, production capacities, budget levels or commercial strategies of Duke Energy or its customers or suppliers.

(e) The information requested to be withheld reveals aspects of the Duke Energy funded (either wholly or as part of a consortium) development plans or programs of commercial value to Duke Energy.

(f) The information requested to be withheld consists of patentable ideas.

The information in this submittal is held in confidence for the reasons set forth in paragraphs 4(ii)(a), 4(ii)(b) and 4(ii)(c) above. Rationale for this declaration is the use of this information by Duke Energy provides a competitive advantage to Duke Energy over vendors and consultants, its public disclosure would diminish the information's marketability, and its use by a vendor or consultant would reduce their expenses to duplicate similar information. The information consists of analysis methodology details that provides a competitive advantage to Duke Energy.

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is that which is marked in Enclosure 1, Attachment 28, RAI 3.5.2.2.2.6-4 of the Oconee Subsequent License Renewal Application Responses to NRC Request for Additional Information Set 2 (Duke Energy letter dated February 14, 2022, Serial number RA-22-0036). This information enables Duke Energy to support the Oconee Subsequent License Renewal Application.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke Energy.

(a) Duke Energy uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke Energy.

5. Public disclosure of this information is likely to cause harm to Duke Energy because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke Energy to recoup a portion of its expenditures or benefit from the sale of the information.

Steve Snider affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on February 4, 2022.

Steven M. Snider