July 11, 2022

L-2022-108
10 CFR 54.17

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11545 Rockville Pike
One White Flint North
Rockville, MD 20852-2746

St. Lucie Nuclear Plant Units 1 and 2
Dockets 50-335 and 50-389
Renewed Facility Operating Licenses DPR-67 and NPF-16

SUBSEQUENT LICENSE RENEWAL APPLICATION - AGING MANAGEMENT REQUESTS FOR ADDITIONAL INFORMATION (RAI) SET 2 RESPONSE

References:

1. FPL Letter L-2021-192 dated October 12, 2021 – Subsequent License Renewal Application – Revision 1 (ADAMS Accession No. ML21285A107)
2. FPL Letter L-2022-043 dated April 7, 2022 – Subsequent License Renewal Application Revision 1 – Supplement 1 (ADAMS Accession No. ML22097A202)
3. FPL Letter L-2022-044 dated April 13, 2022 – Subsequent License Renewal Application Revision 1 – Supplement 2 (ADAMS Accession No. ML22103A014)
4. FPL Letter L-2022-071 dated May 19, 2022 – Subsequent License Renewal Application Revision 1 – Supplement 3 (ADAMS Accession No. ML22139A083)
5. NRC Email and Attachment dated June 8, 2022, St. Lucie SLRA RAI Safety Set 2 Final (ADAMS Accession Nos. ML22160A366, ML22160A367)

Florida Power & Light Company (FPL), owner and licensee for St. Lucie Nuclear Plant (PSL) Units 1 and 2, has submitted a revised and supplemented subsequent license renewal application (SLRA) for the Facility Operating Licenses for PSL Units 1 and 2 (References 1 - 4). Based on the NRC's review of the SLRA, the NRC issued its Set 1 RAIs to FPL (Reference 5). The attachments to this letter provide the response to those information requests.

For ease of reference, the index of attached information is provided on page 3 of this letter. Certain attachments include associated revisions to the SLRA (Enclosure 3 Attachment 1 of Reference 1, as supplemented by References 2 - 4) denoted by ~~strikethrough~~ (deletion) and/or **bold red underline** (insertion) text. Previous SLRA revisions are denoted by **bold black** text. SLRA table revisions are included as excerpts from each affected table.

Should you have any questions regarding this submittal, please contact me at (561) 304-6256 or William.Maher@fpl.com.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 11th day of July 2022.

Sincerely,

William Maher Digitally signed by William Maher
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Steam Generator AMP – Feedring and Support Inspections

RAI B.2.3.10-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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:

Section B.2.3.10 of Subsequent License Renewal Application (SLRA) Appendix B, dated August 3, 2021 (ADAMS Package Accession No. ML21215A314), stated that a visual inspection of the feedring and its supports in the Unit 2 steam generators (SGs) is performed every outage due to a history of water hammer events. During the audit of the Steam Generators program, the applicant stated that the feedring and its supports in the Unit 2 SGs are visually inspected each outage. The applicant also stated during the audit of the Steam Generators program that visual inspection of the accessible portions of the feedring and its supports in the Unit 1 SGs will be monitored in the future as part of the Steam Generators program.

Issue:

Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ADAMS Accession No. ML22097A202), revised Section B.2.3.10 of SLRA Appendix B by changing the inspection frequency of the feedring and its supports in the Unit 2 SGs from “every outage” to “regularly.” However, the description of the change to Section B.2.3.10 of SLRA Appendix B does not discuss the inspection frequency change of the feedring and its supports in the Unit 2 SGs. Therefore, it is unclear to the NRC staff why the change in inspection frequency of the feedring and its supports in the Unit 2 SGs was made. In addition, the SLRA does not discuss the inspection frequency of the feedring and its supports in the Unit 1 SGs.

Request:

Please discuss the basis, including how the inspection frequency supports prevention of loose parts that could impact SG tube integrity, for changing the inspection frequency of the feedring and its supports in the Unit 2 SGs from “every outage” to “regularly” given operating experience with water hammer events. In addition, please discuss the inspection frequency of the accessible portions of the feedring and its supports in the Unit 1 SGs.

PSL Response:

Since the discovery and reporting (ADAMS Accession No. ML15209A646) of damage to the St. Lucie Unit 2 steam generator (SG) feedring and supports in Spring 2014 (SL2-21), these components have been visually inspected every outage in which SG eddy-current (ECT) inspection was performed. SG ECT inspection was scheduled every outage from installation through SL2-24. Visual inspection results for the SG feedring and supports after the SL2-21 discovery are documented in Section G of past SG Tube Inspection Reports (ADAMS Accession Nos. ML16111B235, ML17257A080 and ML19081A146).

Review of plant-specific data revealed that in each case where feedring support deformation was found, it was linked to an identified pressure transient (water hammer) event in the prior operating cycle. As a result, limitations on SG secondary-side inspection and operation were put in place in 2016 (which required visual inspection of the affected SG feedring and supports at the next refueling outage) if established water-hammer monitoring criteria were met during the prior operating cycle. No inspection of the SG feedring and supports was performed in SL2-25 since SG ECT inspection was not scheduled, and the water-hammer monitoring criteria was not met in the prior operating cycle. In SL2-26 (Fall 2021), SG ECT inspection was scheduled. In addition, the water-hammer monitoring criteria was met in the prior operating cycle; therefore, inspection of the feedring and supports was performed as required by each of these conditions. Repairs to the feedring supports have been implemented during every refueling outage following a cycle where the feedring water-hammer monitoring criteria was met. In summary, the inspection frequency of the St. Lucie Unit 2 feedring and supports in each SG is every outage in which SG ECT inspection is performed. In addition, if a qualifying water-hammer event occurs during the cycle, the affected SG feedring and supports are visually inspected no later than the next refueling outage. This inspection schedule will continue into the subsequent period of extended operation.

As documented in the Response to RAI #3 on the SL2-24 SG Tube Inspection Report (ADAMS Accession No. ML19233A273), the feedring water-hammer events experienced in Cycles 21 and 23 on Unit 2 were the largest magnitude transients possible and did not generate loose parts based on the subsequent secondary-side inspections performed in the following outages. In addition, there has been no loose parts generated since re-design of the SG feedring support system in SL2-21. In the event a loose part is generated, the Unit 2 steam generators include a fine-mesh, loose part trapping (debris) screen designed to preclude loose parts from entering the tube bundle. Therefore, the inspection frequency described above supports prevention of loose parts that could impact SG tube integrity.

Accessible portions of the St. Lucie Unit 1 feedring and supports will be visually inspected at every outage in which SG ECT inspection is performed. In response to the visual inspection results for the feedring noted in Section G of the SL1-30 SG Tube Inspection Report (ADAMS Accession No. ML21305A868), the inspection frequency of these components was increased to allow an additional inspection and monitoring of the feedring and supports between scheduled SG ECT inspections. Therefore, visual inspection of accessible portions of the St. Lucie Unit 1 feedring and supports will be performed at least 2 times during each SG ECT inspection interval.

In addition to the changes made to SLRA Section B.2.3.10 in Attachment 18 of FPL Letter L-2022-043 (ADAMS Accession No. ML22097A202), the SLRA is revised to further discuss the inspection frequencies of the steam generator feedrings and supports.

References:

None.

Associated SLRA Revisions:

SLRA Section B.2.3.10 pages B-92 and B-93 are revised as follows:

Plant Specific Operating Experience

- A program health report review was performed covering the range of the first quarter of 2015 through the fourth quarter 2020. Health report attributes including owner proficiency, infrastructure, implementation, and equipment were Green for the majority of the review period. However, high tubing plugging rate in Unit 2 steam generator **A** due to mechanical wear at AVBs, and water hammer condition vulnerability on Unit 2 steam generators were gaps to Green. The plugging rate is being monitored during every outage. <u>**As a corrective action measure to address a prior history of condensation-induced water hammer events, v**</u>~~V~~isual inspection of the feedring and its supports in the Unit 2 steam generators is <u>**currently being**</u> performed ~~**regularly**~~ <u>**every outage in which SG ECT inspection is performed as well as in any outage where the water-hammer monitoring criteria was met during the prior operating cycle**</u>~~due to a prior history of condensation-induced water hammer events~~. **Due to design differences the Unit 1 feedring supports are not completely accessible for inspection. However, limited Unit 1 feedwater feedring support inspections** <u>**are currently performed at least two times during each steam generator eddy current tube inspection interval and**</u> **have not revealed any support degradation.**

Associated Enclosures:

None.

Steam Generator AMP – Tube Repair

RAI B.2.3.10-3

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 April 7, 2022 (ADAMS Accession No. ML22097A202), revised Section 19.2.2.10 of Appendices A1 and A2, Section B.2.3.10 of SLRA Appendix B to clarify that steam generator (SG) tubes not meeting the performance criteria are plugged, not repaired. St. Lucie Units 1 and 2 are not approved for alternate repair criteria or alternate repair methods.

Issue:

The Program Description of Section B.2.3.10 of SLRA Appendix B still includes the following instances related to SG tube repair:

- Second sentence of second paragraph - “repair of flawed tubes” and “acceptable tube repair methods.”
- Introductory text of fourth paragraph - “repair.”
- Last sentence of eighth paragraph - “repair criteria of flawed tubes.”

Request:

Please discuss the inclusion of the instances related to SG tube repair noted above. Alternatively, since St. Lucie is not approved for alternate repair criteria or alternate repair methods, revise Section B.2.3.10 of SLRA Appendix B to remove the instances related to SG tube repair noted above.

PSL Response:

St. Lucie does not perform any repair of steam generator tubes. SLRA Section B.2.3.10 is revised accordingly. In addition to the changes made to SLRA Section B.2.3.10 in Attachment 18 of FPL Letter L-2022-043 (ADAMS Accession No. ML22097A202), the SLRA is revised to remove references to repair of steam generator tubes.

References:

None.

Associated SLRA Revisions:

SLRA Section B.2.3.10 pages B-88 through B-90 are revised as follows:

B.2.1.1 Steam Generators

Program Description

The PSL Steam Generators AMP, previously the PSL Steam Generator Integrity program, is an existing AMP that manages the aging of steam generator tubes, plugs, divider plate assemblies, **tube-to-tubesheet welds,** heads (interior surfaces of channel or lower heads), tubesheet(s) (primary side), and secondary side components that are contained within the steam generator (i.e., secondary side internals). The aging of steam generator pressure vessel welds is managed by other AMPs such as the PSL ASME Section XI ISI, Subsections IWB, IWC, and IWD AMP (Section B.2.3.1), and the PSL Water Chemistry AMP (Section B.2.3.2).

The establishment of a steam generator program for ensuring steam generator tube integrity is required by the PSL Technical Specifications (TS). Additionally, Administrative Control requires tube integrity to be maintained to specific performance criteria, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging~~ or repair~~ of flawed tubes~~, acceptable tube repair methods~~, and leakage monitoring requirements. **Condition monitoring assessments are performed to determine whether tube integrity was met during the prior operating interval. Operational assessments are performed to ensure that tube integrity will be maintained until the next inspection.** The NDE techniques used to inspect steam generator components covered by this AMP are intended to identify components (e.g., tubes, plugs) with degradation that may need to be removed from service (e.g., tubes), or replaced **(e.g., plugs)**, as appropriate.

The PSL Steam Generators AMP is based on the guidelines provided in NEI 97-06 (Reference 1.6.32), Revision 3, “Steam Generator Program Guidelines.” As such, this AMP incorporates the following industry guidelines:

- EPRI 3002007572, “PWR Steam Generator Examination Guidelines” (Reference 1.6.33);
- EPRI 1022832, “PWR Primary-to-Secondary Leak Guidelines” (Reference 1.6.34);
- EPRI 3002000505, “Pressurized Water Reactor Primary Water Chemistry Guidelines”;
- EPRI 3002010645, “Pressurized Water Reactor Secondary Water Chemistry Guidelines”;
- EPRI 3002007571, “Steam Generator Integrity Assessment Guidelines” (Reference 1.6.35); and

- EPRI 3002007856, "Steam Generator In-Situ Pressure Test Guidelines" (Reference 1.6.36).

Through these guidelines, a balance of prevention, mitigation, inspection, evaluation, ~~repair,~~ and leakage monitoring measures are incorporated. Specifically, this AMP incorporates the following from NEI 97-06:

a. Performance criteria are intended to provide assurance that tube integrity is being maintained consistent with the CLB.

b. Guidance for monitoring and maintaining the tubes, which provides assurance that the performance criteria are met at all times between scheduled tube inspections.

Since degradation of divider plate assemblies, **tube-to-tubesheet welds,** channel heads (internal surfaces), or tubesheets (primary side) may have safety implications, the PSL Steam Generators AMP addresses degradation associated with steam generator tubes, plugs, divider plates, **tube-to-tubesheet welds,** interior surfaces of channel heads, tubesheets (primary side), and secondary side components that are contained within the steam generator (i.e., secondary side internals). This AMP does not include in its scope the steam generator secondary side shell, any nozzles attached to the secondary side shell or steam generator head, or the welds associated with these components. In addition, the scope of this AMP does not include steam generator primary side chamber welds (other than general corrosion of these welds caused as a result of degradation (defects/flaws) in the primary side cladding).

The PSL Steam Generators AMP includes preventive and mitigative actions for addressing degradation. This includes foreign material exclusion as a means to inhibit wear degradation and secondary side maintenance/cleaning activities, such as sludge lancing, for removing deposits that may contribute to degradation. Sludge lancing occurs when the steam generator is inspected, and inspections for remaining foreign material are performed after sludge lancing is completed. Primary side preventive maintenance activities include tube plug inspection, and replacement of tube plugs which are suspected of leakage. Additionally, this AMP works in conjunction with the PSL Water Chemistry AMP (Section B.2.3.2), which monitors and maintains water chemistry to reduce susceptibility to SCC or IGSCC.

The procedures associated with this AMP provide parameters to be monitored or inspected except for steam generator divider plates, **tube-to-tubesheet welds,** channel heads, and tubesheets. **PSL has submitted a license amendment request (ML21265A285) for the Technical Specifications conversion to NUREG-1432 Revision 5 in accordance with TSTF 577 Revision 1 (ML21098A188). Following NRC review and approval, including any applicable conditions or limitations, implementation of this license amendment will increase the inspection interval for 100% of the tubes to 96 EFPM. The 96 EFPM inspection interval for the tubes will also allow for visual inspection of the divider plates, tube-to-tubesheet welds, channel heads, and tubesheets in accordance with the Steam Generators AMP.** These inspections of the steam generator head interior surfaces, including the divider plate, are intended to identify signs that cracking, or loss of material may be occurring (e.g., through identification of rust stains).

Condition monitoring assessments are performed to determine whether the structural and accident-induced leakage performance criteria were satisfied during the prior operating interval. Operational assessments are performed to verify that structural and leakage integrity will be maintained for the planned operating interval before the next inspection. If tube integrity cannot be maintained for the planned operating interval before the next inspection, corrective actions are taken in accordance with the PSL CAP. Comparisons of the results of the condition monitoring assessment to the predictions of the previous operational assessment are performed to evaluate the adequacy of the previous operational assessment methodology. If the operational assessment was not conservative in terms of the number and/or severity of the condition, corrective actions are taken in accordance with the Steam Generator Integrity Assessment Guidelines. Assessment of tube integrity and plugging ~~or repair~~ criteria of flawed tubes is in accordance with the PSL TS.

Associated Enclosures:

None.

Steam Generator AMP – Tube Supports

RAI B.2.3.10-4

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 straight lengths of the tubes in the Unit 1 steam generators (SGs) are supported by lattice grid tube supports and the u-bend region of the tubes is supported by flat fan bars. The straight lengths of the tubes in the Unit 2 SGs are supported by broached support plates, and the u-bend region of the tubes are supported by anti-vibration bars. The Unit 2 SGs also have v-shaped support pads and v-shaped support bars.

Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ADAMS Accession No. ML22097A202), revised SLRA Tables 2.3.1-5, 3.1-1, and 3.1.2-5 to clarify that tube support plates are unique to the Unit 2 SGs and anti-vibration bars are applicable to both the Units 1 and 2 SGs. In addition, the description of the changes to the aforementioned SLRA tables stated that the anti-vibration bar component type includes the Unit 1 flat fan bars and the Unit 2 v-shaped support pads and v-shaped support bars. A similar statement was made by the applicant during the audit of the Steam Generators program.

Issue:

The SLRA does not include any discussion of the aging management of the Unit 1 flat fan bars (Section B.2.3.10 of SLRA Appendix B does refer to wear at fan bar supports) or the Unit 2 v-shaped support pads and v-shaped support bars. In addition, the SLRA does not include any discussion that the anti-vibration bar component type includes these additional tube support components. Therefore, a comprehensive description of the aging management of the tube support components for both Units 1 and 2 SGs appears to be missing from the SLRA.

Request:

Please justify why the SLRA does not include any discussion of the aging management of the Unit 1 flat fan bars and the Unit 2 v-shaped support pads and v-shaped support bars. Alternatively, revise the SLRA to clearly describe the aging management of the tube support components for both Units 1 and 2 (e.g., a plant-specific note associated with the anti-vibration bar component type that states it includes the Unit 1 flat fan bars and the Unit 2 v-shaped support pads and v-shaped support bars).

PSL Response:

The SLRA is revised to clarify the scope of the component type anti-vibration bars. Note that the v-shaped support pads are considered a subcomponent to the anti-vibration bar system and that v-shaped support bars are not a component type. The terminology difference of “pads” and “bars” is sometimes used to distinguish separate wear interactions involving the v-shaped support pads. This is further described in the St. Lucie response to RAIs 10 and 11 on the SL2-21 steam generator tube inspection report (ADAMS Accession No. ML15190A336). In addition to the changes made to SLRA Table 2.3.1-5 in Attachment 18 of FPL Letter L-2022-043 (ADAMS Accession No. ML22097A202), the SLRA is revised to include a note for the anti-vibration bars.

References:

None.

Associated SLRA Revisions:

SLRA Table 2.3.1-5 on page 2.3-13 is revised as follows:

Table 2.3.1-5
Steam Generator Components Subject to Aging Management Review

Component Type	Component Intended Function(s)
Anti-vibration bars[1]	**Structural support**
Blowdown nozzles	Pressure boundary
Bolting	Mechanical closure
Conical skirt	Structural support
Divider plates	Flow distribution
Feedwater feedring	Structural integrity (attached) Direct flow
Feedwater feedring supports	**Structural integrity (attached)**
Feedwater j-nozzle	Structural integrity (attached) Direct flow
Feedwater nozzle	Pressure boundary
Moisture separators	Structural integrity (attached)
Primary heads	Pressure boundary
Primary inlet and outlet nozzles	Pressure boundary
Primary instrument nozzles	Pressure boundary
Primary manway covers	Pressure boundary
Recirculation nozzles and end caps (Unit 2)	Pressure boundary
Secondary instrument nozzles	Pressure boundary
Secondary manway and handhole closure covers	Pressure boundary
Stay cylinders (Unit 1)	Pressure boundary
Steam generator components with fatigue analysis	Pressure boundary
Steam generator components: external surfaces	Pressure boundary Mechanical closure
Steam outlet nozzle (Unit 2)	Pressure boundary

Table 2.3.1-5
Steam Generator Components Subject to Aging Management Review

Component Type	Component Intended Function(s)
Steam outlet nozzle venturis (Unit 2)	Throttle
Steam outlet nozzle with integral flow orifices (Unit 1)	Pressure boundary Throttle
Tube bundle wrapper and wrapper supports	Structural support Direct flow
Tube-to-tubesheet welds	**Pressure boundary**
Tube plugs	Pressure boundary
Tube stabilizers (stakes) (Unit 2)	Structural support
Tube support lattice bars (Unit 1)	Structural support
Tube support plates (Unit 2)	Structural support
Tubesheets	Pressure boundary
Upper and lower shells, secondary head, transition cone	Pressure boundary
Upper vessel clevises and shear keys	Structural support
U-tubes	Pressure boundary Heat transfer

Note:

1. **This component type includes the Unit 1 flat fan bars and the Unit 2 v-shaped support pads.**

Associated Enclosures:

None.

Fire Water Systems AMP Implementation Schedule

RAI B.2.3.16-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 implementation schedule for the Fire Water System program in Table XI-01 of NUREG-2191, Volume 2," Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17187A204), states, "Program is implemented and [emphasis added] inspections or tests begin 5 years before the subsequent period of extended operation (SPEO). Inspections or tests that are to be completed prior to the subsequent period of extended operation are completed 6 months prior to the subsequent period of extended operation or no later than the last refueling outage prior to the subsequent period of extended operation."

Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ADAMS Accession No. ML22097A202), revised the implementation schedule for the Fire Water System program in Tables 19-3 in SLRA Appendices A1 and A2, and made conforming changes to Section B.2.3.16 in SLRA Appendix B. Specifically, the implementation schedule was revised, in part, to:

> Program inspections or tests begin 5 years before the SPEO…Inspections or tests that are to be completed prior to the SPEO are completed 6 months prior to the SPEO or no later than the last refueling outage prior to the SPEO.
>
> Program and SLR (subsequent license renewal) enhancements are implemented 6 months prior to the SPEO [emphasis added], …

Issue:

Implementing the program and SLR enhancements 6 months prior to the SPEO is not consistent with the GALL-SLR, which states that the program is implemented 5 years before the SPEO. In addition, beginning inspections and tests before implementing the program and SLR enhancements is not consistent with the GALL-SLR, which requires the program be implemented and [emphasis added] inspections or tests begin 5 years before the SPEO. It is unclear to the NRC staff how inspections and tests will be adequately managed (i.e., performance, acceptance criteria, corrective actions, etc.) as part of the Fire Water System program without the inspection and test requirements being incorporated into the Fire Water System program documentation.

Request:

Please provide the basis for the implementation schedule for the Fire Water System program and SLR enhancements, including discussion of how inspections and tests would be adequately managed prior to the Fire Water System program and SLR enhancements being implemented. Alternatively, revise the implementation schedule for the Fire Water System program in Tables 19-3 in SLRA Appendices A1 and A2, and Section B.2.3.16 in SLRA Appendix B to be consistent with the GALL-SLR.

PSL Response:

The Fire Water System aging management program (AMP) implementation schedules in Tables 19-3 in SLRA Appendices A1 and A2, and Section B.2.3.16 in SLRA Appendix B, as updated by Supplement 1 (ADAMS Accession No. ML22097A202), are revised to be consistent with the GALL-SLR (NUREG-2191, Table XI-01).

References:

None.

Associated SLRA Revisions:

SLRA revisions are presented on the following pages.

SLRA Appendix A1, Section 19.4, Table 19-3 commitment No. 19 (page A1-77), is revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
19	Fire Water System (19.2.2.16)	XI.M27	Continue the existing PSL Fire Water System AMP, including enhancement to: a) Update the governing AMP procedure to clearly state which procedures perform visual inspections for detecting loss of material, as well as state which procedures perform surface examinations or ASME Code, Section XI, VT-1 visual examinations for identifying SCC of copper alloy (>15% Zn) valve bodies, nozzles, and strainers. Such visual inspections will require using an inspection technique capable of detecting surface irregularities that could indicate an unexpected level of degradation due to corrosion and corrosion product deposition. Where such irregularities are detected, follow-up volumetric wall thickness examinations shall be performed. The internal inspections will be performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. At a minimum, in each 10-year period during the SPEO, a representative sample of 20% of the population (defined as components having the same combination of material, environment, and aging effect) or a maximum of 19 components per population at each Unit is inspected. Where practical, the inspections will focus on the bounding or lead components most susceptible to aging. b) Update the governing AMP procedure to clearly state which procedures perform volumetric wall thickness inspections. Volumetric inspections shall be conducted on the portions of the	**Program <u>is implemented and</u> inspections or tests begin 5 years before the SPEO <u>(i.e., 03/01/2031)</u>. ~~Inspections of the City Water Storage Tank bottoms begin 10 years before the SPEO.~~** **Inspections or tests that are to be completed prior to the SPEO are completed 6 months prior to the SPEO <u>(i.e., 09/01/2035)</u> or no later than the last refueling outage prior to the SPEO.** **<u>Inspections of the City Water Storage Tank bottoms begin 10 years before the SPEO (i.e., 03/01/2026).</u>** **~~Program and SLR enhancements are implemented 6 months prior to the SPEO, i.e.:~~** ~~PSL1: 09/01/2035~~

SLRA Appendix A2, Section 19.4, Table 19-3 commitment No. 19 (page A2-78), is revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
19	Fire Water System (19.2.2.16)	XI.M27	Continue the existing PSL Fire Water System AMP, including enhancement to: a) Update the governing AMP procedure to clearly state which procedures perform visual inspections for detecting loss of material, as well as state which procedures perform surface examinations or ASME Code, Section XI, VT-1 visual examinations for identifying SCC of copper alloy (>15% Zn) valve bodies, nozzles, and strainers. Such visual inspections will require using an inspection technique capable of detecting surface irregularities that could indicate an unexpected level of degradation due to corrosion and corrosion product deposition. Where such irregularities are detected, follow-up volumetric wall thickness examinations shall be performed. The internal inspections will be performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. At a minimum, in each 10-year period during the SPEO, a representative sample of 20% of the population (defined as components having the same combination of material, environment, and aging effect) or a maximum of 19 components per population at each Unit is inspected. Where practical, the inspections will focus on the bounding or lead components most susceptible to aging. b) Update the governing AMP procedure to clearly state which procedures perform volumetric wall thickness inspections. Volumetric inspections shall be conducted on the portions of the water-based fire protection system components that are periodically subjected to flow but are normally dry. c) Update existing inspection/testing procedures and create new, …	**Program <u>is implemented and</u> inspections or tests begin 5 years before the SPEO <u>(i.e., 04/06/2038)</u>. ~~Inspections of the City Water Storage Tank bottoms begin 10 years before the SPEO.~~** **Inspections or tests that are to be completed prior to the SPEO are completed 6 months prior to the SPEO <u>(i.e., 10/06/2042)</u> or no later than the last refueling outage prior to the SPEO.** **<u>Inspections of the City Water Storage Tank bottoms begin 10 years before the SPEO (i.e., 04/06/2033).</u>** **~~Program and SLR enhancements are implemented 6 months prior to the SPEO, i.e.:~~** ~~PSL2: 10/06/2042~~

SLRA Section B.2.3.16, page B-129, is revised as follows:

The CWSTs are externally inspected annually and internally inspected and volumetrically (UT) examined on a 5-year interval, which includes UT examination of the tank bottoms. These wall thickness examinations will also examine the CWST bottom surfaces in accordance with the NUREG-2191, Table XI.M29 1. Specifically, for each 10-year period **beginning** 10 years before the SPEO, a volumetric inspection will be required to be performed from the inside surface of the tanks. The new tank bottom thickness inspections will use a low-frequency electromagnetic testing (LFET) technique and, as necessary, follow-up ultrasonic examinations. Any regions below nominal plate thickness will have a follow-up ultrasonic thickness reading. If there are areas of significant loss of material that could impact the pressure boundary function, future ultrasonic thickness measurements and trending will be performed.

Results from the various surveillances are evaluated per the respective procedures. Any degradation identified by visual/volumetric inspections or flushes/flow testing is reported, evaluated, and corrected through the PSL corrective action program. Acceptance criteria for observed degradation, flow obstruction, discharge flow/pressures, or minimum wall thickness are defined in the PSL Fire Water System AMP procedures used to perform the respective inspections and tests.

NUREG-2191 Consistency

The PSL Fire Water System AMP, with enhancements, will be consistent without exception to the 10 elements of NUREG-2191, Section XI.M27, “Fire Water System”.

Exceptions to NUREG-2191

None.

Enhancements

The PSL Fire Water System AMP will be enhanced as follows, for alignment with NUREG-2191. **The PSL Fire Water System AMP is implemented 5 years before the SPEO.** **I**nspections and tests begin 5 years **before** the SPEO**, except for the inspections of the City Water Storage Tank bottoms, which begin 10 years before the SPEO**. The inspections and tests are to be completed no later than six months prior to entering the SPEO or no later than the last refueling outage prior to the SPEO.

Associated Enclosures:

None.

Fire Protection AMP – Commitment Schedule

RAI B.2.3.15-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 implementation schedule for the Fire Protection program in Table XI-01 of NUREG-2191, Volume 2," Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report" (ML17187A204), states, "Program and SLR enhancements, when applicable, are implemented 6 months prior to the subsequent period of extended operation." However, Tables 19-3 in SLRA Appendices A1 and A2 state, in part, that the implementation schedule for the Fire Protection program is "No later than 6 months prior to the SPEO, or no later than the last refueling outage prior to the SPEO…"

Issue:

The statement "or no later than the last refueling outage prior to the SPEO" in Tables 19-3 in SLRA Appendices A1 and A2 is not consistent with the GALL-SLR.

Request:

Please provide the basis for including "or no later than the last refueling outage prior to the SPEO" in the implementation schedule for the Fire Protection program. Alternatively, revise the implementation schedule for the Fire Program in Tables 19-3 in SLRA Appendices A1 and A2 to be consistent with the GALL-SLR.

PSL Response:

The implementation schedule for the Fire Protection AMP in Table 19-3 in SLRA Appendices A1 and A2 is revised to delete the phrase "or no later than the last refueling outage prior to the SPEO." To address extent of condition, other AMPs in Table 19-3 in SLRA Appendices A1 and A2 that have the same phrase and for which Tables X-01 and XI-01 of the GALL-SLR do not mention the last refueling outage are similarly changed, unless PSL identifies pre-SPEO inspections or modifications as part of the AMP. The AMPs to which this applies are: X.M1,

X.M2, XI.M1, XI.M2, XI.M3, XI.M10, XI.M11B, XI.M12, XI.M16A, XI.M17, XI.M18, XI.M19, XI.M21A, XI.M23, XI.M24, XI.M26, XI.M31, XI.M36, XI.M38, XI.M39, XI.S4, XI.S5, XI.S6, XI.S7, XI.S8, and the PSL site-specific AMPs on pages A1-64 through A1-77, A1-88, A1-90 through A1-96, and A1-102 through A1-105, A1-109 and A2-65 through A2-77, A2-89 through A2-96, A2-102 through A2-106, and A2-110 as applicable. This correlates to changes to commitments 1 through 18, 22, 26 through 28, 34 through 38, and 47 in both Tables 19-3. A similar change is made to SLRA Appendix B, Section B.2.3.24, in the last sentence of the Program Description subsection on page B-201. The SLRA revisions shown in this RAI response include changes proposed in Attachments 6 (RAI B.2.3.15-2), 7 (RAI B.2.3.15-3), 12 (RAI B.2.3.33-1), and 15 (RAI 3.5.2.2.2.1-1) of this letter. This RAI response also makes an editorial correction in commitment 26 o) for the External Surfaces Monitoring of Mechanical Components AMP (XI.M36) on pages A1-95 and A2-96, and the correlating enhancement (7. Corrective Actions) from Section B.2.3.23 on pages B-195 through B-196.

References:

None.

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.4, Table 19-3, pages A1-64 through A1-77, A1-88, A1-90 through A1-96, A1-102 through A1-105, and A1-109 as appropriate, is revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
1	Fatigue Monitoring (19.2.1.1)	X.M1	Continue the existing PSL Fatigue Monitoring AMP, including enhancement to: a) Update the plant procedure to monitor chemistry parameters that provide inputs to F_{en} factors used in CUF_{en} calculations. b) Update the plant procedure to identify and require monitoring of the 80-year projected plant transients that are utilized as inputs to CUF_{en} calculations. **These transients include:** • **The plant loading/unloading transient and the 10 percent step load increase/decrease transient.** c) Update the plant procedure to monitor and track the **following** transient**s** during the SPEO**:** • **Loss of charging** • **Loss of letdown** • **Loss of regenerative heat exchanger (short-term)** • **Loss of regenerative heat exchanger (long-term)** d) Update the plant procedure to identify the corrective action options to take if component specific fatigue limits are approached.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

2	Neutron Fluence Monitoring (19.2.1.2)	X.M2	Continue the existing PSL Neutron Fluence Monitoring AMP, including enhancement to: a) Follow the related industry efforts, such as by the Pressurized Water Reactor Owners Group (PWROG) and use the information from supplemental nozzle region dosimetry measurements and reference cases or other information to provide additional justification for use of the approved WCAP-18124-NP-A or similar methodology for the determination of RPV fluence in regions above or below the active fuel region. b) Include justification that draws from Westinghouse's NRC approved RPV fluence calculation methodology and includes discussion of the neutron source, synthesis of the flux field and the order of angular quadrature (e.g., S8), etc. used in the estimates for projection of TLAA to 80 years.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

4	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (19.2.2.1)	XI.M1	Continue the existing ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
5	Water Chemistry (19.2.2.2)	XI.M2	Continue the existing PSL Water Chemistry AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
6	Reactor Head Closure Stud Bolting (19.2.2.3)	XI.M3	Continue the existing PSL Reactor Head Closure Stud Bolting AMP, including enhancement to: a) Procure reactor head closure stud materials to limit the maximum yield strength of replacement material to a measured yield strength less than 150 ksi and a maximum tensile strength of 170 ksi.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			b) Preclude the use of molybdenum disulfide (MoS_2) lubricant for the reactor head closure stud bolting. **c) Pursuant to 10 CFR 50.55a(z)(1), submit proposed alternatives for relief from the schedule of reactor pressure vessel (RPV) bolting examinations specified in ASME Section XI Code, Table IWB-2500-1, Category B-G-1, and IWB-2420, in order to accommodate an additional set of reactor vessel closure studs, nuts, and washers that are shared between PSL Units 1 and 2 in rotation. A proposed alternative will be submitted for approval for each subsequent ISI interval through the remainder of the SPEO.**	
7	Boric Acid Corrosion (19.2.2.4)	XI.M10	Continue the existing PSL BAC AMP, including enhancement to: a) Include other potential means to help in the identification of borated water leakage, such as the following, in order to identify potential borated water leaks inside containment that have not been detected during walkdowns and maintenance: • Airborne radioactivity monitoring • Humidity monitoring (for trending increases in humidity levels due to unidentified RCS leakage) • Temperature monitoring (for trending increases in room/area temperatures due to unidentified RCS leakage) • Containment air cooler thermal performance monitoring (for corroborating increases in containment atmosphere temperature or humidity with decreases in cooler efficiency due to boric acid plate out) b) Include a requirement in the PSL Inspection of Internal Surfaces of Miscellaneous Piping and Ducting Components AMP implementing documents to document evidence of boric acid residue (plating out of moist steam) inside containment cooler housings or similar locations such as cooling unit drain pans and to enter evidence in to the corrective action program to be	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			evaluated under a boric acid corrosion control (BACC) program evaluation.	
8	Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (19.2.2.5)	XI.M11B	Continue the existing PSL Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components AMP, including enhancement to: a) Update the plant modification process to ensure that no additional alloy 600 material will be used in reactor coolant pressure boundary applications during the SPEO or that, if used, appropriate baseline and subsequent inspections per MRP inspection guidance will be put in place.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
9	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (19.2.2.6)	XI.M12	Continue the existing PSL Thermal Aging Embrittlement of Cast Austenitic Stainless Steel AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
10	Reactor Vessel Internals (19.2.2.7)	XI.M16A	Continue the existing PSL Reactor Vessel Internals AMP, including enhancement to: a) Implement the results of the gap analysis or implement the latest NRC-approved version of MRP-227 if it addresses 80 years of operation.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
11	Flow-Accelerated Corrosion (19.2.2.8)	XI.M17	Continue the existing PSL Flow-Accelerated Corrosion AMP, including enhancement to: a) Reassess piping systems excluded from wall thickness monitoring due to operation less than 2% of plant operating time (as allowed by NSAC-202L-R4) to ensure the exclusion remains valid and applicable for operation through 80 years. If actual wall thickness information is not available for use in this re-assessment, a representative sampling approach will be used. This re-assessment may result in additional inspections. b) Extend the erosion inspection plan for the duration of the SPEO.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			c) Perform opportunistic visual inspections of internal surfaces during routine maintenance activities to identify degradation. d) Revise or provide procedure(s) for measuring wall thickness due to erosion. Wall thickness should be trended to adjust the monitoring frequency and to predict the remaining service life of the component for scheduling repairs or replacements. e) Revise or provide procedure(s) to evaluate inspection results to determine if assumptions in the extent-of-condition review remain valid. If degradation is associated with infrequent operational alignments, such as surveillances or pump starts/stops, then trending activities should consider the number or duration of these occurrences. f) Revise or provide procedure(s) to perform periodic wall thickness measurements of replacement components until the effectiveness of corrective actions have been confirmed. g) Include long-term corrective actions for erosion mechanisms. The effectiveness of the corrective actions should be verified. Include periodic monitoring activities for any component replaced with an alternative material since no material is completely resistant to erosion.	
12	Bolting Integrity (19.2.2.9)	XI.M18	Continue the existing PSL Bolting Integrity AMP, including enhancement to: a) Ensure references to EPRI Reports 1015336, 1015337, and NUREG-1339 are added and guidance incorporated, as appropriate, for selection of bolting material and the use of lubricants and sealants. b) Ensure lubricants containing molybdenum disulfide (MoS_2) or other lubricants containing sulfur will not be used for pressure-retaining bolting. c) Ensure that the maximum yield strength of replacement or newly procured pressure-retaining bolting material will be limited to an actual yield strength less than 150 ksi (1,034 MPa). In addition, ensure bolting material with a yield strength greater than or equal	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			to 150 ksi (1,034 MPa) or for which yield strength is unknown will not be used for pressure retaining bolting. For closure bolting greater than 2-inches in diameter (regardless of code classification) with actual yield strength greater than or equal to 150 ksi (1,034 MPa) or for which yield strength is unknown is used, volumetric examination will be required in accordance to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 acceptance standards, extent, and frequency of examination. d) Perform alternative means of testing and inspection for closure bolting where leakage is difficult to detect (e.g., piping systems that contain air or gas or submerged bolting). The acceptance criteria for the alternative means of testing will be no indication of leakage from the bolted connections. Required inspections will be performed on a representative sample of the population (defined as the same material and environment combination) of bolt heads and threads over each 10-year period of the SPEO. The representative sample will be 20% of the population (up to a maximum of 19 per unit). The alternative testing will be completed on a case-by-case basis through: • Visual inspections of closure bolting during maintenance activities that make the bolt heads accessible and bolt threads visible; • Visual inspection for discoloration is conducted when leakage of the environment inside the piping systems would discolor the external surfaces; • Monitoring and trending of pressure decay is performed when the bolted connection is located within an isolated boundary; • Soap bubble testing, or;	

			• Thermography testing when the temperature of the fluid is higher than ambient conditions. e) Ensure that bolted joints that are not readily visible during plant operations and refueling outages will be inspected when they are made accessible and at such intervals that would provide reasonable assurance the components' intended functions are maintained. f) Ensure that closure bolting inspections will include consideration of the guidance applicable for pressure boundary bolting in NUREG-1339 and in EPRI NP-5769. g) Project, where practical, identified degradation until the next scheduled inspection. Results will be evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the SPEO based on the projected rate of degradation. For sampling-based inspections, results will be evaluated against acceptance criteria to confirm that the sampling bases (e.g., selection, size, frequency) will maintain the components' intended functions throughout the SPEO operation based on the projected rate and extent of degradation. Adverse results will be evaluated to determine if an increased sample size or inspection frequency is required. h) Evaluate leakage monitoring and sample expansion and add additional inspections if inspection results do not meet acceptance criteria as described in NUREG-2191, Chapter XI.M18, Element 7.	
13	Steam Generators (19.2.2.10)	XI.M19	Continue the existing PSL Steam Generators AMP.	No later than 6 months prior to the SPEO, i.e.: PSL1: 09/01/2035

15	Closed Treated Water Systems (19.2.2.12)	XI.M21A	Continue the existing PSL Closed Treated Water Systems AMP, including enhancement to: a) Ensure that the new visual inspection procedure(s) and/or preventive maintenance requirements evaluate the visual appearance of surfaces for evidence of loss of material on the internal surfaces exposed to the treated closed recirculating cooling water. b) Create new procedure(s) and/or preventive maintenance requirements that perform surface or volumetric examinations and evaluate the examination results for surface discontinuities indicative of cracking on the internal surfaces exposed to the treated closed recirculating cooling water. c) Ensure that visual inspections of closed treated water system components' internal surfaces are conducted whenever the system boundary is opened. When opportunistic visual inspections are conducted while the system boundary is open, they can be credited towards the representative samples for the loss of material and fouling; however, surface, or volumetric examinations must be used to confirm that there is no cracking. d) Create new procedure(s) and/or preventive maintenance requirements to ensure that the inspection requirements from NUREG-2191 are met. At a minimum, in each 10-year period during the SPEO, a representative sample of components is inspected using techniques capable of detecting loss of material, cracking, and fouling, as appropriate. The sample population is defined as follows: • 20% of the population (defined as components having the same material, water treatment program, and aging effect combination) OR; • A maximum of 19 components per population at each Unit since PSL is a two-Unit plant. e) Ensure that the new inspection and test procedure(s) and/or preventive maintenance requirements will evaluate their	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			respective results against acceptance criteria to confirm that the sampling bases (e.g., selection, size, frequency) will maintain the components' intended functions throughout the SPEO based on the projected rate and extent of degradation. Where practical, identified degradation is projected through the next scheduled inspection. f) Ensure that the new inspection and test procedure(s) and/or preventive maintenance requirements report and evaluate any detectable loss of material, cracking, or fouling associated with the surfaces exposed to the treated closed recirculating cooling water per the PSL corrective action program. g) Ensure that the following additional inspections and actions are required if a post-repair/replacement inspection or subsequent inspection of surfaces exposed to the treated closed cooling water environment fails to meet acceptance criteria: • The number of increased inspections is determined in accordance with the PSL corrective action process; however, there are no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20% of each applicable material, environment, and aging effect combination is inspected, whichever is less. • If subsequent inspections do not meet acceptance criteria, an extent-of-condition and extent-of-cause analysis is conducted to determine the further extent of inspections. • Additional samples are inspected for any recurring degradation to ensure corrective actions appropriately address the associated causes. Since PSL is a two-Unit site, the additional inspections include inspections at both Units with the same material, environment, and aging effect combination. • The additional inspections are completed within the interval (e.g., refueling outage interval, 10-year inspection interval) in which the original inspection was conducted.	

16	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (19.2.2.13)	XI.M23	Continue the existing PSL Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems AMP, including enhancement to: a) Update the implementing procedure to state that, for the in-scope systems that are infrequently in service, such as the containment polar cranes, periodic inspections are performed once every refueling cycle just prior to use. b) Update the implementing procedure and inspection procedures to state their respective visual inspection frequencies required by ASME B30.2-2005. According to ASME B30.2-2005, inspections are performed within the following intervals: • "Periodic" visual inspections by a designated person are required and documented yearly for normal service applications • A crane that is used in infrequent service, which has been idle for a period of one year or more, shall be inspected before being placed in service in accordance with the requirements listed in ASME B30.2-2005 paragraph 2-2.1.3 (i.e., periodic inspection) c) Update the implementing procedure to ensure that the inspection procedures for the individual load handling systems are clearly identified and referenced. d) Update the governing procedure to state that any visual indication of loss of material, deformation, or cracking, and any visual sign of loss of bolting preload for NUREG 0612 load handling systems is evaluated according to ASME B30.2-2005. e) Update the governing procedure to state that repairs made to NUREG-0612 load handling systems are performed as specified in ASME B30.2-2005.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
17	Compressed Air Monitoring (19.2.2.14)	XI.M24	Continue the existing PSL Compressed Air Monitoring AMP, including enhancement to formalize compressed air monitoring activities in a new governing procedure addressing the element by element	No later than 6 months prior to the SPEO, ~~or no later than the~~

			requirements presented in NUREG-2191 Section XI.M24. The following enhancements are also to be included into this procedure and other pertinent documents: a) Incorporate the air quality provisions provided in the guidance of the EPRI TR-108147 and consider the related guidance in the ASME OM-2012, Division 2, Part 28. b) Perform opportunistic visual inspections of accessible internal surfaces for signs of corrosion and abnormal corrosion products that might indicate a loss of material within the system. c) Include inspections of internal air line surfaces downstream of the instrument air dryers and emergency diesel generator air start dryers with maintenance, corrective, or other activities that involve opening of the component or system. d) Include inspection methods for the opportunistic inspections with guidance of standards or documents such as ASME OM-2012, Division 2, Part 28. e) Review air quality test results. f) Include requirements for better long-term trending of negative trends, more thorough documentation, and proactive aging management. g) Include monitoring and trending guidance from ASME OM-2012, Division 2, Part 28 as applicable.	~~last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
18	Fire Protection (19.2.2.15)	XI.M26	Continue the existing PSL Fire Protection AMP, including enhancement to: a) Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.**	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			b) Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** c) Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable. d) Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical. e) Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation. **f) Enhance plant procedures to specify that:** • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).** • **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered**	

			by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.	

22	Reactor Vessel Material Surveillance (19.2.2.19)	XI.M31	Continue the existing PSL Reactor Vessel Material Surveillance AMP, including an incremental adjustment to the approved capsule withdrawal schedule to withdraw and test the surveillance capsules located at 263^{o} and 83^{o} in accordance with the NRC approved withdrawal schedule.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

26	External Surfaces Monitoring of Mechanical Components (19.2.2.23)	XI.M36	Continue the existing PSL External Surfaces Monitoring of Mechanical Components AMP, including enhancement to: a) Indicate the material and environment combinations where external examinations could be credited to manage the aging effects of the internal surfaces of components as detailed in the PSL External Surfaces Monitoring of Mechanical Components AMP. b) Incorporate the aging management activities currently performed for external corrosion of insulated piping at PSL in the PSL External Surfaces Monitoring of Mechanical Components program procedure. c) Ensure all components made of stainless steel, aluminum, or copper alloys with greater than 15% Zn or 8% Al inspected by this program will have periodic visual or surface examinations conducted to manage cracking. d) Monitor the aging effects for elastomeric and flexible polymeric components through a combination of visual inspection and manual or physical manipulation of the material. Manual or physical manipulation of the material will include touching, pressing on, flexing, bending, or otherwise manually interacting with the material. The purpose of the manual manipulation will be to reveal changes in material properties, such as hardness, and to make the visual examination process more effective in identifying aging effects such as cracking. Flexing of polymeric components (e.g., expansion joints) exposed directly to sunlight (i.e., not located in a structure restricting access to sunlight such as manholes, enclosures, and vaults or isolated from the environment by coatings) will be conducted to detect potential reduction in impact strength as indicated by a crackling sound or surface cracks when flexed. Examples of inspection parameters for elastomers and polymers will include: • Surface cracking, crazing, scuffing, and dimensional change (e.g., “ballooning” and “necking”),	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			• Loss of thickness, • Discoloration (evidence of a potential change in material properties that could be indicative of polymeric degradation), • Exposure of internal reinforcement for reinforced elastomers, • Hardening as evidenced by a loss of suppleness during manipulation where the component and material are appropriate to manipulation. e) Specify that this program will also manage hardening or loss of strength, loss of preload for heating, ventilation, and air conditioning (HVAC) closure bolting, and blistering using visual inspections. In addition, physical manipulation will be used to manage hardening or loss of strength and reduction in impact strength. f) Specify that, when required by the ASME Code, inspections will be conducted in accordance with the applicable code requirements. And, when non-ASME Code inspections and tests are required, inspections will follow site procedures that include inspection parameters for items such as lighting, distance, offset, surface coverage, and presence of protective coatings. Inspections, except those for cracking and under insulation, will be performed every refueling outage. g) Ensure that periodic visual inspections or surface examinations will be conducted on components made of stainless steel, aluminum, or copper alloys with greater than 15% Zn or 8% Al to manage cracking every 10 years during the SPEO and other inspections will be performed at a frequency not to exceed one refueling cycle. Surfaces that are not readily visible during plant operations and refueling outages are inspected when they are made accessible and at such intervals that would provide reasonable assurance that the components’ intended functions are maintained.	

			h) Specify that, when inspecting to manage cracking of a component's material, either surface examinations conducted in accordance with plant-specific procedures or ASME Code Section XI VT-1 inspections (including those inspections conducted on non-ASME Code components) are conducted on each component inspected. An inspection requires that at least 20% of the surface area of the component is inspected, unless the component is measured in linear feet, such as piping. Any combination of 1-ft length sections and components can be used to meet the recommended extent of 20% of the population of materials and environment combinations, with a maximum of 25 inspections required in each population. An inspection of a component in a more severe environment may be credited as an inspection for the specified environment and for the same material and aging effects in a less severe environment (e.g., an outdoor air environment is more severe than an indoor uncontrolled air environment which is more severe than an indoor controlled air environment, assuming that there are no borated water leaks in the indoor environments). i) Specify that, when inspecting insulated components in an outdoor environment or that may be exposed to condensation in an indoor environment, that the population and sample sizes used for inspections will be determined based on the material type (e.g., steel, stainless steel, copper alloy, aluminum) and environment (e.g., air outdoor, air accompanied by leakage) combination. A minimum of 20% of the in-scope piping length, or 20% of the surface area for components whose configuration does not conform to a 1-ft axial length determination (e.g., valve, accumulator, tank) is inspected after the insulation is removed. Alternatively, any combination of a minimum of twenty-five 1-ft axial length sections and components for each material type is inspected, with a maximum of 25 inspections required in each population. j) Ensure that visual inspections identify indirect indicators of elastomer and flexible polymer hardening or loss of strength, including the presence of surface cracking, crazing, discoloration,	

			and, for elastomers with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Visual inspections will cover 100% of accessible component surfaces. Visual inspection will identify direct indicators of loss of material due to wear to include dimension change, scuffing, and, for flexible polymeric materials with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Manual or physical manipulation can be used to augment visual inspection to confirm the absence of hardening or loss of strength for elastomers and flexible polymeric materials (e.g., heating, ventilation, and air conditioning flexible connectors) where appropriate. The sample size for manipulation will be at least 10% of available surface area. k) Indicate that the following alternatives to removing insulation after the initial inspection will be acceptable: i. Subsequent inspections may consist of examination of the exterior surface of the insulation with sufficient acuity to detect indications of damage to the jacketing or protective outer layer (if the protective outer layer is waterproof) of the insulation when the results of the initial inspections meet the following criteria: • No loss of material due to general, pitting, or crevice corrosion beyond that which could have been present during initial construction is observed during the first set of inspections, and • No evidence of SCC is observed during the first set of inspections. If: (a) the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or jacketing, (b) there is evidence of water intrusion through the insulation (e.g., water seepage through insulation seams/joints), or (c) the protective outer layer (where jacketing is not installed) is not waterproof, then periodic inspections under the insulation should continue as conducted for the initial inspection.	

			ii. Removal of tightly adhering insulation that is impermeable to moisture is not required unless there is evidence of damage to the moisture barrier. If the moisture barrier is intact, the likelihood of corrosion under insulation is low for tightly adhering insulation. Tightly adhering insulation is considered to be a separate population from the remainder of insulation installed on in-scope components. The entire population of in-scope piping that has tightly adhering insulation is visually inspected for damage to the moisture barrier with the same frequency as for other types of insulation inspections. These inspections are not credited towards the inspection quantities for other types of insulation. l) Specify that results are evaluated against acceptance criteria to confirm that the sampling bases (e.g., selection, size, frequency) will maintain the components' intended functions throughout the SPEO based on the projected rate and extent of degradation. m) Include evaluation and acceptance guidance from EPRI TR-1009743, "Aging Identification and Assessment Checklist," for visual/tactile inspections where appropriate. n) Specify that inspections to detect cracking in aluminum, stainless steel, and applicable copper alloy components will have additional inspections conducted if one of the inspections does not meet the acceptance criteria due to current or projected degradation (i.e., trending) unless the cause of the aging effect for each applicable material and environment is corrected by repair or replacement for all components constructed of the same material and exposed to the same environment. The number of increased inspections will be determined in accordance with the site's corrective action process; however, there will be no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20% of each applicable material, environment, and aging effect combination is inspected, whichever is less. The additional inspections are completed within the interval in which the original inspection was conducted. If subsequent inspections do not meet acceptance criteria, an extent of condition and extent of cause analysis will be conducted to determine the further extent	

			of inspections. Additional samples will be inspected for any recurring degradation to provide reasonable assurance that corrective actions appropriately address the associated causes. The additional inspections include populations with the same material, environment, and aging effect combinations at both Unit 1 and Unit 2. o) Require that any projected inspection results **that** will not meet acceptance criteria prior to the next scheduled inspection, will have their inspection frequencies adjusted as determined by the corrective action program.	
27	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (19.2.2.24)	XI.M38	Implement the new PSL Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
28	Lubricating Oil Analysis (19.2.2.25)	XI.M39	Continue the existing PSL Lubricating Oil Analysis AMP, including enhancement to: a) Perform sampling and testing of old oil following periodic oil changes or on a schedule consistent with equipment manufacturer's recommendations or industry standards [e.g., ASTM D6224-02]. Plant specific OE associated with oil systems may also be used to adjust the schedule for periodic sampling and testing, when justified by prior sampling results. b) Ensure guidance indicates that phase-separated water in any amount is not acceptable. If phase-separated water is identified in the sample, then corrective actions are to be initiated to identify the source and correct the issue (e.g., repair/replace component or modify operating conditions).	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

34	10 CFR Part 50, Appendix J (19.2.2.31)	XI.S4	Continue the existing PSL 10 CFR Part 50, Appendix J AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
35	Masonry Walls (19.2.2.32)	XI.S5	Continue the existing PSL Masonry Walls AMP, including enhancement to: a) Revise the implementing procedure to monitor and inspect for gaps between the supports and masonry walls that could potentially impact the intended function or potentially invalidate its evaluation basis. b) Revise the implementing procedure to include specific monitoring, measurement, and trending of 1) widths and lengths of cracks in masonry walls and mortar joints, and 2) gaps between supports and masonry walls. c) Revise the implementing procedure to include specific guidance for the assessment of the acceptability of the widths and lengths of cracks in masonry walls and mortar joints and of gaps between supports and masonry walls to confirm that degradation has not invalidated the original evaluation assumptions or impacted the capability to perform the intended functions.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035
36	Structures Monitoring (19.2.2.33)	XI.S6	Continue the existing PSL Structures Monitoring AMP, including enhancements to: a) Monitor and inspect steel edge supports on masonry walls. b) Specify the use of high-strength bolt storage requirements discussed in Section 2 of the Research Council for Structural Connections publication, "Specification for Structural Joints Using High-Strength Bolts," for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. c) Inspect concrete structures for increase in porosity and permeability, loss of strength, **indications of cracking and**	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			expansion due to reaction with aggregates, and reduction in concrete anchor capacity due to local concrete degradation. d) Inspect elastomers for loss of material and **cracking**. e) Inspect stainless steel and aluminum components for pitting and crevice corrosion, and evidence of cracking due to SCC. f) Include monitoring and trending of leakage volumes and chemistry for signs of concrete or steel reinforcement degradation if active through-wall leakage or groundwater infiltration is identified. **g) Specify that all bolting is monitored for loss of material, loose bolts, missing or loose nuts, and other conditions indicative of loss of preload.** **h**) Include tactile inspection in addition to visual inspection of elastomeric elements to detect hardening. **i**) Include evidence of water in-leakage as a finding requiring further evaluation. This may include engineering evaluation, more frequent inspections, or destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels. When leakage volumes allow, assessment may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the water. **j**) **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection**	

			intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval. **k**) Require inspections of the Condensate Storage Tank (CST) and Auxiliary Feedwater (AFW) Structures and Piping Inspections in the Trenches every third refueling outage, which will ensure that these inspections are performed at least once per 5 years. **l) Include stainless steel ASME Class 1, 2, or 3 support members, welds, bolted connections, or anchorage in the engineering evaluation of acceptance criteria, expansion criteria, and examination frequency if cracking due to SCC in the uncontrolled indoor and outdoor air at PSL is detected for stainless steel mechanical or non-ASME structural components.**	
37	Inspection of Water-Control Structures Associated with Nuclear Power Plants (19.2.2.34)	XI.S7	Continue the existing PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including enhancement to: **a) Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** **b)** Revise the implementing procedure to specify the use of **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,” for	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

			structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. **c)** Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow. **d)** Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events. e) **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

38	Protective Coating Monitoring and Maintenance (19.2.2.35)	XI.S8	Continue the existing PSL Protective Coating Monitoring and Maintenance AMP, including enhancement to: a) Ensure the implementing documents reference ASTM D5163-08 and clarify the parameter monitored to include blistering, cracking, rusting or physical damage. b) Ensure any follow-up inspections are performed by individuals trained and certified in the applicable reference standards of ASTM Guide D5498-12. c) Ensure inspections include the specific inspection and documentation parameters and observation and testing methods listed in ASTM D5163-08 subparagraph 10.2.1 through 10.2.6, 10.3, and 10.4. d) Ensure implementing documents reference the guidance of Regulatory Position C4 of RG 1.54 Revision 3.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

47	Pressurizer Surge Line (19.2.2.44)	N/A – PSL Site-Specific Program	Continue the existing PSL Pressurizer Surge Line AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL1: 09/01/2035

SLRA Appendix A2, Section 19.4, Table 19-3, pages A2-65 through A2-77, A2-89 through A2-96, A2-102 through A2-106, and A2-110 as appropriate, is revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
1	Fatigue Monitoring (19.2.1.1)	X.M1	Continue the existing PSL Fatigue Monitoring AMP, including enhancement to: a) Update the plant procedure to monitor chemistry parameters that provide inputs to F_{en} factors used in CUF_{en} calculations. b) Update the plant procedure to identify and require monitoring of the 80-year projected plant transients that are utilized as inputs to CUF_{en} calculations. **These transients include:** • **The plant loading/unloading transient and the 10 percent step load increase/decrease transient, and the cold feedwater following hot standby transient.** • **The primary coolant pump starting/stopping transient.** • **The auxiliary spray at power 1, auxiliary spray at power 2, and the main spray term in cooldown transient transients.** c) Update the plant procedure to monitor and track the **following** transient**s** during the SPEO**:** • **Loss of charging** • **Loss of letdown** • **Loss of regenerative heat exchanger (short-term)** • **Loss of regenerative heat exchanger (long-term)**	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			d) Update the plant procedure to identify the corrective action options to take if component specific fatigue limits are approached.	
2	Neutron Fluence Monitoring (19.2.1.2)	X.M2	Continue the existing PSL Neutron Fluence Monitoring AMP, including enhancement to: a) Follow the related industry efforts, such as by the Pressurized Water Reactor Owners Group (PWROG), and use the information from supplemental nozzle region dosimetry measurements and reference cases or other information to provide additional justification for use of the approved WCAP-18124-NP-A or similar methodology for the determination of RPV fluence in regions above or below the active fuel region. b) Include justification that draws from Westinghouse's NRC approved RPV fluence calculation methodology and includes discussion of the neutron source, synthesis of the flux field and the order of angular quadrature (e.g., S8), etc. used in the estimates for projection of TLAA to 80 years.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

4	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (19.2.2.1)	XI.M1	Continue the existing ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
5	Water Chemistry (19.2.2.2)	XI.M2	Continue the existing PSL Water Chemistry AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

6	Reactor Head Closure Stud Bolting (19.2.2.3)	XI.M3	Continue the existing PSL Reactor Head Closure Stud Bolting AMP, including enhancement to: a) Procure reactor head closure stud materials to limit the maximum yield strength of replacement material to a measured yield strength less than 150 ksi and a maximum tensile strength of 170 ksi. b) Preclude the use of molybdenum disulfide (MoS_2) lubricant for the reactor head closure stud bolting. **c) Pursuant to 10 CFR 50.55a(z)(1), submit proposed alternatives for relief from the schedule of reactor pressure vessel (RPV) bolting examinations specified in ASME Section XI Code, Table IWB-2500-1, Category B-G-1, and IWB-2420, in order to accommodate an additional set of reactor vessel closure studs, nuts, and washers that are shared between PSL Units 1 and 2 in rotation. A proposed alternative will be submitted for approval for each subsequent ISI interval through the remainder of the SPEO.**	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
7	Boric Acid Corrosion (19.2.2.4)	XI.M10	Continue the existing PSL Boric Acid Corrosion AMP, including enhancement to: a) Include other potential means to help in the identification of borated water leakage, such as the following, in order to identify potential borated water leaks inside containment that have not been detected during walkdowns and maintenance: • Airborne radioactivity monitoring • Humidity monitoring (for trending increases in humidity levels due to unidentified RCS leakage) • Temperature monitoring (for trending increases in room/area temperatures due to unidentified RCS leakage) • Containment air cooler thermal performance monitoring (for corroborating increases in containment atmosphere	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			temperature or humidity with decreases in cooler efficiency due to boric acid plate out) b) Include a requirement in the PSL Inspection of Internal Surfaces of Miscellaneous Piping and Ducting Components AMP implementing documents to document evidence of boric acid residue (plating out of moist steam) inside containment cooler housings or similar locations such as cooling unit drain pans and to enter evidence in to the corrective action program to be evaluated under a boric acid corrosion control (BACC) program evaluation.	
8	Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (19.2.2.5)	XI.M11B	Continue the existing PSL Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components AMP, including enhancement to: a) Update the plant modification process to ensure that no additional alloy 600 material will be used in reactor coolant pressure boundary applications during the SPEO or that, if used, appropriate baseline and subsequent inspections per MRP inspection guidance will be put in place.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
9	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (19.2.2.6)	XI.M12	Continue the existing PSL Thermal Aging Embrittlement of Cast Austenitic Stainless Steel AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
10	Reactor Vessel Internals (19.2.2.7)	XIM.16A	Continue the existing PSL Reactor Vessel Internals AMP, including enhancement to: a) Implement the results of the gap analysis or implement the latest NRC-approved version of MRP-227 if it addresses 80 years of operation.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

11	Flow-Accelerated Corrosion (19.2.2.8)	XI.M17	Continue the existing PSL Flow-Accelerated Corrosion AMP, including enhancement to: a) Reassess piping systems excluded from wall thickness monitoring due to operation less than 2% of plant operating time (as allowed by NSAC-202L-R4) to ensure the exclusion remains valid and applicable for operation through 80 years. If actual wall thickness information is not available for use in this re-assessment, a representative sampling approach will be used. This re-assessment may result in additional inspections. b) Extend the erosion inspection plan for the duration of the SPEO. c) Perform opportunistic visual inspections of internal surfaces during routine maintenance activities to identify degradation. d) Revise or provide procedure(s) for measuring wall thickness due to erosion. Wall thickness should be trended to adjust the monitoring frequency and to predict the remaining service life of the component for scheduling repairs or replacements. e) Revise or provide procedure(s) to evaluate inspection results to determine if assumptions in the extent-of-condition review remain valid. If degradation is associated with infrequent operational alignments, such as surveillances or pump starts/stops, then trending activities should consider the number or duration of these occurrences. f) Revise or provide procedure(s) to perform periodic wall thickness measurements of replacement components until the effectiveness of corrective actions have been confirmed. g) Include long-term corrective actions for erosion mechanisms. The effectiveness of the corrective actions should be verified. Include periodic monitoring activities for any component replaced with an alternative material since no material is completely resistant to erosion.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

12	Bolting Integrity (19.2.2.9)	XI.M18	Continue the existing PSL Bolting Integrity AMP, including enhancement to: a) Ensure references to EPRI Reports 1015336, 1015337, and NUREG-1339 are added and guidance incorporated, as appropriate, for selection of bolting material and the use of lubricants and sealants. b) Ensure lubricants containing molybdenum disulfide (MoS_2) or other lubricants containing sulfur will not be used for pressure-retaining bolting. c) Ensure that the maximum yield strength of replacement or newly procured pressure-retaining bolting material will be limited to an actual yield strength less than 150 ksi (1,034 MPa). In addition, ensure bolting material with a yield strength greater than or equal to 150 ksi (1,034 MPa) or for which yield strength is unknown will not be used for pressure retaining bolting. For closure bolting greater than 2-inches in diameter (regardless of code classification) with actual yield strength greater than or equal to 150 ksi (1,034 MPa) or for which yield strength is unknown is used, volumetric examination will be required in accordance to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 acceptance standards, extent, and frequency of examination. d) Perform alternative means of testing and inspection for closure bolting where leakage is difficult to detect (e.g., piping systems that contain air or gas or submerged bolting). The acceptance criteria for the alternative means of testing will be no indication of leakage from the bolted connections. Required inspections will be performed on a representative sample of the population (defined as the same material and environment combination) of bolt heads and threads over each 10-year period of the SPEO. The representative sample will be 20% of the population (up to a maximum of 19 per unit). The alternative testing will be completed on a case-by-case basis through:	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			• Visual inspections of closure bolting during maintenance activities that make the bolt heads accessible and bolt threads visible; • Visual inspection for discoloration is conducted when leakage of the environment inside the piping systems would discolor the external surfaces; • Monitoring and trending of pressure decay is performed when the bolted connection is located within an isolated boundary; • Soap bubble testing, or; • Thermography testing when the temperature of the fluid is higher than ambient conditions. e) Ensure that bolted joints that are not readily visible during plant operations and refueling outages will be inspected when they are made accessible and at such intervals that would provide reasonable assurance the components' intended functions are maintained. f) Ensure that closure bolting inspections will include consideration of the guidance applicable for pressure boundary bolting in NUREG-1339 and in EPRI NP-5769. g) Project, where practical, identified degradation until the next scheduled inspection. Results will be evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the SPEO based on the projected rate of degradation. For sampling-based inspections, results will be evaluated against acceptance criteria to confirm that the sampling bases (e.g., selection, size, frequency) will maintain the components' intended functions throughout the SPEO operation based on the projected rate and extent of degradation. Adverse results will be evaluated to determine if an increased sample size or inspection frequency is required.	

			h) Evaluate leakage monitoring and sample expansion and add additional inspections if inspection results do not meet acceptance criteria as described in NUREG-2191, Chapter XI.M18, Element 7.	
13	Steam Generators (19.2.2.10)	XI.M19	Continue the existing PSL Steam Generators AMP.	No later than 6 months prior to the SPEO, i.e.: PSL2: 10/06/2042

15	Closed Treated Water Systems (19.2.2.12)	XI.M21A	Continue the existing PSL Closed Treated Water Systems AMP, including enhancement to: a) Ensure that the new visual inspection procedure(s) and/or preventive maintenance requirements evaluate the visual appearance of surfaces for evidence of loss of material on the internal surfaces exposed to the treated closed recirculating cooling water. b) Create new procedure(s) and/or preventive maintenance requirements that perform surface or volumetric examinations and evaluate the examination results for surface discontinuities indicative of cracking on the internal surfaces exposed to the treated closed recirculating cooling water. c) Ensure that visual inspections of closed treated water system components' internal surfaces are conducted whenever the system boundary is opened. When opportunistic visual inspections are conducted while the system boundary is open, they can be credited towards the representative samples for the loss of material and fouling; however, surface, or volumetric examinations must be used to confirm that there is no cracking. d) Create new procedure(s) and/or preventive maintenance requirements to ensure that the inspection requirements from NUREG-2191 are met. At a minimum, in each 10-year period during the SPEO, a representative sample of components is inspected using techniques capable of detecting loss of material, cracking, and fouling, as appropriate. The sample population is defined as follows: • 20% of the population (defined as components having the same material, water treatment program, and aging effect combination) OR; • A maximum of 19 components per population at each Unit since PSL is a two-Unit plant. e) Ensure that the new inspection and test procedure(s) and/or preventive maintenance requirements will evaluate their	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			respective results against acceptance criteria to confirm that the sampling bases (e.g., selection, size, frequency) will maintain the components' intended functions throughout the SPEO based on the projected rate and extent of degradation. Where practical, identified degradation is projected through the next scheduled inspection. f) Ensure that the new inspection and test procedure(s) and/or preventive maintenance requirements report and evaluate any detectable loss of material, cracking, or fouling associated with surfaces exposed to the treated closed recirculating cooling water per the PSL corrective action program. g) Ensure that the following additional inspections and actions are required if a post-repair/replacement inspection or subsequent inspection of surfaces exposed to the treated closed cooling water environment fails to meet acceptance criteria: • The number of increased inspections is determined in accordance with the PSL corrective action process; however, there are no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20% of each applicable material, environment, and aging effect combination is inspected, whichever is less. • If subsequent inspections do not meet acceptance criteria, an extent-of-condition and extent-of-cause analysis is conducted to determine the further extent of inspections. • Additional samples are inspected for any recurring degradation to ensure corrective actions appropriately address the associated causes. Since PSL is a two-Unit site, the additional inspections include inspections at both Units with the same material, environment, and aging effect combination. • The additional inspections are completed within the interval (e.g., refueling outage interval, 10-year inspection interval) in which the original inspection was conducted.	

16	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (19.2.2.13)	XI.M23	Continue the existing PSL Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems AMP, including enhancement to: a) Update the implementing procedure to state that, for the in-scope systems that are infrequently in service, such as the containment polar cranes, periodic inspections are performed once every refueling cycle just prior to use. b) Update the implementing procedure and inspection procedures state their respective visual inspection frequencies required by ASME B30.2-2005. According to ASME B30.2-2005, inspections are performed within the following intervals: • "Periodic" visual inspections by a designated person are required and documented yearly for normal service applications • A crane that is used in infrequent service, which has been idle for a period of one year or more, shall be inspected before being placed in service in accordance with the requirements listed in ASME B30.2-2005 paragraph 2-2.1.3 (i.e., periodic inspection) c) Update the implementing procedure to ensure that the inspection procedures for the individual load handling systems are clearly identified and referenced. d) Update the governing procedure to state that any visual indication of loss of material, deformation, or cracking, and any visual sign of loss of bolting preload for NUREG 0612 load handling systems is evaluated in accordance with ASME B30.2-2005. e) Update the governing procedure to state that repairs made to NUREG-0612 load handling systems are performed as specified in ASME B30.2-2005.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

17	Compressed Air Monitoring (19.2.2.14)	XI.M24	Continue the existing PSL Compressed Air Monitoring AMP, including enhancement to formalize compressed air monitoring activities in a new governing procedure addressing the element by element requirements presented in NUREG-2191 Section XI.M24. The following enhancements are also to be included into this procedure and other pertinent documents: a) Incorporate the air quality provisions provided in the guidance of the EPRI TR-108147 and consider the related guidance in the ASME OM-2012, Division 2, Part 28. b) Perform opportunistic visual inspections of accessible internal surfaces for signs of corrosion and abnormal corrosion products that might indicate a loss of material within the system. c) Include inspections of internal air line surfaces downstream of the instrument air dryers and emergency diesel generator air start dryers with maintenance, corrective, or other activities that involve opening of the component or system. d) Include inspection methods for the opportunistic inspections with guidance of standards or documents such as ASME OM-2012, Division 2, Part 28. e) Review air quality test results. f) Include requirements for better long term trending of negative trends, more thorough documentation, and proactive aging management. g) Include monitoring and trending guidance from ASME OM-2012, Division 2, Part 28 as applicable.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
18	Fire Protection (19.2.2.15)	XI.M26	Continue the existing PSL Fire Protection AMP, including enhancement to: a) Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are**	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			no significant indications of cracking and loss of material that could result in the loss of the fire protection capability. b) Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** c) Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable. d) Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical. e) Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation. **f) Enhance plant procedures to specify that:** • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).**	

			• **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.**	

22	Reactor Vessel Material Surveillance (19.2.2.19)	XI.M31	Continue the existing PSL Reactor Vessel Material Surveillance AMP, including an incremental adjustment to the approved capsule withdrawal schedule to withdraw and test the surveillance capsule located at 277° in accordance with the NRC approved withdrawal schedule.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

26	External Surfaces Monitoring of Mechanical Components (19.2.2.23)	XI.M36	Continue the existing PSL External Surfaces Monitoring of Mechanical Components AMP, including enhancement to: a) Indicate the material and environment combinations where external examinations could be credited to manage the aging effects of the internal surfaces of components as detailed in the PSL External Surfaces Monitoring of Mechanical Components AMP. b) Incorporate the aging management activities currently performed for external corrosion of insulated piping at PSL in the PSL External Surfaces Monitoring of Mechanical Components program procedure. c) Ensure all components made of stainless steel, aluminum, or copper alloys with greater than 15% Zn or 8% Al inspected by this program will have periodic visual or surface examinations conducted to manage cracking. d) Monitor the aging effects for elastomeric and flexible polymeric components through a combination of visual inspection and manual or physical manipulation of the material. Manual or physical manipulation of the material will include touching, pressing on, flexing, bending, or otherwise manually interacting with the material. The purpose of the manual manipulation will be to reveal changes in material properties, such as hardness, and to make the visual examination process more effective in identifying aging effects such as cracking. Flexing of polymeric components (e.g., expansion joints) exposed directly to sunlight (i.e., not located in a structure restricting access to sunlight such as manholes, enclosures, and vaults or isolated from the environment by coatings) will be conducted to detect potential reduction in impact strength as indicated by a crackling sound or surface cracks when flexed. Examples of inspection parameters for elastomers and polymers will include: • Surface cracking, crazing, scuffing, and dimensional change (e.g., “ballooning” and “necking”),	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			• Loss of thickness, • Discoloration (evidence of a potential change in material properties that could be indicative of polymeric degradation), • Exposure of internal reinforcement for reinforced elastomers, • Hardening as evidenced by a loss of suppleness during manipulation where the component and material are appropriate to manipulation. e) Specify that this program will also manage hardening or loss of strength, loss of preload for heating, ventilation, and air conditioning (HVAC) closure bolting, and blistering using visual inspections. In addition, physical manipulation will be used to manage hardening or loss of strength and reduction in impact strength. f) Specify that, when required by the ASME Code, inspections will be conducted in accordance with the applicable code requirements. And, when non-ASME Code inspections and tests are required, inspections will follow site procedures that include inspection parameters for items such as lighting, distance, offset, surface coverage, and presence of protective coatings. Inspections, except those for cracking and under insulation, will be performed every refueling outage. g) Ensure that periodic visual inspections or surface examinations will be conducted on components made of stainless steel, aluminum, or copper alloys with greater than 15% Zn or 8% Al to manage cracking every 10 years during the SPEO and other inspections will be performed at a frequency not to exceed one refueling cycle. Surfaces that are not readily visible during plant operations and refueling outages are inspected when they are made accessible and at such intervals that would provide reasonable assurance that the components' intended functions are maintained.	

			h) Specify that, when inspecting to manage cracking of a component's material, either surface examinations conducted in accordance with plant-specific procedures or ASME Code Section XI VT-1 inspections (including those inspections conducted on non-ASME Code components) are conducted on each component inspected. An inspection requires that at least 20% of the surface area of the component is inspected, unless the component is measured in linear feet, such as piping. Any combination of 1-ft length sections and components can be used to meet the recommended extent of 20% of the population of materials and environment combinations, with a maximum of 25 inspections required in each population. An inspection of a component in a more severe environment may be credited as an inspection for the specified environment and for the same material and aging effects in a less severe environment (e.g., an outdoor air environment is more severe than an indoor uncontrolled air environment which is more severe than an indoor controlled air environment, assuming that there are no borated water leaks in the indoor environments). i) Specify that, when inspecting insulated components in an outdoor environment or that may be exposed to condensation in an indoor environment, that the population and sample sizes used for inspections will be determined based on the material type (e.g., steel, stainless steel, copper alloy, aluminum) and environment (e.g., air outdoor, air accompanied by leakage) combination. A minimum of 20% of the in-scope piping length, or 20% of the surface area for components whose configuration does not conform to a 1-ft axial length determination (e.g., valve, accumulator, tank) is inspected after the insulation is removed. Alternatively, any combination of a minimum of twenty-five 1-ft axial length sections and components for each material type is inspected, with a maximum of 25 inspections required in each population. j) Ensure that visual inspections identify indirect indicators of elastomer and flexible polymer hardening or loss of strength, including the presence of surface cracking, crazing, discoloration,	

			and, for elastomers with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Visual inspections will cover 100% of accessible component surfaces. Visual inspection will identify direct indicators of loss of material due to wear to include dimension change, scuffing, and, for flexible polymeric materials with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Manual or physical manipulation can be used to augment visual inspection to confirm the absence of hardening or loss of strength for elastomers and flexible polymeric materials (e.g., heating, ventilation, and air conditioning flexible connectors) where appropriate. The sample size for manipulation will be at least 10% of available surface area. k) Indicate that the following alternatives to removing insulation after the initial inspection will be acceptable: i. Subsequent inspections may consist of examination of the exterior surface of the insulation with sufficient acuity to detect indications of damage to the jacketing or protective outer layer (if the protective outer layer is waterproof) of the insulation when the results of the initial inspections meet the following criteria: • No loss of material due to general, pitting, or crevice corrosion beyond that which could have been present during initial construction is observed during the first set of inspections, and • No evidence of SCC is observed during the first set of inspections. If: (a) the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or jacketing, (b) there is evidence of water intrusion through the insulation (e.g., water seepage through insulation seams/joints), or (c) the protective outer layer (where jacketing is not installed) is not waterproof, then periodic	

			inspections under the insulation should continue as conducted for the initial inspection. ii. Removal of tightly adhering insulation that is impermeable to moisture is not required unless there is evidence of damage to the moisture barrier. If the moisture barrier is intact, the likelihood of corrosion under insulation is low for tightly adhering insulation. Tightly adhering insulation is considered to be a separate population from the remainder of insulation installed on in-scope components. The entire population of in-scope piping that has tightly adhering insulation is visually inspected for damage to the moisture barrier with the same frequency as for other types of insulation inspections. These inspections are not credited towards the inspection quantities for other types of insulation. l) Specify that results are evaluated against acceptance criteria to confirm that the sampling bases (e.g., selection, size, frequency) will maintain the components' intended functions throughout the SPEO based on the projected rate and extent of degradation. m) Include evaluation and acceptance guidance from EPRI TR-1009743, "Aging Identification and Assessment Checklist," for visual/tactile inspections where appropriate. n) Specify that inspections to detect cracking in aluminum, stainless steel, and applicable copper alloy components will have additional inspections conducted if one of the inspections does not meet the acceptance criteria due to current or projected degradation (i.e., trending) unless the cause of the aging effect for each applicable material and environment is corrected by repair or replacement for all components constructed of the same material and exposed to the same environment. The number of increased inspections will be determined in accordance with the site's corrective action process; however, there will be no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20% of each applicable material, environment, and aging effect combination is inspected,	

			whichever is less. The additional inspections are completed within the interval in which the original inspection was conducted. If subsequent inspections do not meet acceptance criteria, an extent of condition and extent of cause analysis will be conducted to determine the further extent of inspections. Additional samples will be inspected for any recurring degradation to provide reasonable assurance that corrective actions appropriately address the associated causes. The additional inspections include populations with the same material, environment, and aging effect combinations at both Unit 1 and Unit 2. o) Require that any projected inspection results **that** will not meet acceptance criteria prior to the next scheduled inspection, will have their inspection frequencies adjusted as determined by the corrective action program.	
27	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (19.2.2.24)	XI.M38	Implement the new PSL Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
28	Lubricating Oil Analysis (19.2.2.25)	XI.M39	Continue the existing PSL Lubricating Oil Analysis AMP, including enhancement to: a) Perform sampling and testing of old oil following periodic oil changes, or on a schedule consistent with equipment manufacturer's recommendations or industry standards [e.g., ASTM D6224-02]. Plant specific OE associated with oil systems may also be used to adjust the schedule for periodic sampling and testing, when justified by prior sampling results. b) Ensure guidance indicates that phase-separated water in any amount is not acceptable. If phase-separated water is identified in the sample, then corrective actions are to be initiated to identify the source and correct the issue (e.g., repair/replace component or modify operating conditions).	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

34	10 CFR Part 50, Appendix J (19.2.2.31)	XI.S4	Continue the existing PSL 10 CFR Part 50, Appendix J AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
35	Masonry Walls (19.2.2.32)	XI.S5	Continue the existing PSL Masonry Walls AMP, including enhancement to: a) Revise the implementing procedure to monitor and inspect for gaps between the supports and masonry walls that could potentially impact the intended function or potentially invalidate its evaluation basis. b) Revise the implementing procedure to include specific monitoring, measurement, and trending of 1) widths and lengths of cracks in masonry walls and mortar joints, and 2) gaps between supports and masonry walls. c) Revise the implementing procedure to include specific guidance for the assessment of the acceptability of the widths and lengths of cracks in masonry walls and mortar joints and of gaps between supports and masonry walls to confirm that degradation has not invalidated the original evaluation assumptions or impacted the capability to perform the intended functions.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042
36	Structures Monitoring (19.2.2.33)	XI.S6	Continue the existing PSL Structures Monitoring AMP, including enhancement to: a) Monitor and inspect steel edge supports on masonry walls. b) Specify the use of high-strength bolt storage requirements discussed in Section 2 of the Research Council for Structural Connections publication, “Specification for Structural Joints Using High-Strength Bolts,” for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. c) Inspect concrete structures for increase in porosity and permeability, loss of strength, **indications of cracking and**	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			expansion due to reaction with aggregates, and reduction in concrete anchor capacity due to local concrete degradation. d) Inspect elastomers for loss of material and **cracking**. e) Inspect stainless steel and aluminum components for pitting and crevice corrosion, and evidence of cracking due to SCC. f) Include monitoring and trending of leakage volumes and chemistry for signs of concrete or steel reinforcement degradation if active through-wall leakage or groundwater infiltration is identified. **g) Specify that all bolting is monitored for loss of material, loose bolts, missing or loose nuts, and other conditions indicative of loss of preload.** **h**) Include tactile inspection in addition to visual inspection of elastomeric elements to detect hardening. **i**) Include evidence of water in-leakage as a finding requiring further evaluation. This may include engineering evaluation, more frequent inspections, or destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels. When leakage volumes allow, assessment may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the water. j) **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection**	

			intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval. **k**) Require inspections of the Condensate Storage Tank (CST) and Auxiliary Feedwater (AFW) Structures and Piping Inspections in the Trenches every third refueling outage, which will ensure that these inspections are performed at least once per 5 years. **l) Include stainless steel ASME Class 1, 2, or 3 support members, welds, bolted connections, or anchorage in the engineering evaluation of acceptance criteria, expansion criteria, and examination frequency if cracking due to SCC in the uncontrolled indoor and outdoor air at PSL is detected for stainless steel mechanical or non-ASME structural components.**	
37	Inspection of Water-Control Structures Associated with Nuclear Power Plants (19.2.2.34)	XI.S7	Continue the existing PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including enhancement to: **a) Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** **b)** Revise the implementing procedure to specify the use of **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,” for	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. **c)** Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow. **d)** Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events. **e) Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

			f) Revise the AMP and implementing procedure to more clearly reflect the following parameters monitored or inspected: • **The intake cooling water canal earthen embankments are inspected for settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, and degradation of slope protection features.** • **The intake cooling water canal erosion protection, concrete paving & grout filled fabric are inspected for, loss of material, cracking, increase in porosity and permeability, loss of strength, loss of bond, distortion, and loss of form.** • **The emergency cooling canal is inspected for loss of form, loss of material, is monitored for sedimentation, debris, and instability of slopes that may impair the function of the canals under extreme low flow conditions.** • **Diver inspections include evidence of undercutting at the UHS dam.**	
38	Protective Coating Monitoring and Maintenance (19.2.2.35)	XI.S8	Continue the existing PSL Protective Coating Monitoring and Maintenance AMP, including enhancement to: a) Ensure the implementing documents reference ASTM D5163-08 and clarify the parameter monitored to include blistering, cracking, rusting or physical damage. b) Ensure any follow-up inspections are performed by individuals trained and certified in the applicable reference standards of ASTM Guide D5498-12. c) Ensure inspections include the specific inspection and documentation parameters and observation and testing methods listed in ASTM D5163-08 subparagraph 10.2.1 through 10.2.6, 10.3, and 10.4.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

			d) Ensure implementing documents reference the guidance of Regulatory Position C4 of RG 1.54 Revision 3.	

47	Pressurizer Surge Line (19.2.2.44)	N/A – PSL Site-Specific Program	Continue the existing PSL Pressurizer Surge Line AMP.	No later than 6 months prior to the SPEO, ~~or no later than the last refueling outage prior to the SPEO~~ i.e.: PSL2: 10/06/2042

SLRA Appendix B, Section B.2.3.23, Enhancements subsection, 7. Corrective Actions affected element on pages B-195 through B-196, is revised as follows:

Element Affected	Enhancement
7. Corrective Actions	• Revise procedures to specify that inspections to detect cracking in aluminum, SS, and applicable copper alloy components will have additional inspections conducted if one of the inspections does not meet the acceptance criteria due to current or projected degradation (i.e., trending) unless the cause of the aging effect for each applicable material and environment is corrected by repair or replacement for all components constructed of the same material and exposed to the same environment. The number of increased inspections will be determined in accordance with the site's corrective action process; however, there will be no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20% of each applicable material, environment, and aging effect combination is inspected, whichever is less. The additional inspections are completed within the interval in which the original inspection was conducted. If subsequent inspections do not meet acceptance criteria, an extent of condition and extent of cause analysis will be conducted to determine the further extent of inspections. Additional samples will be inspected for any recurring degradation to provide reasonable assurance that corrective actions appropriately address the associated causes. The additional inspections include populations with the same material, environment, and aging effect combinations at both Unit 1 and Unit 2. • Revise procedures to require that any projected inspection results **that** will not meet acceptance criteria prior to the next scheduled inspection, will have their inspection frequencies adjusted as determined by the corrective action program.

SLRA Appendix B, Section B.2.3.24, Program Description subsection on page B-201, is revised as follows:

- All receivers, piping, thermowells, and valve bodies managed by this program (but not filters, silencers, or dryers) in the instrument air and bulk gas supply systems;
- All fan housings, thermowells, valve bodies, SS piping, and carbon steel piping and piping components where the internal and external portions of the pipe are exposed to an air environment in the ventilation system;

- Hardening or loss of strength for the internal surfaces of elastomeric materials:
    - All flex connections managed by this program in the containment cooling system and ventilation system;
    - All flexible hoses managed by this program in the instrument air and bulk gas supply systems.

Inspections not conducted in accordance with ASME Code Section XI requirements will be conducted in accordance with plant-specific procedures including inspection parameters such as lighting, distance, offset and surface conditions. Acceptance criteria will be such that the component will meet its intended function until the next inspection or the end of the SPEO. Qualitative acceptance criteria will be clear enough to reasonably assure a singular decision is derived based on observed conditions. Corrective actions will be performed as required based on the inspections results.

This AMP is also used to manage cracking due to stress corrosion cracking in aluminum and SS components exposed to aqueous solutions and air environments containing halides. This AMP is not used to manage components where visual inspection of internal surfaces is not possible.

The PSL Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components AMP will have new governing inspection procedures consistent with NUREG-2191, Section XI.M38.

The PSL Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components AMP implementation will be completed no later than 6 months prior to the SPEO ~~or no later than the last refueling outage prior to the SPEO~~.

NUREG-2191 Consistency

The PSL Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components AMP will be consistent without exception to the ten elements of NUREG-2191, Section XI.M38, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components”.

Exceptions to NUREG-2191

None.

Associated Enclosures:

None.

Fire Protection AMP – Acceptance Criteria

RAI B.2.3.15-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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:

During the audit of the Fire Protection program, the staff identified acceptance criteria (length and width limits) for elastomer, subliming, and silicate materials in Revision 0 of NEESL00008-REPT-051, "St. Lucie Units 1 and 2 Subsequent License Renewal Aging Management Program Basis Document – Fire Protection."

Issue:

The Subsequent License Renewal Application (SLRA) does not provide additional information related to the length and width limits. During the audit, the staff was unable to find information related to the basis for the length and width limits stated in Revision 0 of NEESL00008-REPT-051. The applicant stated during the audit that the length and width limits were in mechanical specifications and were related to Thermo-Lag®, and that they would look into the applicability to other fire barriers. The applicant did not provide additional information related to the length and width limits in SLRA Supplement 1, dated April 7, 2022 (ML22097A202).

The SLRA does not provide any discussion of the basis for the length and width limits for Thermo-Lag® or applicability of the length and width limits to other fire barriers.

Request:

Please provide the basis for the length and width limits for Thermo-Lag® and discuss the applicability of the length and width limits to other fire barriers. If procedure changes are required to clarify the acceptance criteria for Thermo-Lag® and/or other fire barriers, then please provide documentation that identifies the issue and how it will be corrected (e.g., enhancement to the Fire Protection program).

PSL Response:

SLRA Appendices A1 and A2, Section 19.4, Table 19-3, commitments 18 a) and 18 b) on pages A1-76, A1-77, A2-77, and A2-78 and Appendix B, Section B.2.3.15, the first two enhancements on pages B-122 and B-123 are revised to state that the acceptance criteria for elastomer, subliming, cementitious, and silicate fire barrier materials is no significant indications of cracking and loss of material that could result in the loss of the fire protection capability. This revision ensures that the Fire Protection AMP aligns with NUREG-2191, Chapter XI.M26, Element 6 (specifically criterion (b) and the first sentence of the Acceptance Criteria element). Accordingly, applicable procedures will be revised to remove the length and width limits for fire barriers. The SLRA revisions shown in this RAI response include changes proposed in Attachments 5 (RAI B.2.3.15-1) and 7 (RAI B.2.3.15-3) of this RAI response set.

References:

None.

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.4, Table 19-3, pages A1-76 and A1-77, is revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
18	Fire Protection (19.2.2.15)	XI.M26	Continue the existing PSL Fire Protection AMP, including enhancement to: a) Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** b) Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** c) Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable. d) Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical. e) Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation.	No later than 6 months prior to the SPEO, i.e.: PSL1: 09/01/2035

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			f) Enhance plant procedures to specify that: • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).** • **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.**	

SLRA Appendix A2, Section 19.4, Table 19-3, pages A2-77 and A2-78, is revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
18	Fire Protection (19.2.2.15)	XI.M26	Continue the existing PSL Fire Protection AMP, including enhancement to: a) Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** b) Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** c) Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable. d) Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical. e) Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation.	No later than 6 months prior to the SPEO, i.e.: PSL2: 10/06/2042

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			f) Enhance plant procedures to specify that: • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).** • **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.**	

SLRA Appendix B, Section B.2.3.15, the first two enhancements on pages B-122 and B-123, is revised as follows:

Enhancements

The PSL Fire Protection AMP will be enhanced as follows, for alignment with NUREG-2191. The enhancements are to be implemented no later than 6 months prior to entering the SPEO.

Element Affected	Enhancement
1. Scope of Program 3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.**
1. Scope of Program 3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.**
1. Scope of Program 3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable.

Element Affected	Enhancement
3. Parameters Monitored or Inspected **6. Acceptance Criteria**	**Enhance plant procedures to specify that:** • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).** • **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.**
5. Monitoring and Trending	Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical.
5. Monitoring and Trending	Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation.

Associated Enclosures:

None.

Fire Protection AMP – Fire Doors

RAI B.2.3.15-3

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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:

During the audit of the Fire Protection program, the staff identified corrective actions in Revision 5 of 1-FMM-99.12, "Unit 1 Fire Door Inspection," and Revision 6 of 2-FMM-99.12, "Unit 2 Fire Door Inspection," related to installing steel screws in holes on doors and frames.

Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ML22097A202), stated, "Fire door inspection procedures will have limitations added to the hole size for which installation of a steel screw would be acceptable for a corrective action."

Issue:

SLRA Supplement 1 appears to be making a commitment to add limitations related to installing screws in holes on doors and frames, however, the supplement did not include an associated enhancement to the Fire Protection program. In addition, SLRA Supplement 1 did not describe the limitations and discuss their basis.

Request:

Please provide a description of the limitations, and their basis, that will be added to the fire door inspection procedures. In addition, please provide documentation that identifies the issue and how it will be corrected (e.g., enhancement to the Fire Protection program).

PSL Response:

The allowance for using machine screws to be inserted into holes in a fire door comes from a functional equivalency engineering evaluation which has been uploaded to the ePortal. This equivalency evaluation was determined by fire protection engineers who are members of the Society of Fire Protection Engineers (SFPE) to be suitable for transition to NFPA 805. The determination of suitability of the equivalency evaluation was documented in PSL-FPER-11-007

(Reference 1, Attachment 1, Page 2), which was accepted by the NRC as part of the PSL NFPA 805 license amendment request (Reference 2, Section 4.2.2). This acceptance was documented in the associated safety evaluation report (Reference 3, Section 3.5.1.4). To ensure compliance with the equivalency evaluation accepted by the NRC on this matter, an additional enhancement to the Fire Protection AMP is added to SLRA Appendices A1 and A2, Section 19.4, Table 19-3 as new commitment 18 f) on pages A1-76, A1-77, A2-77, and A2-78. The new enhancement is also added to SLRA, Appendix B, Section B.2.3.15 to elements 3 and 6 on pages B-122 and B-123. The SLRA revisions shown in this RAI response include changes proposed in Attachments 5 (RAI B.2.3.15-1) and 6 (RAI B.2.3.15-2) of this RAI response set.

References:

1. PSL-FPER-11-007, NFPA 805 Transition Review of Existing Engineering Equivalency Evaluations, Revision 1.

2. St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition), dated March 22, 2013, ADAMS Accession No. ML13088A173

3. Safety Evaluation by the Office of Nuclear Reactor Regulation for Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) Amendment No. 231 to Renewed Facility Operating License No. DPR-67 and Amendment No. 181 to Renewed Facility Operating License No. NPF-16 Florida Power & Light Company St. Lucie Units Nos. 1 and 2 Docket Nos. 50-335 and 50-389, ADAMS Accession No. ML15344A346.

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.4, Table 19-3, pages A1-76 and A1-77, is revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
18	Fire Protection (19.2.2.15)	XI.M26	Continue the existing PSL Fire Protection AMP, including enhancement to: a) Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** b) Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** c) Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable. d) Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical.	No later than 6 months prior to the SPEO, i.e.: PSL1: 09/01/2035

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			e) Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation. **f) Enhance plant procedures to specify that:** • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).** • **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.**	

SLRA Appendix A2, Section 19.4, Table 19-3, pages A2-77 and A2-78, is revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
18	Fire Protection (19.2.2.15)	XI.M26	Continue the existing PSL Fire Protection AMP, including enhancement to: a) Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** b) Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.** c) Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable. d) Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical. e) Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended	No later than 6 months prior to the SPEO, i.e.: PSL2: 10/06/2042

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			functions throughout the subsequent period of extended operation based on the projected rate of degradation. f) Enhance plant procedures to specify that: • Holes in fire doors should be filled with steel sheet metal screws or through-bolts. • Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame. • Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.). • Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.	

SLRA Appendix B, Section B.2.3.15, the first two enhancements on pages B-122 and B-123, is revised as follows:

Enhancements

The PSL Fire Protection AMP will be enhanced as follows, for alignment with NUREG-2191. The enhancements are to be implemented no later than 6 months prior to entering the SPEO.

Element Affected	Enhancement
1. Scope of Program 3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Enhance plant procedures to specify that penetration seals will be inspected for indications of increased hardness and loss of strength such as cracking, seal separation from walls and components, separation of layers of material, rupture, and puncture of seals. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.**
1. Scope of Program 3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Enhance plant procedures to specify that subliming, cementitious, and silicate materials used in fireproofing and fire barriers will be inspected for loss of material**, separation**, change in material properties, and cracking/delamination. **Inspections will be acceptable if there are no significant indications of cracking and loss of material that could result in the loss of the fire protection capability.**
1. Scope of Program 3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Enhance plant procedures to specify that any loss of material (e.g., general, pitting, or crevice corrosion) to the fire damper **housings** is unacceptable.

Element Affected	Enhancement
3. Parameters Monitored or Inspected **6. Acceptance Criteria**	**Enhance plant procedures to specify that:** • **Holes in fire doors should be filled with steel sheet metal screws or through-bolts.** • **Steel conduits which have been attached to fire door frames with proper fittings are considered acceptable provided the conduit does not pass completely through the frame.** • **Flexible steel conduit (with or without water resistant coatings) may be attached to fire door frames for security contacts and latches. They must be attached with steel hardware (sheet metal screws, through bolts, etc.).** • **Holes in one surface of fire doors or frames larger than that capable of being filled by sheet-metal screws and not larger than typical conduit penetrations may be covered by a 16 gauge steel plate which overlaps the hole on all sides. The plate should be attached with steel sheet-metal screws or may be welded.**
5. Monitoring and Trending	Enhance plant procedures to require projection of identified degradation to the next scheduled inspection for all monitored fire protection SSCs, where practical.
5. Monitoring and Trending	Enhance plant procedures to require that projections are evaluated against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended functions throughout the subsequent period of extended operation based on the projected rate of degradation.

Associated Enclosures:

None.

Fire Protection AMP – Fire Barrier Penetrations

RAI B.2.3.15-4

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 April 7, 2022 (ML22097A202), revised SLRA Table 3.5.2-1 by adding calcium silicate penetrations (mechanical), thermal insulation (type III semi-hot penetrations) and citing Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Aging Management Review (AMR) item 3.2-1, 087 with Industry Standard Note I and plant-specific notes 13 and 11. The intended function for this component is cited as insulate (thermal). Plant-specific note 11 states, "Component also provides a fire barrier function [emphasis added] as evaluated in the Fire Protection Program Design Document that is physically equivalent to the structural functions managed under the associated containment structural programs or other applicable AMPs."

Issue:

SLRA Table 3.5.2-1 does not cite fire barrier as an intended function for penetrations (mechanical), thermal insulation (type III semi-hot penetrations). In addition, SLRA Section 3.5.2.1.1 does not identify the Fire Protection program for managing aging effects for containment building structures. Therefore, it is not clear whether the penetrations (mechanical), thermal insulation (type III semi-hot penetrations) have a fire barrier intended function and, consequently, do not need to be included in the Fire Protection Program.

Request:

Please clarify whether the penetrations (mechanical), thermal insulation (type III semi-hot penetrations) have a fire barrier intended function and make any necessary changes to the SLRA (e.g., revise the SLRA to reflect a fire barrier intended function, applicable aging effects associated with a fire barrier intended function, program capable of managing aging effects associated with a fire barrier intended function).

PSL Response:

Plant specific note 11 is deleted from the SLRA Table 3.5.2-1 item for penetrations (mechanical), thermal insulation (type III semi-hot penetrations) on page 3.5-80. The penetrations (mechanical), thermal insulation (type III semi-hot penetrations) do not have a fire barrier function. No change is required for SLRA Section 3.5.2.1.1.

References:

None.

Associated SLRA Revisions:

SLRA Table 3.5.2-1, page 3.5-80, is revised as follows:

Table 3.5.2-1: Containment Building Structures – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	Table 1 Item	Notes
Penetration **bellows** (mechanical)	Fire barrier Pressure boundary Structural support	Stainless steel	Air – indoor uncontrolled	Cumulative fatigue damage	TLAA - Section 4.6, "Containment Liner Plate, Metal Containments, and Penetrations Fatigue"	II.A3.C-13	3.5-1, 009	A, **11**
Penetrations (mechanical), thermal insulation (type I hot penetrations)	Insulate (thermal)	Calcium silicate	Air – indoor uncontrolled	None	None	VIII.H.S-403	3.4-1, 064	I, 5
Penetrations (mechanical), thermal insulation (type III semi-hot penetrations)	**Insulate (thermal)**	**Calcium silicate**	**Air – indoor uncontrolled**	**None**	**None**	**V.E.E-422**	**3.2-1, 087**	**I, 13**~~, 11~~
Pressure retaining bolting	Pressure boundary Structural support	Steel	Air – indoor uncontrolled	Loss of material	ASME Section XI, Subsection IWE (B.2.3.29)	II.A3.CP-148	3.5-1, 031	A
Pressure retaining bolting	Pressure boundary Structural support	Stainless steel	Air – indoor uncontrolled	Loss of material	ASME Section XI, Subsection IWE (B.2.3.29)	N/A	N/A	F, 6
Pressure retaining bolting	Pressure boundary Structural support	Stainless steel	Air – indoor uncontrolled	Loss of preload	ASME Section XI, Subsection IWE (B.2.3.29)	N/A	N/A	F, 6
Pressure retaining bolting	Pressure boundary Structural support	Steel	Air – indoor uncontrolled	Loss of preload	10 CFR Part 50, Appendix J (B.2.3.31) ASME Section XI, Subsection IWE (B.2.3.29)	II.A3.CP-150	3.5-1, 030	A

Associated Enclosures:

None.

Fire Protection AMP – Emergency Diesel Generator Buildings AMPs

RAI B.2.3.15-5

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 April 7, 2022 (ML22097A202), revised SLRA Table 3.5.2-6 by adding the Fire Protection program for managing applicable aging effects for various concrete components with a fire barrier intended function in the emergency diesel generator buildings.

Issue:

SLRA Section 3.5.2.1.6 does not include the Fire Protection program for managing aging effects for emergency diesel generator building components.

Request:

Please discuss the omission of the Fire Protection program from SLRA Section 3.5.2.1.6. Alternatively, revise SLRA Section 3.5.2.1.6 to include the Fire Protection program for managing aging effects for emergency diesel generator building components.

PSL Response:

The Fire Protection AMP manages the aging effects of emergency diesel generator building components with a fire barrier intended function. As such, the Fire Protection AMP is added to SLRA Section 3.5.2.1.6 on page 3.5-8.

References:

None.

Associated SLRA Revisions:

SLRA Section 3.5.2.1.6 on page 3.5-8, is revised as follows:

Aging Effects Requiring Management

The following aging effects associated with the emergency diesel generator buildings require management:

- Cracking
- Distortion
- Increase in porosity and permeability
- Loss of bond
- Loss of material
- Loss of preload
- Loss of strength

Aging Management Programs

The following AMPs manage the aging effects for the emergency diesel generator building components:

- **Fire Protection (B.2.3.15)**
- Structures Monitoring (B.2.3.33)

3.5.2.1.7 Fuel Handling Buildings

Materials

The materials of construction for the fuel handling building components are:

- Aluminum alloy
- Boron carbide
- Caulking and Sealants
- Concrete block (reinforced, unreinforced)
- Concrete (reinforced, unreinforced)
- Galvanized steel
- Stainless Steel
- Steel
- 6061 aluminum alloy reinforced w/ type 1 ASTM C-750 boron carbide

Environments

The fuel handling building components are exposed to the following environments:

- Air – indoor uncontrolled
- Air – outdoor
- Air with borated water leakage
- Groundwater / soil
- Soil
- Treated borated water
- Water – flowing

Associated Enclosures:

None.

Fire Protection AMP – Curbs

RAI B.2.3.15-6

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 April 7, 2022 (ML22097A202), revised SLRA Table 3.5.2-13 to change the intended function for the concrete curbs (Unit 2, 2B switchgear room) (accessible) from “fire protection” to “fire prevention.” SLRA Table 2.1-1, dated August 3, 2021 (ML21215A314 (Package)), defines fire prevention as confining or retarding a fire from spreading.

The SLRA cites GALL-SLR (Generic Aging Lessons Learned for Subsequent License Renewal) AMR (Aging Management Review) items 3.5-1, 054, 066, and 067 for managing the aging effects for this component by the Structures Monitoring program. SLRA Supplement 1 revised the discussion of these GALL-SLR AMR items in SLRA Table 3.5-1 to state the following:

- 3.5-1, 054: The Fire Protection AMP (Aging Management program) also manages cracking due to chemical reaction for structural fire barriers.

- 3.5-1, 066: The Fire Protection AMP also manages cracking and loss of material (spalling, scaling) due to corrosion of reinforcement for structural fire barriers.

- 3.5-1, 067: The Fire Protection AMP also manages cracking and loss of material (spalling, scaling) due to chemical reaction for structural fire barriers.

Issue:

It is unclear to the staff if the intent was to manage the applicable aging effects for the concrete curbs (Unit 2, 2B switchgear room) (accessible) similar to how the aging effects for the reinforced concrete structural fire barriers are managed with both the Fire Protection and Structures Monitoring programs.

Request:

Please discuss whether the intent was to manage the applicable aging effects for the concrete curbs (Unit 2, 2B switchgear room) (accessible) with both the Fire Protection and Structures Monitoring programs. If not, please discuss the basis.

PSL Response:

The concrete curbs (Unit 2, 2B switchgear room) (accessible) with a fire prevention intended function have their aging effects managed by both the Structures Monitoring AMP and the Fire Protection AMP. Fire prevention curbs are added to the discussions regarding the use of the Fire Protection AMP to manage the aging effects for Table 3.5-1 Items 054, 066, and 067 on pages 3.5-60 and 3.5-63. The Fire Protection AMP is added to the list of aging management programs for components with a fire prevention intended function in Table 3.5.2-13 on page 3.5-125. The notes for components with a fire prevention intended function in Table 3.5.2-13 on page 3.5-125 have General Note E and a new Plant Specific Note 3 added. The new Plant Specific Note 3 on page 3.5-126 indicates that “Consistent with NUREG 2191 material, environment, and aging effect, but the Fire Protection AMP, in conjunction with the Structures Monitoring AMP, is credited with managing the fire prevention function on curbs in the turbine building with a fire prevention intended function.” The Fire Protection AMP is added to the list of AMPs in Section 3.2.1.13.

References:

None.

Associated SLRA Revisions:

SLRA Section 3.5.2.1.13 on page 3.5-14, is revised as follows:

Aging Management Programs

The following AMPs manage the aging effects for the switchyard components:

- Masonry Walls (B.2.3.32)
- Structures Monitoring (B.2.3.33)

3.5.2.1.13 Turbine Buildings

Materials

The materials of construction for the turbine building components are:

- Caulking and sealants
- Concrete (reinforced)
- Concrete block (unreinforced)
- Steel

Environments

The turbine building components are exposed to the following environments:

- Air – indoor uncontrolled
- Air – outdoor
- Groundwater / soil
- Soil
- Water – flowing

Aging Effects Requiring Management

The following aging effects associated with the turbine building require management:

- Cracking
- Distortion
- Increase in porosity and permeability
- Loss of bond
- Loss of material
- Loss of preload
- Loss of sealing
- Loss of strength

Aging Management Programs

The following AMPs manage the aging effects for the turbine building components:

- **Fire Protection (B.2.3.15)**
- Masonry Walls (B.2.3.32)
- Structures Monitoring (B.2.3.33)

Associated SLRA Revisions:

SLRA Table 3.5-1, Items 3.5-1, 054, 066, and 067 on pages 3.5-60 and 3.5-63, is revised as follows:

Table 3.5-1: Summary of Aging Management Evaluations for the Containments, Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
3.5-1, 054	All groups except 6: concrete (accessible areas): all	Cracking due to expansion from reaction with aggregates	AMP XI.S6, "Structures Monitoring"	No	Consistent with NUREG-2191. Group 2 and 9 structures are not applicable to PSL. PSL concrete components did not come from a region known to yield suspected of or known to cause aggregate reactions. The Structures Monitoring (B.2.3.33) AMP includes examination for cracking, darkened crack edges, water ingress and misalignment that would be indicative of reaction with aggregates. **The Fire Protection AMP also manages cracking due to chemical reaction for structural fire barriers and fire prevention curbs.** The Structures Monitoring (B.2.3.33) AMP is credited with managing cracking of accessible concrete exposed to uncontrolled indoor air, and outdoor air environments.

Table 3.5-1: Summary of Aging Management Evaluations for the Containments, Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
3.5-1, 066	Groups 1-5, 7, 9: concrete (accessible areas): interior and above-grade exterior	Cracking, Loss of bond, Loss of material (spalling, scaling) due to corrosion of embedded steel	AMP XI.S6, "Structures Monitoring"	No	Consistent with NUREG-2191 The Structures Monitoring (B.2.3.33) AMP is credited with managing cracking, loss of bond, and loss of material for accessible plant structure concrete exposed to uncontrolled indoor air, and outdoor air environments. **The Fire Protection AMP also manages cracking and loss of material (spalling, scaling) due to corrosion of reinforcement for structural fire barriers and fire prevention curbs.**
3.5-1, 067	Groups 1-5, 7, 9: Concrete: interior; above-grade exterior, Groups 1-3, 5, 7-9 - concrete (inaccessible areas): below-grade exterior; foundation, Group 6: concrete (inaccessible areas): all	Increase in porosity and permeability, Cracking, Loss of material (spalling, scaling) due to aggressive chemical attack	AMP XI.S6, "Structures Monitoring"	No	Consistent with NUREG-2191. The Structures Monitoring (B.2.3.33) AMP is credited with managing potential increase in porosity and permeability, cracking, and loss of material due to aggressive chemical attack for inaccessible plant structure concrete in uncontrolled indoor air, outdoor air, and groundwater/soil environments. **The Fire Protection AMP also manages cracking and loss of material (spalling, scaling) due to chemical reaction for structural fire barriers and fire prevention curbs.**

SLRA Table 3.5.2-13 on page 3.5-125, is revised as follows:

Table 3.5.2-13: Turbine Buildings – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	Table 1 Item	Notes
Concrete curbs (Unit 2, 2B switchgear room) (accessible)	Fire **prevention**	Concrete (reinforced)	Air – indoor uncontrolled	Cracking	<u>**Fire Protection (B.2.3.15)**</u> Structures Monitoring (B.2.3.33)	III.A3.TP-25	3.5-1, 054	<u>E, 3</u> B
Concrete curbs (Unit 2, 2B switchgear room) (accessible)	Fire **prevention**	Concrete (reinforced)	Air – indoor uncontrolled	Cracking Loss of bond Loss of material	<u>**Fire Protection (B.2.3.15)**</u> Structures Monitoring (B.2.3.33)	III.A3.TP-26	3.5-1, 066	<u>E, 3</u> B
Concrete curbs (Unit 2, 2B switchgear room) (accessible)	Fire **prevention**	Concrete (reinforced)	Air – indoor uncontrolled	Cracking Increase in porosity and permeability Loss of material	<u>**Fire Protection (B.2.3.15)**</u> Structures Monitoring (B.2.3.33)	III.A3.TP-28	3.5-1, 067	<u>E, 3</u> B
Concrete: foundation/base mat	Structural support	Concrete (reinforced)	Soil	Cracking and distortion	Structures Monitoring (B.2.3.33)	III.A3.TP-30	3.5-1, 044	B
Concrete: foundation/base mat (inaccessible)	Structural support	Concrete (reinforced)	Groundwater/soil	Cracking Loss of bond Loss of material	Structures Monitoring (B.2.3.33)	III.A3.TP-212	3.5-1, 065	B
Concrete: foundation/base mat (inaccessible)	Structural support	Concrete (reinforced)	Groundwater/soil	Cracking Increase in porosity and permeability Loss of material	Structures Monitoring (B.2.3.33)	III.A3.TP-29	3.5-1, 067	B
Concrete: foundation/base mat (inaccessible)	Structural support	Concrete (reinforced)	Groundwater/soil	Cracking	Structures Monitoring (B.2.3.33)	III.A3.TP-204	3.5-1, 043	B, 1
Concrete: foundation/base mat (inaccessible)	Structural support	Concrete (reinforced)	Water – flowing	Increase in porosity and permeability Loss of strength	Structures Monitoring (B.2.3.33)	III.A3.TP-67	3.5-1, 047	B, 1, 2
Concrete: slabs	Structural support	Concrete (reinforced)	Air – outdoor	Cracking Increase in porosity and permeability Loss of material	Structures Monitoring (B.2.3.33)	III.A3.TP-28	3.5-1, 067	B

SLRA Table 3.5.2-13 General and Plant Specific Notes on page 3.5-126, are revised as follows:

General Notes

A. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.
B. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
E. **Consistent with NUREG 2191 material, environment, and aging effect but a different AMP is credited or NUREG 2191 identifies a plant specific AMP.**

Plant Specific Notes

1. Whereas the NUREG-2191/2192 item calls for a plant-specific AMP, PSL credits an existing AMP based on SLR-ISG-2021-03-STRUCTURES, “Updated Aging Management Criteria for Structures Portions of Subsequent License Renewal Guidance”.
2. Groundwater is considered to be water-flowing.
3. **Consistent with NUREG 2191 material, environment, and aging effect, but the Fire Protection AMP, in conjunction with the Structures Monitoring AMP, is credited with managing the fire prevention function on curbs in the turbine building with a fire prevention intended function.**

Associated Enclosures:

None.

Aging Management of Cracking Due to SCC of Containment Penetrations

RAI 3.5.2.2.1.6-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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:

SRP-SLR guidance in Section 3.5.2.2.1.6 states that the containment in-service inspection (ISI) IWE and leak rate testing may not be sufficient to detect cracks, especially for dissimilar metal welds, and additional appropriate examinations to detect stress corrosion cracking (SCC) in the listed stainless steel (SS) components and dissimilar metal welds (DMW), considering SCC susceptibility and applicable operating experience (OE) (e.g., cracking of two-ply bellows) related to detection, are recommended to address this issue.

Subsequent License Renewal Application (SLRA) Section 3.5.2.2.1.6, as amended by SLRA Supplement 1 dated April 7, 2022 (ADAMS Accession No. ML22097A202), states that the ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J AMPs will manage cracking due to SCC for the SS mechanical penetration expansion bellows and Unit 2 electrical penetration DMWs exposed to an uncontrolled indoor air environment. Any visual evidence of cracking will be evaluated for acceptability and additional surface examinations or enhanced visual examinations, if required, will be conducted in accordance with the site’s corrective action process. In addition, SLRA Supplement 1 clarified that only the Unit 2 electrical penetrations include a dissimilar metal weld inside containment.

SLRA Table 3.5.2-1, as amended by SLRA Supplement 1, lists the following AMR items subject to aging management of cracking due to SCC:

1. Electrical penetration dissimilar metal welds

2. SS fuel transfer tube, expansion bellows, and flange

3. SS mechanical penetration bellows

Issue:

The staff noticed that PSL included provisions in enhancements of “Detection of Aging Effects” and “Corrective Actions” to manage aging effects of cracking due to SCC for the Unit 2 electrical penetration dissimilar metal welds. However, it is unclear to the staff how PSL will manage aging effects of cracking due to SCC for the SS fuel transfer tube, expansion bellows, flange, and SS mechanical penetration bellows exposed to an uncontrolled indoor air environment at Units 1 and 2.

Request:

1. Explain how aging effects of cracking due to SCC will be adequately managed for the SS fuel transfer tube, expansion bellows, flange, and SS mechanical penetration bellows exposed to an uncontrolled indoor air environment at Units 1 and 2.

2. Update the SLRA based on the response above.

PSL Response:

The numbered responses below correspond to the numbered requests above.

1. As indicated in the background information above, SLRA Table 3.5.2-1, as amended by SLRA Supplement 1 dated April 7, 2022 (ADAMS Accession No. ML22097A202), states that the ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J AMPs will manage cracking due to SCC for the Unit 2 electrical penetration dissimilar metal welds and the PSL Units 1 and 2 SS fuel transfer tubes, expansion bellows, flanges and SS mechanical penetration bellows exposed to an uncontrolled indoor air environment. However, unlike the Unit 2 electrical penetration dissimilar metal welds, no provisions were included in the enhancements of the “Detection of Aging Effects” and “Corrective Actions” elements to manage the aging effect of cracking due to SCC for the SS fuel transfer tubes, expansion bellows, flanges, and SS mechanical penetration bellows. Aging management of each of these SS components is addressed below.

    a) SS fuel transfer tubes, expansion bellows, flanges

       The PSL Units 1 and 2 SS fuel transfer tubes, expansion bellows, and flanges are components of the Type V fuel transfer tube mechanical penetration assemblies. Details of the fuel transfer tube mechanical penetration assemblies are provided in Section 3.8.2.1.10 and Section 3.8.2.1.1 of the PSL Units 1 and 2 UFSARs, respectively. Each penetration assembly includes a 36-inch SS fuel transfer tube installed inside a 48-inch pipe sleeve. The 36-inch SS fuel transfer tube is fitted with a double O-ring seal SS blind flange in the refueling canal located inside containment. SS expansion bellows are provided on the 48-inch pipe sleeve to compensate for building settlement and differential seismic motion between the reactor building and the fuel handling building.

Local leak rate testing (LLRT) of the PSL Units 1 and 2 fuel transfer tube mechanical penetration assemblies is currently performed in accordance with the 10 CFR Part 50, Appendix J AMP. The LLRT is performed by pressurizing the annular space between the 36-inch SS fuel transfer tube and the 48-inch pipe sleeve with the SS expansion bellows. The LLRTs are credited with managing the aging effect of cracking due to SCC for the PSL Units 1 and 2 SS fuel transfer tube and expansion bellows and will be continued through the subsequent period of extended operation (SPEO).

As stated above, each of the PSL Units 1 and 2 36-inch SS fuel transfer tubes is fitted with a double O-ring seal SS blind flange in the refueling canal located inside containment. The blind flange is normally installed and is only removed to support plant refueling operations. The blind flange forms part of the containment pressure boundary and LLRT of the double O-ring seal is included in the 10 CFR Part 50, Appendix J AMP. The blind flange is pressure tested for integrity during the PSL Units 1 and 2 10 CFR Part 50, Appendix J integrated leak rate tests (ILRTs). The ILRTs are credited with managing the aging effect of cracking due to SCC for the PSL Units 1 and 2 SS fuel transfer tube blind flanges and will be continued through the subsequent period of extended operation (SPEO).

As stated in SLRA Section B.2.3.31, the current 10 CFR Part 50, Appendix J AMP does not require enhancement for the SPEO. In addition, enhancements of the “Detection of Aging Effects” and “Corrective Actions” elements of the 10 CFR Part 50, Appendix J AMP are not required for managing the aging effect of cracking due to SCC for the SS fuel transfer tube, expansion bellows, and flange SS components.

b) SS mechanical penetration bellows

As discussed in Section 3.8.2.1.10 and 3.8.2.1.1 of the St. Lucie Units 1 and 2 UFSARs, respectively, containment mechanical penetrations are categorized as follows:

- Type I - those which must accommodate considerable thermal movements (hot penetrations)
- Type II - those which are not required to accommodate thermal movements (low temperature penetrations)
- Type III - those which must accommodate moderate thermal movements (semi-hot penetrations)
- Type IV - containment sump recirculation suction lines
- Type V - fuel transfer tubes

Each type of penetration assembly includes a SS expansion bellow(s). As discussed in Item a) above, cracking due to SCC for the PSL Units 1 and 2 Type V fuel transfer tube penetration SS expansion bellows exposed to an uncontrolled indoor air environment is managed by the 10 CFR Part 50, Appendix J AMP. Aging management of cracking due to SCC for the Type I, II, III, and IV mechanical penetration SS bellows exposed to an uncontrolled indoor air environment is discussed below.

Type I penetration assemblies are used on the two main steam and two main feedwater lines per unit. Type I penetrations include multiple flued heads that are provided as integral parts of the process piping. For each penetration, a guard pipe, which encloses the process pipe, is welded to the flued head. A primary expansion bellows is welded to the flued head and the containment vessel penetration nozzle to accommodate thermal movements. (Note that the Unit 1 Type I penetration design includes one primary expansion bellows and the Unit 2 design includes two primary expansion bellows). The primary expansion bellows are part of the containment vessel pressure boundary and are subject to the Type B LLRT requirements of the 10 CFR 50, Appendix J AMP. Similar to the fuel transfer tube expansion bellows above, these LLRTs are credited with managing the aging effect of cracking due to SCC and as such enhancements of the “Detection of Aging Effects” and “Corrective Actions” elements of the 10 CFR Part 50, Appendix J AMP are not required for the Type I SS primary expansion bellows.

Type I, II, III, and IV penetration assemblies each include a shield building penetration sleeve and a single SS secondary expansion bellows, with the exception of the Unit 2 Type I penetrations, which include two secondary expansion bellows. The SS secondary expansion bellows are designed to withstand a design pressure of 5 psig and provide a leak-tight seal consistent with the overall allowable shield building leakage. Type I, II and III penetration assemblies include test taps which allow for leak testing of the SS secondary expansion bellows.

Management of the aging effect of cracking due to SCC for the Type I, II, III, and IV SS secondary expansion bellows exposed to an uncontrolled indoor air environment will be similar to what is proposed for the Unit 2 electrical penetration dissimilar metal welds. An enhancement of the “Detection of Aging Effects” element for the PSL ASME Section XI, Subsection IWE AMP is added to implement supplemental one time surface examinations (magnetic particle, dye penetrant) or enhanced visual examinations (EVT 1 or equivalent), performed by qualified personnel using methods capable of detecting cracking, of a representative sample of SS secondary expansion bellows. NUREG-2191, Section XI.S1, ASME Section XI, Subsection IWE, Element 4, states that where feasible, appropriate Appendix J leak rate tests (NUREG-2191, Section XI.S4) capable of detecting cracking may be performed or credited in lieu of the supplemental surface examinations. Therefore, leak rate testing of the SS secondary expansion bellows using installed test fittings can be performed as an alternative test method for detecting cracking of the SS secondary expansion bellows as similar testing

is currently performed for Type I penetration SS primary bellows and Type V fuel transfer tube SS expansion bellows. The leak rate testing will be performed using methods consistent with the 10 CFR 50, Appendix J AMP at the design pressure of 5 psig with an acceptance criteria of zero leakage.

The total population of secondary expansion bellows is 49 for PSL Unit 1 and 62 for PSL Unit 2. A total of 10 and 13 secondary expansion bellows constitute representative samples for these populations for PSL Units 1 and 2, respectively. An enhancement to the PSL ASME Section XI, Subsection IWE AMP "Detection of Aging Effects" element will be added to perform these supplemental one-time inspections for PSL Units 1 and 2.

In addition, an enhancement related to the "Corrective Actions" element upon identification of cracking due to SCC as a result of these supplemental one-time inspections for the secondary expansion bellows will be specified for the PSL ASME Section XI, Subsection IWE AMP. If needed, follow-on inspections will include surface examinations (magnetic particle, dye penetrant), enhanced visual examinations (EVT-1 or equivalent), or leak rate testing of the SS secondary expansion bellows. The frequencies of inspections will be consistent with the approved IWE intervals for each unit.

2. SLRA Sections 3.5.2.2.1.6, Table 19-3 (Appendices A1 and A2), and Section B.2.3.29 are updated to reflect these changes. Note that an additional change is made to SLRA Table 3.5.2-1 to correct the NUREG-2191 Item number for a Unit 2 electrical penetration dissimilar metal weld entry.

References:

None.

Associated SLRA Revisions:

SLRA Section 3.5.2.2.1.6, page 3.5-25, is revised as follows:

3.5.2.2.1.6 Cracking Due to Stress Corrosion Cracking

Stress corrosion cracking (SCC) of stainless steel (SS) penetration sleeves, penetration bellows, vent line bellows, suppression chamber shell (interior surface), and dissimilar metal welds could occur in PWR and/or BWR containments. The existing program relies on ASME Code Section XI, Subsection IWE and 10 CFR Part 50, Appendix J, to manage this aging effect. Further evaluation, including consideration of SCC susceptibility and applicable operating experience (OE) related to detection, is recommended of additional appropriate examinations/evaluations implemented to detect this aging effect for these SS components and dissimilar metal welds.

The PSL Units 1 and 2 ~~S~~stainless steel fuel transfer tubes, expansion bellows, flanges, and mechanical penetration expansion bellows, and PSL Unit 2 electrical penetration dissimilar metal welds are exposed to an uncontrolled indoor air environment at PSL. A review of PSL OE confirms halides are potentially present in the indoor environments at PSL. Additionally, these indoor components, particularly those in standby or periodically operated systems, could conservatively see an accumulation of contaminants. As such, the PSL Units 1 and 2 stainless steel fuel transfer tubes, expansion bellows, flanges, and mechanical penetration expansion bellows, and PSL Unit 2 electrical penetration dissimilar metal welds exposed to uncontrolled indoor air ~~in the containment buildings~~ are susceptible to SCC and require management via an appropriate program.

Consistent with the recommendation of NUREG-2191, cracking of these components will be managed by the ASME Section XI, Subsection IWE (B.2.3.29) AMP and 10 CFR Part 50, Appendix J (B.2.3.31) AMP. The ASME Section XI, Subsection IWE AMP provides for the management of aging effects through supplemental surface examinations (magnetic particle, dye penetrant) or enhanced visual examinations (EVT-1 or equivalent), performed by qualified personnel using methods capable of detecting cracking.

NUREG-2191, Section XI.S1, ASME Section XI, Subsection IWE, Element 4, states that where feasible, appropriate Appendix J leak rate tests (NUREG-2191 AMP XI.S4) capable of detecting cracking may be performed or credited in lieu of the supplemental surface examinations. Therefore, leak rate testing of the SS mechanical secondary expansion bellows using installed test fittings can be used as an alternative test method for detecting cracking of the SS secondary expansion bellows as similar testing is currently performed for Type I penetration SS primary bellows and Type V fuel transfer tube SS expansion bellows. The leak rate testing will be performed using methods consistent with the 10 CFR 50, Appendix J AMP at the design pressure of 5 psig with an acceptance criteria of zero

leakage. Any ~~visual~~ evidence of cracking will be evaluated for acceptability and additional supplemental surface examinations, ~~or~~ enhanced visual examinations, or Appendix J leak rate tests, if required, will be conducted in accordance with the site's corrective action process. The 10 CFR Part 50, Appendix J AMP is a performance monitoring program that monitors the leakage rates through the containment system, its vessel, associated welds, penetrations, isolation valves, fittings, and other access openings to detect degradation of the containment pressure boundary. Adverse conditions will be documented in accordance with the 10 CFR Part 50, Appendix B Corrective Action Program.

SLRA Table 3.5-1, page 3.5-51, is revised as follows:

Table 3.5-1: Summary of Aging Management Evaluations for the Containments, Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Program / TLAA	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)	Consistent with NUREG-2191. The ASME Section XI, Subsection IWE (B.2.3.29) AMP and 10 CFR Part 50, Appendix J (B.2.3.31) AMP manage cracking of ~~Type III (semi-hot) stainless steel and dissimilar metal weld~~ **PSL Units 1 and 2 stainless steel fuel transfer tubes, expansion bellows, flanges, and mechanical** penetration **expansion** bellows~~assemblies~~, **and PSL Unit 2 electrical penetration dissimilar metal welds** exposed to an uncontrolled indoor air environment. Further evaluation is documented in Section 3.5.2.2.1.6.

SLRA Table 3.5.2-1, page 3.5-78, is revised as follows:

Table 3.5.2-1: Containment Building Structures – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	Table 1 Item	Notes
Penetrations (electrical), Unit 2	Fire barrier Pressure boundary Structural support	Dissimilar metal welds	Air – indoor uncontrolled	Cracking	10 CFR Part 50, Appendix J (B.2.3.31) ASME Section XI, Subsection IWE (B.2.3.29)	**II.~~B4~~A3.CP-38**	3.5-1, **010**	A, **11**

SLRA Section A-1, Section 19.4, commitment No. 32 portion of Table 19-3 beginning on page A1-99, is revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
32	ASME Section XI, Subsection IWE (19.2.2.29)	XI.S1	Continue the existing PSL ASME Section XI, Subsection IWE AMP, including enhancement to: a) Augment existing procedures to **reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, update procedures to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** b) Augment existing procedures to specify the **use of** preventive actions for storage, lubricant s**election**, 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," **for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts**. **c**) Augment existing procedures to implement a one-time supplemental volumetric **examination** of **containment vessel** shell surfaces **for both units** that samples one-foot square locations **including both randomly-selected and focused areas most likely to experience degradation based on OE and/or other relevant considerations such as environment** if triggered by plant-specific OE after the date of issuance of the first renewed license **in either** unit. This sampling is conducted to demonstrate, with 95% confidence, that 95% of the accessible portion of the **containment vessel** shell is not experiencing greater than 10% wall loss.	**Program one-time inspections for cracking due to SCC begin 5 years before the SPEO. Inspections that are to be completed prior to the SPEO are completed 6 months prior to the SPEO or no later than the last refueling outage prior to the SPEO.** **Program and SLR enhancements are implemented 6 months prior to the SPEO, i.e.:** PSL1: 09/01/2035

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			d) Augment existing procedures to implement supplemental one time surface examinations (magnetic particle, dye penetrant) or enhanced visual examinations (EVT 1 or equivalent), performed by qualified personnel using methods capable of detecting cracking, comprising a representative sample (ten) of the PSL Unit 1 mechanical penetration stainless steel secondary expansion bellows. Leak rate testing of the stainless steel secondary expansion bellows using the installed test fittings can be used as an alternative test method for detecting cracking. The leak rate testing will be performed using methods consistent with the 10 CFR 50, Appendix J AMP at the design pressure of 5 psig with an acceptance criteria of zero leakage. These inspections are intended to confirm the absence of SCC aging effects. If SCC is identified as a result of these inspections, the appropriate corrective action will be taken and additional surface examinations (magnetic particle, dye penetrant), enhanced visual examinations (EVT 1 or equivalent), or leak rate testing will be conducted in accordance with the site's corrective action process. This will include testing or inspection of additional PSL Unit 1 stainless steel secondary expansion bellows until cracking is no longer detected. Periodic inspection of PSL Unit 1 stainless steel secondary expansion bellows for cracking will be added to the PSL ASME Section XI, Subsection IWE AMP if necessary, depending on the inspection results. Frequency of inspections will be consistent with the approved IWE inspection interval.	

SLRA Appendix A2, Section 19.4, commitment No. 32 portion of Table 19-3 beginning on page A2-99, is revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
32	ASME Section XI, Subsection IWE (19.2.2.29)	XI.S1	Continue the existing PSL ASME Section XI, Subsection IWE AMP, including enhancement to: a) Augment existing procedures to **reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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**. **Additionally, update procedures to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** b) Augment existing procedures to specify the **use of** preventive actions for storage, lubricant s**election**, 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,” **for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts**. **c**) Augment existing procedures to implement supplemental one-time surface **examinations (magnetic particle, dye penetrant)** or enhanced visual examinations **(EVT-1 or equivalent)**, performed by qualified personnel using methods capable of detecting cracking, comprising a representative sample (**ten**) of the **PSL Unit 2 electrical** penetration dissimilar metal welds. These inspections are intended to confirm the absence of SCC aging effects. If SCC is identified as a result of the**se** inspections, additional **surface examinations (magnetic particle, dye penetrant) or enhanced visual examinations (EVT-1 or equivalent)** will be conducted in accordance with the site’s corrective action process. This will include additional **PSL Unit 2 electrical** penetration dissimilar metal welds until cracking is no longer detected. Periodic inspection of **PSL Unit 2 electrical**	**Program one-time inspections for cracking due to SCC begin 5 years before the SPEO. Inspections that are to be completed prior to the SPEO are completed 6 months prior to the SPEO or no later than the last refueling outage prior to the SPEO.** **Program and SLR enhancements are implemented 6 months prior to the SPEO, i.e.:** PSL2: 10/06/2042

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			penetration dissimilar metal welds for cracking will be added to the PSL ASME Section XI, Subsection IWE AMP if necessary, depending on the inspection results. **Frequency of inspections will be consistent with the approved IWE inspection interval.** **d)** Augment existing procedures to implement a one-time supplemental volumetric **examination** of **containment** vessel shell surfaces **for both units** that samples **one-foot square** locations **including both randomly-selected and focused areas most likely to experience degradation based on OE and/or other relevant considerations such as environment** if triggered by plant-specific OE after the date of issuance of the first renewed license **in either** unit. This sampling is conducted to demonstrate, with 95% confidence, that 95% of the accessible portion of the **containment vessel** shell is not experiencing greater than 10% wall loss. **e) Augment existing procedures to implement supplemental one time surface examinations (magnetic particle, dye penetrant) or enhanced visual examinations (EVT 1 or equivalent), performed by qualified personnel using methods capable of detecting cracking, comprising a representative sample (thirteen) of the PSL Unit 2 mechanical penetration stainless steel secondary expansion bellows. These inspections are intended to confirm the absence of SCC aging effects. If SCC is identified as a result of these inspections, additional surface examinations (magnetic particle, dye penetrant), enhanced visual examinations (EVT 1 or equivalent), or leak rate testing will be conducted in accordance with the site's corrective action process. This will include additional PSL Unit 2 stainless steel secondary expansion bellows until cracking is no longer detected. Periodic inspection of PSL Unit 2 stainless steel secondary expansion bellows for**	

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			cracking will be added to the PSL ASME Section XI, Subsection IWE AMP if necessary, depending on the inspection results. Frequency of inspections will be consistent with the approved IWE inspection interval.	

SLRA Appendix B, Section B.2.3.29, continuing paragraph at top of page B-228, is revised as follows:

> addressed in the Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis TLAA for SLR (Section 4.6). This AMP will also include supplemental one-time inspections **beginning** 5 years **before** the SPEO for a representative sample of **PSL Units 1 and 2 stainless steel mechanical penetration secondary expansion bellows and** **PSL Unit 2 electrical** penetration dissimilar metal welds that may be susceptible to SCC.

SLRA Appendix B, Section B.2.3.29, Enhancements Table beginning on page B-228 is revised as follows:

Element Affected	Enhancement
2. Preventive Actions	Procedures will be revised to **reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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**. **Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.**
2. Preventive Actions	Procedures will be revised to specify the **use of** preventive actions for storage, lubricant s**election**, 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,” **for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts**.
4. Detection of Aging Effects	Procedures will be revised to implement supplemental one-time surface **examinations (magnetic particle, dye penetrant)** or enhanced visual examinations **(EVT-1 or equivalent)**, performed by qualified personnel using methods capable of detecting cracking, comprising a representative sample (**ten**) of the **PSL Unit 2 electrical** penetration dissimilar metal welds. These inspections are intended to confirm the absence of SCC aging effects.
4. Detection of Aging Effects	Procedures will be revised to specify a one-time volumetric examination of **containment vessel** shell surfaces that are inaccessible from one side if triggered by plant-specific OE identified after the date of issuance of the first renewed license **in either** unit. If triggered, this inspection will be performed **for both units** by sampling **one-foot square** locations **including both randomly-selected and focused areas most likely to experience degradation based on OE and/or other relevant considerations such as environment**. The trigger for this one-time examination is site-specific occurrence or recurrence of **containment vessel** shell corrosion (base metal material loss exceeding 10% of nominal plate thickness) that is determined to originate from inaccessible **areas**. Guidance provided in EPRI TR-107514 will be considered when establishing a sampling plan. This sampling is conducted to demonstrate, with 95% confidence, that 95% of the accessible portion of the **containment vessel** shell is not experiencing greater than 10% wall loss.

Element Affected	Enhancement
4. Detection of Aging Effects	**Procedures will be revised to implement supplemental one time surface examinations (magnetic particle, dye penetrant), enhanced visual examinations (EVT 1 or equivalent), or leak rate testing performed by qualified personnel using methods capable of detecting cracking, comprising a representative sample (ten for PSL Unit 1 and thirteen for PSL Unit 2) of mechanical penetration stainless steel secondary expansion bellows. These inspections are intended to confirm the absence of SCC aging effects.**
7. Corrective Actions	If SCC is identified as a result of the supplemental one-time inspections, additional **surface examinations (magnetic particle, dye penetrant)**, ~~or~~ **enhanced visual examinations (EVT-1 or equivalent)**, **or leak rate tests,** will be conducted, **as appropriate,** in accordance with the site's corrective action process. This will include additional **PSL Units 1 and 2 stainless steel secondary expansion bellows and PSL Unit 2 electrical** penetration dissimilar metal welds until cracking is no longer detected. Periodic inspection of **PSL Units 1 and 2 stainless steel secondary expansion bellows and PSL Unit 2 electrical** penetration dissimilar metal welds for cracking will be added to the PSL ASME Section XI, Subsection IWE AMP if necessary, depending on the inspection results. **Frequency of inspections will be consistent with the approved IWE inspection interval.**

Associated Enclosures:

None.

Structures Monitoring AMP – Management of Aging Effects of Water-Control Structures Exposed to an Aggressive Groundwater/Soil Environment

RAI B.2.3.33-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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) Section B.2.3.33, “Structures Monitoring,” stated that that groundwater/soil at St. Lucie Nuclear Plant (PSL) is judged to be aggressive (chlorides > 500 ppm).

SLRA Section B.2.3.34, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” states that the PSL Inspection of Water-Control Structures Associated with Nuclear Power Plant Aging Management Program (AMP) with enhancements will be consistent with exception to the 10 elements of NUREG-2191, Section XI.S7, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.”

For the “detection of aging effects” program element, the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report AMP states that for plants with aggressive groundwater or soil (pH < 5.5, chlorides > 500 ppm, or sulfates > 1,500 ppm) and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the subsequent period of extended operation (SPEO). The GALL-SLR Report also provides examples of what actions may be implemented as part of the plant-specific AMP. The Standard Review Plan for Subsequent License Renewal (SRP-SLR) Appendix A provides the staff positions and guidance for one acceptable way to implement the plant-specific AMP and/or program actions to demonstrate that the effects of aging for structures and components will be adequately managed.

Issue:

SLRA Section B.2.3.33, as amended by SLRA Supplement 1 dated April 7, 2022 (ADAMS Accession No. ML 22097A202), provides a site-specific enhancement to the “detection of aging effects” program element of the Structures Monitoring program for revising applicable

procedures to conduct a baseline visual inspection, pH analysis, a chloride concentration test, and evaluation to address the potential degradation of concrete due to exposure of aggressive chemical attack in groundwater/soil or leaching and carbonation in the water-flowing environment, and to perform the focused periodic inspections and evaluation updates on an interval not to exceed 5 years throughout the SPEO to ensure that aging effects of inaccessible concrete are adequately managed.

However, the Inspection of Water-Control Structures Associated with Nuclear Power Plants program does not provide an enhancement to the “detection of aging effects” program element similar to the Structures Monitoring program. It is unclear how the Inspection of Water-Control Structures Associated with Nuclear Power Plants program will adequately manage the aging effects of water-control structures exposed to an aggressive groundwater/soil environment.

Request:

Clarify how the Inspection of Water-Control Structures Associated with Nuclear Power Plants program will adequately manage the aging effects of water-control structures exposed to an aggressive groundwater/soil environment and provide an enhancement if necessary.

PSL Response:

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is implemented by the same implementing procedure that also implements the PSL Structures Monitoring AMP. As a result, any enhancements to the PSL Structures Monitoring AMP related to the water control structures will also be applicable to the PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including the enhancements regarding management of the aging effects of water-control structures exposed to an aggressive groundwater/soil and water flowing environment.

In order to ensure that the necessary enhancements are applied appropriately to the water-control structures, the PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is updated to include the same site-specific enhancement as the PSL Structures Monitoring AMP related to aging management of structures exposed to aggressive groundwater/soil and water flowing environment. The site-specific enhancement includes the following:

a. A baseline inspection of inaccessible concrete will be conducted prior to the SPEO.

   i. The baseline inspection locations will consider site-specific OE. OE considered will include known degradation due to chlorides in ambient air and the potential for further degradation due to the aggressive groundwater as well as whether leaching and carbonation is occurring in the water-flowing environment.

   ii. The baseline inspection will include excavation, visual inspection, and physical inspection of the inaccessible concrete through pH analysis and a chloride concentration test at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection of these two locations is a representative sample since the baseline sample is 20 percent of the population of structures most likely to experience degradation associated with groundwater.

b. A baseline evaluation will be performed prior to the SPEO.

    i. The baseline evaluation will consider the baseline inspection results to determine the additional actions (if any) that are warranted. Any observed degradation will be entered into the corrective action program. The baseline inspection results will be evaluated based on acceptance criteria provided in ACI 349.3R and will also consider the correlation between the chloride ion concentration necessary to induce corrosion and alkalinity level of the concrete (Reference 1). The highly alkaline environment of concrete protects the steel reinforcement from corrosion (Reference 2). Additional actions will be based on the baseline inspection results and corrective action program and may include: enhanced inspection techniques and/or frequency, destructive testing, and focused inspections of representative accessible concrete (leading indicator) or below grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil (or to leaching and carbonation in water-flowing if determined to impact intended function).

    ii. The baseline inspection and evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years) for the SPEO. The minimum inspection interval of every 5 years will be reduced if the observed degradation could adversely affect structural function prior to the next scheduled inspection. The subsequent inspection sample size will include the two baseline inspection locations and may be expanded based on any corrective action program results (as applicable to degradation) to include additional locations. Additional locations may be based on the aging effect (cracking, loss of material (spalling, scaling), increase in porosity and permeability, loss of strength, loss of bond), location (close to the coastline/intake or main plant area), existing technical information, structure design, material of construction (concrete), environment, operating conditions, and OE. For example, if degradation outside of the acceptance criteria is identified at the coastline/intake location, an additional location at the coastline should be inspected.

c. Periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years) will be performed.

    i. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.

    ii. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.

d. Periodic evaluation updates will be performed (not to exceed 5 years) throughout the SPEO.

    i. Updates will be based on OE and focused periodic inspection results (and/or opportunistic inspection results, if applicable) during the interval.

ii. The periodic evaluation results will update subsequent inspection requirements and inspection intervals (not to exceed 5 years) for the SPEO as required.

Accordingly, SLRA Sections 19.2.2.34 in Appendices A1 and A2 and B.2.3.34, and Table 19-3 in Appendices A1 and A2 are revised to include this additional detail.

References:

1. NUREG/CR-5466 (NISTIR 89-4086), Service Life of Concrete, Published November 1989 (ADAMS Accession No. ML061430380)

2. ACI 222.3R, Design and Construction Practices to Mitigate Corrosion of Reinforcement in Concrete Structures

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.2.2.34, pages A1-35 and A1-36, is revised as follows:

19.2.2.34 Inspection of Water Control Structures Associated with Nuclear Power Plants

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is an existing AMP that is currently implemented as part of the PSL Structures Monitoring Program. The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP was evaluated as a portion of the PSL Systems and Structures Monitoring AMP in the initial license renewal application. The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is evaluated separately in the subsequent license renewal application and is compared to the NUREG-2191, Section XI.S7 program. This condition monitoring AMP addresses age-related deterioration, degradation due to environmental conditions, and the effects of natural phenomena that may affect water-control structures.

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP consists of inspection and surveillance of control structures for raw water. The structures within the scope of the PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP include the intake cooling water canal (the portion between the emergency cooling canal and the intake structure), emergency cooling canal, Unit 1 and Unit 2 intake structures, and ultimate heat sink dam. The program also includes structural steel and structural bolting associated with water-control structures. Parameters monitored are in accordance with Section C.2 of RG 1.127 and quantitative measurements are recorded for findings that exceed the acceptance criteria for applicable parameters monitored or inspected. Inspections occur at least once every 5 years. Evaluation of ground water chemistry is performed under the scope of the PSL Structures Monitoring AMP. **Due to aggressive groundwater chemistry (chlorides > 500 parts per million), the AMP includes a site-specific enhancement to conduct a baseline visual inspection, pH analysis, a chloride concentration test, and evaluation to address the potential degradation of concrete due to exposure of aggressive**

chemical attack in groundwater/soil or leaching and carbonation in the water-flowing environment. The baseline evaluation will consider site-specific OE and the baseline inspection results and will determine the additional actions (if any) that are warranted. Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.

SLRA Appendix A1, Section 19.4, commitment No. 37 portion of Table 19-3 on pages A1-104 and A1-105, as amended by Attachment 5 to this letter, is further revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
37	Inspection of Water-Control Structures Associated with Nuclear Power Plants (19.2.2.34)	XI.S7	Continue the existing PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including enhancement to: **a) Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** **b**) Revise the implementing procedure to specify the use of **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," for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. **c**) Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow.	No later than 6 months prior to the SPEO, i.e.: PSL1: 09/01/2035

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			d) Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events. **e) Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

SLRA Appendix A2, Section 19.2.2.34, page A2-35, is revised as follows:

19.2.2.34 Inspection of Water Control Structures Associated with Nuclear Power Plants

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is an existing AMP that is currently implemented as part of the PSL Structures Monitoring Program. The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP was evaluated as a portion of the PSL Systems and Structures Monitoring AMP in the initial license renewal application. The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is evaluated separately in the subsequent license renewal application and it is compared to the NUREG-2191, Section XI.S7 program. This condition monitoring AMP addresses age-related deterioration, degradation due to environmental conditions, and the effects of natural phenomena that may affect water-control structures.

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP consists of inspection and surveillance of control structures for raw water. The structures within the scope of the PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP include the ICW canal (the portion between the emergency cooling canal and the intake structure), emergency cooling canal, Unit 2 intake structure, and ultimate heat sink dam. The AMP also includes structural steel and structural bolting associated with water-control structures. Parameters monitored are in accordance with Section C.2 of RG 1.127 and quantitative measurements are recorded for findings that exceed the acceptance criteria for applicable parameters monitored or inspected. Inspections occur at least once every 5 years. Evaluation of ground water chemistry is performed under the scope of the PSL Structures Monitoring AMP. **Due to aggressive groundwater chemistry (chlorides > 500 parts per million), the AMP includes a site-specific enhancement to conduct a baseline visual inspection, pH analysis, a chloride concentration test, and evaluation to address the potential degradation of concrete due to exposure of aggressive chemical attack in groundwater/soil or leaching and carbonation in the water-flowing environment. The baseline evaluation will consider site-specific OE and the baseline inspection results and will determine the additional actions (if any) that are warranted. Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**

SLRA Appendix A2, Section 19.4, commitment No. 37 portion of Table 19-3 on pages A2-105, as amended by Attachment 5 to this letter, is further revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
37	Inspection of Water-Control Structures Associated with Nuclear Power Plants (19.2.2.34)	XI.S7	Continue the existing PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including enhancement to: **a) Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** **b**) Revise the implementing procedure to specify the use of **preventive actions for storage, lubricant selection, and bolting and coating material selection** discussed in Section 2 of the Research Council for Structural Connections publication, “Specification for Structural Joints Using High-Strength Bolts,” for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. **c**) Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow.	No later than 6 months prior to the SPEO, i.e.: PSL2: 10/06/2042

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			d) Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events. **e) Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

SLRA Appendix B, Section B.2.3.34, page B-253, is revised as follows:

B.2.3.34 Inspection of Water-Control Structures Associated with Nuclear Power Plants

Program Description

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is an existing AMP that was evaluated as a portion of the PSL Systems and Structures Monitoring AMP in the initial license renewal application. The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is evaluated separately in the subsequent license renewal application and it is compared to the NUREG 2191, Section XI.S7 program.

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP is a condition monitoring AMP that addresses age-related deterioration, degradation due to environmental conditions, and the effects of natural phenomena that may affect water-control structures. The program is implemented in association with the existing implementing procedure for the Structures Monitoring (Section B.2.3.33) AMP. The structures within the scope of the PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP include the intake cooling water canal (the portion between the emergency cooling canal and the intake structure), emergency cooling canal, Unit 1 and Unit 2 intake structures, and ultimate heat sink dam. Structural steel and bolting associated with these structures is within the scope of the program inspections. Flood protection features are managed by the Structures Monitoring (Section B.2.3.33) AMP. The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP performs periodic monitoring of water-control structures at least every five years so that the consequences of age-related deterioration and degradation can be prevented or mitigated in a timely manner. Submerged concrete structures are inspected when dewatered or using video divers with videos reviewed by qualified Responsible Engineer or evaluator. Areas covered by silt, vegetation, or marine growth are not considered inaccessible and are cleaned and inspected in accordance with the standard inspection frequency. ~~Inspection of inaccessible areas is performed under the scope of the PSL Structures Monitoring (Section B.2.3.33) AMP.~~

The groundwater/soil at PSL is judged to be aggressive (chlorides > 500 ppm). Since the chloride levels for seawater are much greater than 500 ppm, there is reasonable certainty that any groundwater/soil chemistry tests will consistently result in chloride level readings that are greater than 500 ppm, which indicates an aggressive groundwater/soil classification, and periodic sampling and testing of groundwater is not necessary. Due to the presence of high chloride levels in the groundwater, a site-specific enhancement to manage the concrete aging during the SPEO will include a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO. The inspection will include a location close to the coastline/intake and a location in the main plant area for comparison and consider site-specific OE. The baseline inspection results will be used to conduct a baseline evaluation that will determine if additional actions are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Functionality throughout the period of the next scheduled inspection will be monitored. Periodic inspections (focused) and evaluation updates (not

to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.

SLRA Appendix B, Section B.2.3.34, page B-254, is revised as follows:

Exceptions to NUREG-2191

Each PSL Unit 1 and Unit 2 intake structure includes four (4) intake wells. The four intake wells for each unit are dewatered and inspected on a rotating basis. All four intake wells for each unit (100%) are inspected within a 6-year interval. This is an exception to the NUREG-2191 guidance that 100% of submerged structural elements should be inspected every 5 years.

Enhancements

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP will be enhanced as follows, for alignment with NUREG-2191. The enhancements are to be implemented no later than 6 months prior to entering the SPEO.

Element Affected	Enhancement
1. Scope	**Inaccessible concrete/foundations exposed to groundwater/soil and water-flowing.**
2. Preventive Actions	**Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.**
2. Preventive Actions	Revise the implementing procedure to specify the use of **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,” for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts.
3. Parameters Monitored or Inspected	**Monitoring of the condition of inaccessible concrete, including pH and chloride concentration, of concrete exposed to groundwater/soil and water-flowing environment for evidence of aggressive chemical attack or leaching and carbonation.**

Element Affected	Enhancement
4. Detection of Aging Effects	Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow.
4. Detection of Aging Effects	Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events.
4. Detection of Aging Effects	**Guidance on baseline inspection with excavation, visual inspection, and physical inspection of the inaccessible concrete through pH analysis and a chloride concentration test of concrete exposed to groundwater/soil and water-flowing at a location near the coastline and a location in the main plant area for comparison prior to the SPEO. Include periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years). Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.** **Degradation of accessible areas due to aging mechanisms such as chemical attack, leaching of calcium hydroxide, and carbonation can be used as an indicator for reinforced concrete conditions in inaccessible areas. Carbonation test results from accessible areas may be used as an indicator to determine the extent of condition of inaccessible reinforced concrete that is exposed to similar environmental conditions and of similar age.**
5. Monitoring and Trending	**Guidance for the evaluation of the baseline inspection results and related OE, with concrete exposed to ambient air and to groundwater/soil, for concrete susceptible to aging effects related to an aggressive environment prior to the SPEO to determine subsequent inspection/evaluation requirements and intervals (not to exceed 5 years), with periodic updates based on periodic inspections and OE.** **Guidance for the evaluation of baseline inspection results and related OE of concrete exposed to water-flowing for evidence of leaching of calcium hydroxide and carbonation will be developed prior to the SPEO.**

Element Affected	Enhancement
	The baseline evaluation will determine whether leaching and carbonation are occurring or causing adverse effects. Subsequent inspection/evaluation requirements and intervals (not to exceed 5 years), with periodic updates based on periodic inspections and OE, will be developed if leaching or carbonation is occurring in accessible or inaccessible areas that impacts intended function. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.
6. Acceptance Criteria	Acceptance criteria for concrete inspections will be consistent with ACI 349.3R and consider the correlation between the chloride ion concentration within the concrete cover necessary to induce corrosion and alkalinity level of the concrete covering the rebar for inaccessible concrete exposed to groundwater/soil and water-flowing environments.

Associated Enclosures:

None.

Structures Monitoring AMP – Acceptability of Inaccessible Areas Based on Condition in Accessible Areas

RAI B.2.3.33-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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) Section B.2.3.33, "Structures Monitoring," states that the PSL Structures Monitoring AMP with enhancements will be consistent with exception to the 10 elements of NUREG-2191, Section XI.S6, "Structures Monitoring."

For the "detection of aging effects" program element, the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report Aging Management Program (AMP) states that 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 ADM-17.32 "Structures Monitoring Program", it was noted that, in Section 3.1, the current plant procedure includes a provision to ensure that inaccessible, below grade concrete will be visually inspected in accordance with NUREG-1801 (GALL), when excavated for any reason (opportunistic inspections), which effectively addressed 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 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.

Request:

Clarify how the Structures Monitoring program is consistent with the GALL-SLR Report for evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas, and provide an enhancement if necessary.

PSL Response:

NUREG-2191 (GALL-SLR) provides recommendations for addressing detection of aging effects for inaccessible, below-grade concrete structural elements. For plants with non-aggressive groundwater/soil, the “detection of aging effects” program element recommends: (a) evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examining representative samples of the exposed portions of the below-grade concrete, when excavated for any reason.

As stated in Section B.2.3.33 of the SLRA, the groundwater/soil at PSL is considered to be aggressive due to its location and the chloride levels of seawater. Therefore, the above-stated recommendations are not applicable to PSL.

For plants with aggressive groundwater/soil and/or where the concrete structural elements have experienced degradation, GALL-SLR recommends that a plant-specific AMP accounting for the extent of degradation experienced should be implemented to manage the concrete aging during the subsequent period of extended operation. In order to meet this recommendation, the PSL Structures Monitoring AMP includes a plant-specific enhancement. The plant-specific enhancement was expanded to include additional details in Attachment 30 of SLRA Supplement 1 (ADAMS Accession No. ML22097A202). The plant-specific enhancement includes: (a) a baseline inspection of inaccessible concrete conducted prior to the SPEO, (b) a baseline evaluation performed prior to the SPEO, (c) periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years), and (d) periodic evaluation updates throughout the SPEO at a frequency not to exceed 5 years.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP – Hurricane Protection Sheet Piles

RAI B.2.3.34-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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) Section B.2.3.34, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," states that the St. Lucie Nuclear Plant (PSL) Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program (AMP), with enhancements, will be consistent without exception to the ten program elements of Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report AMP XI.S7, "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

SLRA Section 2.4.14 states that "the ultimate heat sink dam evaluation boundary is at the exterior surface of the structure" and "adjacent hurricane protection sheet piles are also considered within the evaluation boundary." SLRA Table 2.4.14 and UFSAR 3.8.1.1.5 show that steel sheet piling (beneath dam) is subject to Aging Management Review (AMR).

For the "scope of program" program element, the GALL-SLR Report AMP states that the scope of the program includes structural steel, and structural bolting associated with water-control structures, steel or wood piles and sheeting required for the stability of embankments and channel slopes, and miscellaneous steel, such as sluice gates and trash racks.

Issue:

The staff is unclear where the hurricane protection sheet piles are located, or if they are the same item as the sheet piling beneath the dam, since there is no discussion of hurricane protection sheet piles provided in SLRA Section B.2.3.34. In addition, the staff was unable to identify AMR line items in the SLRA that addressed this commodity.

It is unclear whether the hurricane protection sheet piles discussed in SLRA Section 2.4.14 are within the scope of subsequent license renewal and subject to AMR.

Request:

1. Provide the description of the hurricane protection sheet piles and their intended functions.

2. Clarify whether the hurricane protection sheet piles are within the scope of subsequent license renewal and subject to AMR.

3. If the hurricane protection sheet piles are within the scope of subsequent license renewal and subject to AMR, explain how aging management will be accomplished, provide associated AMR Table 2 items and update SLRA as necessary.

PSL Response:

The numbered responses below correspond to the numbered requests above.

1. The hurricane protection sheet piles are steel sheet piles that are installed along the banks of Big Mud Creek on either side of the ultimate heat sink dam to protect against erosion of the banks during hurricanes. The sheet piles run mostly parallel to the road for a combined total of 598'-6". The sheet piles are driven below ground to elevation –26.5 ft. The top elevation of the sheet piles varies between +11.00 ft and +16.00 ft. The top of the sheet piles is visible at ground level.

2. As stated in SLRA Section 2.4.14 (ADAMS Accession No. ML21285A110), the hurricane protection sheet piles are considered within the evaluation boundary. During initial license renewal, the hurricane protection sheet piles were evaluated and determined not to be within the scope of license renewal as they did not support an intended function. For subsequent license renewal, the hurricane sheet piles are conservatively considered to be within the scope of SLR and subject to AMR.

3. Aging of the hurricane protection sheet piles will be managed by the PSL Structures Monitoring AMP, consistent with SRP Table 1 Item 3.5-1, 079 and NUREG-2191 Item III.A3.TP-219 for "steel components: piles" exposed to "soil, groundwater" and with SRP Table 1 Item 3.5-1, 077 and NUREG-2191 Item III.A3.TP-302 for "steel components: all structural steel" exposed to "air-outdoor." The implementing procedures for the PSL Structures Monitoring AMP currently includes the hurricane protection sheet piles; therefore, no additional enhancements are necessary.

References:

None.

Associated SLRA Revisions:

SLRA Table 2.4-14, page 2.4-30, is revised as follows:

Table 2.4-14
Ultimate Heat Sink Dam (Barrier Wall)
Components Subject to Aging Management Review

Component Type	Component Intended Function(s)
Checkered plate Grating Handrails Ladders Platforms Stairs	Structural support
Concrete: foundation	Structural support
Concrete: foundations, roof, slabs, walls	Missile barrier Shelter, protection Structural support
Miscellaneous steel (i.e., missile barriers, hatch covers, etc.)	Shelter, protection Missile barrier
Steel sheet piling (beneath dam)	Shelter, protection
Steel sheet piling (hurricane protection)	**Shelter, protection**
Structural bolting	Structural support

SLRA Table 3.5-1 Item Number 3.5-1, 079, page 3.5-65, is revised as follows:

Table 3.5-1: Summary of Aging Management Evaluations for the Containments, Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
3.5-1, 079	Steel components: piles	Loss of material due to corrosion	AMP XI.S6, "Structures Monitoring"	No	Consistent with NUREG-2191. The Structures Monitoring (B.2.3.33) AMP is credited with managing loss of material of steel piles for the Ultimate Heat Sink Dam (Barrier Wall)**, hurricane protection,** and discharge nose wave protection (Yard Structures).

SLRA Table 3.5.2-14, page 3.5-128, is revised as follows:

Table 3.5.2-14: Ultimate Heat Sink Dam (Barrier Wall) – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	Table 1 Item	Notes
Concrete: foundations, roof, slabs, walls (accessible)	Missile barrier Shelter, protection Structural support	Concrete (reinforced)	Raw water	Cracking	Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.2.3.34)	III.A6.T-34	3.5-1, 096	A
Concrete: foundations, roof, slabs, walls (accessible)	Missile barrier Shelter, protection Structural support	Concrete (reinforced)	Raw water	Increase in porosity and permeability Loss of strength	Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.2.3.34)	III.A6.TP-37	3.5-1, 061	A
Concrete: foundations, roof, slabs, walls (accessible)	Missile barrier Shelter, protection Structural support	Concrete (reinforced)	Raw water	Cracking Loss of bond Loss of material	Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.2.3.34)	III.A6.TP-38	3.5-1, 059	A
Concrete: foundations, roof, slabs, walls	Missile barrier Shelter, protection Structural support	Concrete (reinforced)	Water – flowing	Loss of material	Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.2.3.34)	III.A6.T-20	3.5-1, 056	A
Miscellaneous steel (i.e., hatch covers, missile barriers, etc.)	Missile barrier Shelter, protection	Steel	Air – indoor uncontrolled Air – outdoor	Loss of material	Structures Monitoring (B.2.3.33)	III.A8.TP-302	3.5-1, 077	B
Steel sheet piling (beneath dam)	Shelter, protection	Steel	Groundwater/soil	Loss of material	Structures Monitoring (B.2.3.33)	III.A3.TP-219	3.5-1, 079	B
Steel sheet piling (hurricane protection)	**Shelter, protection**	**Steel**	**Groundwater/soil**	**Loss of material**	**Structures Monitoring (B.2.3.33)**	**III.A3.TP-219**	**3.5-1, 079**	**B**

Associated Enclosures:

None.

Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP – Parameters Monitored and Inspected and Acceptance Criteria

RAI 3.5.2.2.2.1-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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) Section B.2.3.34, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” stated that the Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program (AMP), with enhancements will be consistent without exception to the ten program elements of Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report AMP XI.S7, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.”

SLRA Sections 2.4.8 and 2.4.14 list the cooling canals, earthen canal dikes and steel sheet piles within the scope of subsequent license renewal.

For the “parameters monitored or inspected” program element, the GALL-SLR Report AMP states that parameters to be monitored and inspected for earthen embankment structures include settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, proper functioning of drainage systems, and degradation of slope protection features, and parameters monitored for channels and canals include erosion or degradation that may impose constraints on the function of the cooling system and present a potential hazard to the safety of the plant. The GALL-SLR Report AMP also states that submerged emergency canals (e.g., artificially dredged canals at the riverbed or the bottom of the reservoir) are monitored for sedimentation, debris, or instability of slopes that may impair the function of the canals under extreme low flow conditions.

For the “acceptance criteria” program element, the GALL-SLR Report AMP states that acceptance criteria for earthen structures, such as canals and embankments, are consistent with programs falling within the regulatory jurisdiction of the Federal Energy Regulatory Commission (FERC) or the United State Army Corps of Engineers (USACE).

Issue:

During the review of the SLRA AMP basis document and procedure ADM-17.32, "Structures Monitoring Program," Revision 7, the staff could not find the "parameters monitored or inspected" and the "acceptance criteria" program elements for the cooling canals, earthen canal dikes and steel sheet piles.

Request:

1. Provide the "parameters monitored or inspected" and the "acceptance criteria" program elements for the cooling canals, earthen canal dikes and steel sheet piles.

2. Clarify whether enhancements to the Inspection of Water-Control Structures Associated with Nuclear Power Plants program are necessary in order to be consistent with the ten program elements of GALL-SLR Report AMP XI.S7, "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

PSL Response:

The numbered responses below correspond to the numbered requests above.

1. The description of the Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP in SLRA Section B.2.3.34, which was updated in SLRA Supplement 1 (ADAMS Accession No. ML22097A202) regarding an exception and clarifying bolting related enhancements, generally touches on the parameters monitored or inspected for the cooling canals and earthen dikes, as well as the inaccessible steel sheet piles. Submerged concrete structures (such as the portion of the cooling canal between the emergency cooling canal and the intake structure) are inspected when dewatered or using video divers with videos reviewed by a qualified responsible engineer or evaluator. Areas covered by silt, vegetation, or marine growth (such as earthen dikes) are not considered inaccessible and are cleaned and inspected in accordance with the standard inspection frequency.

   The implementing procedure directs the inspection of the canals and visual examination of the concrete paving and grout-filled fabric between the intake structure and the ultimate heat sink (UHS) dam. Inspection of the canals and dikes is performed to manage the aging effects of loss of material and change in material properties. However, parameters monitored or inspected for the cooling canals, including the earthen dikes, for the Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP will be updated as provided below to provide clearer detail on the possible changes in material properties:

   - The intake cooling water canal earthen embankments are inspected for settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, and degradation of slope protection features.

- The intake cooling water canal erosion protection, concrete paving, and grout filled fabric are inspected for loss of material, cracking, increase in porosity and permeability, loss of strength, loss of bond, distortion, and loss of form.

- The earthen emergency cooling canal is inspected for loss of form and loss of material and is monitored for sedimentation, debris, and instability of slopes that may impair the function of the canals under extreme low flow conditions.

- Diver inspections include evidence of undercutting at the UHS dam.

Steel sheet piles are located in the soil fully beneath the UHS dam and, as such, are inaccessible. Other steel sheet piles protect the nose of the discharge canal bank. As discussed for item 3.5-1, 079 in SLRA Rev. 1 (ADAMS Accession No. ML2185A110), both of these sheet piles (similar to other inaccessible structural components) are inspected when excavated for any reason (opportunistic inspections) under the Structures Monitoring (SLRA Section B.2.3.33) AMP for loss of material. The response to RAI B.2.3.33-1, Attachment 12 to this letter, includes a corresponding enhancement to the Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP.

The SSCs inspected under the Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP at St. Lucie do not fall under the jurisdiction of the Federal Energy Regulatory Commission (FERC) or the United States Army Corps of Engineers (USACE). The existing acceptance criteria presently documented in the Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP will be utilized for the cooling canals, earthen canal dikes and steel sheet piles.

2. Enhancements to the Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP are warranted for clearer consistency with the ten program elements of GALL-SLR Report AMP XI.S7, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.” These enhancements are detailed in the Associated SLRA Revisions shown below.

References:

None.

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.4, commitment No. 37 portion of Table 19-3 beginning on page A1-104, as amended by Attachments 5 and 12 to this letter, is further revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
37	Inspection of Water-Control Structures Associated with Nuclear Power Plants (19.2.2.34)	XI.S7	Continue the existing PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including enhancement to: a) **Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** **b**) Revise the implementing procedure to specify the use of **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,” for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. **c**) Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include	No later than 6 months prior to the SPEO, i.e.: PSL1: 09/01/2035

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow. **d**) Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events **e) Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			f) Revise the AMP and implementing procedure to more clearly reflect the following parameters monitored or inspected: • The intake cooling water canal earthen embankments are inspected for settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, and degradation of slope protection features. • The intake cooling water canal erosion protection, concrete paving & grout filled fabric are inspected for, loss of material, cracking, increase in porosity and permeability, loss of strength, loss of bond, distortion, and loss of form. • The emergency cooling canal is inspected for loss of form, loss of material, is monitored for sedimentation, debris, and instability of slopes that may impair the function of the canals under extreme low flow conditions. • Diver inspections include evidence of undercutting at the UHS dam.	

SLRA Appendix A2, Section 19.4, commitment No. 37 portion of Table 19-3 on page A2-105, as amended by Attachments 5 and 12 to this letter, is further revised as follows:

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
37	Inspection of Water-Control Structures Associated with Nuclear Power Plants (19.2.2.34)	XI.S7	Continue the existing PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP, including enhancement to: **a) Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.** **b**) Revise the implementing procedure to specify the use of **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," for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. **c**) Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow.	No later than 6 months prior to the SPEO, i.e.: PSL2: 10/06/2042

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			d) Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events. e) **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.** **f) Revise the AMP and implementing procedure to more clearly reflect the following parameters monitored or inspected:** • **The intake cooling water canal earthen embankments are inspected for settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from**	

Table 19-3
List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			originally constructed slopes), seepage, and degradation of slope protection features. • The intake cooling water canal erosion protection, concrete paving & grout filled fabric are inspected for, loss of material, cracking, increase in porosity and permeability, loss of strength, loss of bond, distortion, and loss of form. • The emergency cooling canal is inspected for loss of form, loss of material, is monitored for sedimentation, debris, and instability of slopes that may impair the function of the canals under extreme low flow conditions. • Diver inspections include evidence of undercutting at the UHS dam.	

SLRA Appendix B, Section B.2.3.34, page B-254, as amended by Attachment 12 to this letter, is further revised as follows:

Enhancements

The PSL Inspection of Water-Control Structures Associated with Nuclear Power Plants AMP will be enhanced as follows, for alignment with NUREG-2191. The enhancements are to be implemented no later than 6 months prior to entering the SPEO.

Element Affected	Enhancement
1. Scope	**Inaccessible concrete/foundations exposed to groundwater/soil and water-flowing.**
2. Preventive Actions	**Revise the implementing procedure to reference EPRI Reports 1015336 and 1015337 and to incorporate guidance for 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. Additionally, procedures will be updated to explicitly prohibit the use of molybdenum disulfide and other lubricants containing sulfur on structural bolting.**
2. Preventive Actions	Revise the implementing procedure to specify the use of **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," for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts.
3. Parameters Monitored or Inspected	**Monitoring of the condition of inaccessible concrete, including pH and chloride concentration, of concrete exposed to groundwater/soil and water-flowing environment for evidence of aggressive chemical attack or leaching and carbonation.**
3. Parameters Monitored or Inspected	**Revise the AMP and implementing procedure to more clearly reflect the following parameters monitored or inspected:** • **The intake cooling water canal earthen embankments are inspected for settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, and degradation of slope protection features.** • **The intake cooling water canal erosion protection, concrete paving & grout filled fabric are inspected for, loss of material, cracking, increase in porosity and permeability, loss of strength, loss of bond, distortion, and loss of form.** • **The emergency cooling canal is inspected for loss of form, loss of material, is monitored for sedimentation, debris, and instability of slopes that**

Element Affected	Enhancement
	may impair the function of the canals under extreme low flow conditions. • Diver inspections include evidence of undercutting at the UHS dam.
4. Detection of Aging Effects	Revise the implementing procedure to state that further evaluation of evidence of groundwater infiltration or through-concrete leakage may also include destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels, and that assessments may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the leakage water if leakage volumes allow.
4. Detection of Aging Effects	Revise the severe weather implementing procedure to include performance of structural inspections after major unusual events such as hurricanes, floods, or seismic events.
4. Detection of Aging Effects	**Guidance on baseline inspection with excavation, visual inspection, and physical inspection of the inaccessible concrete through pH analysis and a chloride concentration test of concrete exposed to groundwater/soil and water-flowing at a location near the coastline and a location in the main plant area for comparison prior to the SPEO. Include periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years). Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.** **Degradation of accessible areas due to aging mechanisms such as chemical attack, leaching of calcium hydroxide, and carbonation can be used as an indicator for reinforced concrete conditions in inaccessible areas. Carbonation test results from accessible areas may be used as an indicator to determine the extent of condition of inaccessible reinforced concrete that is exposed to similar environmental conditions and of similar age.**
5. Monitoring and Trending	**Guidance for the evaluation of the baseline inspection results and related OE, with concrete exposed to ambient air and to groundwater/soil, for concrete susceptible to aging effects related to an aggressive environment prior to the SPEO to determine subsequent inspection/evaluation requirements and intervals (not to exceed 5 years), with periodic updates based on periodic inspections and OE.**

Element Affected	Enhancement
	Guidance for the evaluation of baseline inspection results and related OE of concrete exposed to water-flowing for evidence of leaching of calcium hydroxide and carbonation will be developed prior to the SPEO. **The baseline evaluation will determine whether leaching and carbonation are occurring or causing adverse effects. Subsequent inspection/evaluation requirements and intervals (not to exceed 5 years), with periodic updates based on periodic inspections and OE, will be developed if leaching or carbonation is occurring in accessible or inaccessible areas that impacts intended function.** **The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.**
6. Acceptance Criteria	**Acceptance criteria for concrete inspections will be consistent with ACI 349.3R and consider the correlation between the chloride ion concentration within the concrete cover necessary to induce corrosion and alkalinity level of the concrete covering the rebar for inaccessible concrete exposed to groundwater/soil and water-flowing environments.**

Associated Enclosures

None.

Cracking due to Expansion from Alkali Silica Reaction

RAI 3.5.2.2.2.1-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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:

Cracking due to expansion from reaction with aggregates could occur in inaccessible concrete areas of Groups 1-5, 7-9 structures, and Group 6 structures as discussed in the Standard Review Plan for Subsequent License Renewal (SRP-SLR) guidance. The related SRP-SLR Sections 3.5.2.2.1.8, 3.5.2.2.2.1 item 2, and 3.5.2.2.2.3 item 2, associated with SRP-SLR Table 3.5-1 items 3.5.1-012, 3.5.1-043, and 3.5.1-050, respectively, recommend further evaluation to determine if a plant-specific Aging Management Program (AMP) is required to manage this aging effect.

The corresponding review procedures/criteria in SRP-SLR Sections 3.5.3.2.1.8, 3.5.3.2.2.1 item 2, and 3.5.3.2.2.3 item 2, state that a plant-specific evaluation or program is required to manage cracking due to reaction with aggregates if (1) reactivity tests or petrographic examinations of concrete samples identify reaction with aggregates, or (2) accessible concrete exhibits visual indications of aggregate reactions, such as “map” or “patterned” cracking, alkali-silica gel exudations, surface staining, expansion causing structural deformation, relative movement or displacement, or misalignment/distortion of attached components.

Subsequent License Renewal Application (SLRA) Sections 3.5.2.2.1.8, 3.5.2.2.2.1 item 2, and 3.5.2.2.2.3 item 2, state that the Structures Monitoring AMP has been refined, based on industry/fleet information, to include visual examination for patterned cracking, darkened crack edges, water ingress and misalignment that would be indicative of reaction with aggregates, such as alkali silica reaction (ASR) and alkali carbonate reaction (ACR), and includes opportunistic inspection of inaccessible concrete locations.

On November 30 - December 1, 2021, the staff performed an on-site audit at PSL to gain a general overview of current conditions of the structures compared to the provided operational experience (OE), and an understanding on the pattern cracking or crazed concrete cracking identified in the Turbine Building and the Reactor Auxiliary Building. During the walkdown of the Turbine Building and the Reactor Auxiliary Building, the staff noticed the pattern cracking or

crazed concrete cracks on the top of the concrete roof slab. In the review of operating experience AR 01693560 and AR 01725652, the staff also noted that the applicant initially identified potential ASR issue in the roof protecting the Unit 1 Reactor Auxiliary Building, but the applicant's subsequent engineering evaluation determined that ASR was not present because the critical ASR characteristics were missing. The ARs also stated that ASR can be visually detected based on the "crazing" pattern on the concrete surfaces, but confirmation requires petrographic analysis.

Issue:

It is unclear how the applicant's engineering evaluation of the identified crazing pattern cracks determined that ASR was not present at the St. Lucie Nuclear Plant without the confirmation of petrographic analysis.

During the review of procedure ADM-17.32, "Structures Monitoring Program", Revision 9, the staff noted that the current procedure does not include activities for detecting or monitoring ASR degradation in the "parameters to be monitored or inspected" and the "detection of aging effects" program elements. Therefore, it is unclear how the Structures Monitoring AMP incorporated the visual examination for patterned cracking, darkened crack edges, water ingress and misalignment to detect degradations indicative of reaction with aggregates, as stated in the SLRA Sections 3.5.2.2.1.8, 3.5.2.2.2.1 item 2, and 3.5.2.2.2.3 item 2.

In addition, the staff reviewed the SLRA Sections 3.5.2.2.1.8, 3.5.2.2.2.1 item 2 and 3.5.2.2.2.3 item 2, and found that these SLRA Sections lack the description information of operating experiences related to the crazing pattern cracks.

Request:

1. Clarify whether the procedure ADM-17.32 will be revised to include the "parameters to be monitored or inspected" and the "detection of aging effects" program elements for concrete degradations due to ASR and provide an enhancement to the Structures Monitoring program if necessary.

2. Describe the operating experiences related to the crazing pattern cracks identified for the concrete elements for Groups 1-5, 7-9 structures, and Group 6 structures in the SLRA Sections 3.5.2.2.1.8, 3.5.2.2.2.1 item 2, and 3.5.2.2.2.3 item 2.

3. Clarify how the applicant's evaluation of the identified crazing pattern cracks determined that ASR was not present at the St. Lucie Nuclear Plant without the confirmation of petrographic analysis.

4. Evaluate whether a plant-specific evaluation or program is required to manage cracking due to reaction with aggregates in the SLRA Sections 3.5.2.2.1.8, 3.5.2.2.2.1 item 2, and 3.5.2.2.2.3 item 2.

5. Update the SLRA accordingly based on the responses above.

PSL Response:

The numbered responses below correspond to the numbered requests above.

1. Yes, the current revision of procedure ADM-17.32 includes the necessary “parameters monitored or inspected” and “detection of aging effects” elements for concrete degradations due to ASR. After submission of the SLRA, the lack of information in the procedure relative to ASR was identified and a new revision was processed in 2022. The information had been added to a previous procedure based on NRC Information Notice 2011-20 but was inadvertently left out of the procedure that superseded it. Since the current revision of ADM-17.32 has corrected this finding, no additional changes to the procedure regarding ASR/ACR are anticipated. The updated procedure also implements the enhancements committed to in SLRA Supplement 1 Attachment 5 (ADAMS Accession No. ML22097A202). A copy of the current revision of procedure ADM-17.32 is now available on the ePortal.

2. As described in SLRA Section B.2.3.33 (ADAMS Accession No. ML21285A110), PSL performed an evaluation in response to operating experience at Seabrook that detailed the discovery of indications of alkali-silica reaction in concrete structures protecting safety-related equipment. The PSL evaluation is documented in Action Request (AR) 01693560 and concluded that ASR deficiencies have not surfaced in St. Lucie structures. The evaluation considered site configuration and structures monitoring program inspections, which have not identified ASR indications or deficiencies that would indicate a potential ASR problem in any building. A procedure change was initiated to provide guidance for identifying the presence of ASR in procedure SCEG-009 (which has since been superseded by procedure ADM-17.32). As part of this evaluation, potential ASR indications on the Unit 1 reactor auxiliary building (RAB) roof were also reviewed. Apart from these “potential” ASR indications on the RAB roof, there have been no crazed pattern cracking indications on structures in the scope of subsequent license renewal, as indicated in SLRA Section 3.5.2.2.1.8. Additionally, as described in SLRA Section 2.4.13 (ADAMS Accession No. ML21285A110), the turbine buildings are primarily open steel frame structures.

   AR 0175652 documented the 2012 discovery and evaluation of concrete cracks in the roof of the Unit 1 RAB that displayed a craze pattern potentially indicative of ASR. The evaluation of the U1 RAB roof degradation, which was prepared, reviewed, and approved consistent with plant procedures, noted that, in addition to the crazing pattern, ASR also presents other characteristics such as gel deposits and/or dark stains as a result of the gel. The degradation in the U1 RAB roof did not display any evidence of gel deposits or dark stains. Subsequent structures monitoring inspections did not identify any further degradation or issues.

3. As discussed in the response #2 above, AR 01725652 documents operating experience regarding evidence of cracks in a crazing pattern on the 82’ elevation of the Unit 1 RAB roof potentially being due to ASR. There are numerous causes of pattern cracking in concrete, such as temperature or shrinkage, which are common causes of cracking in south Florida. In addition to craze cracking, characteristics of ASR include the formation of gel, which is created by the reaction, and dark discoloration in the cracks, which is caused by the gel formation. ASR-induced degradation can only be confirmed by optical

microscopy performed as part of petrographic examination of concrete core samples. However, ASR can be eliminated as a cause of degradation if the additional characteristics are not present. The engineering evaluation noted that these additional characteristics were not present in the cracking observed in the Unit 1 RAB roof. Given that the crazed cracking was not accompanied by the formation of gel and or discoloration, as described in response #2 above, the Seabrook station's engineering department was consulted, due to their extensive experience in identifying ASR. The reviewed and approved engineering evaluation concluded that the cracking was not caused by ASR/ACR and that further investigation, e.g., to rule out ASR, was not necessary. Thus, no petrographic analysis was required or performed. Further, it was the 3" concrete cover slightly sloped for drainage with welded wire fabric reinforcement that exhibited the craze cracking, not the 2' thick reinforced concrete roof structure.

4. SLRA (ADAMS Accession No. ML21285A110) Sections 3.5.2.2.1.8, 3.5.2.2.2.1 Item 2, and 3.5.2.2.2.3 Item 2 state that a plant-specific program or plant-specific enhancement of the Structures Monitoring AMP is not required to manage cracking due to expansion with aggregates. SLRA Supplement 1 Attachment 5 (ADAMS Accession No. ML22097A202) altered the discussion in Sections 3.5.2.2.1.8, 3.5.2.2.2.1 Item 2, and 3.5.2.2.2.3 Item 2 to indicate that the Structures Monitoring AMP will be enhanced to include visual examination for evidence of cracking due to expansion with aggregates. As discussed in response #1 above, implementing procedure ADM-17.32 was revised after initial submittal of the SLRA to include "parameters monitored or inspected" and "detection of aging effects" elements for concrete degradations due to ASR. Therefore, the enhancements referred to in SLRA Supplement 1 Attachment 5 have already been implemented in the implementing procedure and the enhancements are no longer applicable.

5. The SLRA is updated as described below to more clearly reflect that a plant-specific program or plant-specific enhancement of the Structures Monitoring AMP is not required to manage cracking due to expansion with aggregates.

References:

None.

Associated SLRA Revisions:

SLRA Section 3.5.2.2.1.8, beginning on page 3.5-26, is revised as follows:

> However, the Structures Monitoring (B.2.3.33) AMP **includes opportunistic inspection of inaccessible concrete locations and ~~will~~** **has** be**en** **~~enhanced~~refined**, based on industry/fleet information, to include visual examination for patterned cracking, darkened crack edges, water ingress and misalignment that would be indicative of reaction with aggregates**, such as alkali silica reaction (ASR) and alkali carbonate reaction (ACR), and includes opportunistic inspection of inaccessible concrete locations**. As such, **a plant specific program or plant specific enhancement of the Structures Monitoring (B.2.3.33) AMP is not required to manage this aging effect; rather,**

inspections and evaluations performed in accordance with the Structures Monitoring (B.2.3.33) AMP will identify the presence of expansion and cracking due to reaction with aggregates should it occur.

SLRA Section 3.5.2.2.2.1 Item 2, on page 3.5-29 (paragraph continued from page 3.5-28), is revised as follows:

with pertinent ASTM standards at the time of construction**.** The Structures Monitoring (B.2.3.33) AMP **includes opportunistic inspection of inaccessible concrete locations and ~~will~~**has be**en** ~~**enhanced**~~**refined**, based on industry/fleet information, to include visual examination for patterned cracking, darkened crack edges, water ingress and misalignment that would be indicative of reaction with aggregates**, such as alkali silica reaction (ASR) and alkali carbonate reaction (ACR), and includes opportunistic inspection of inaccessible concrete locations. As such, a plant specific program or plant specific enhancement of the Structures Monitoring (B.2.3.33) AMP is not required to manage this aging effect; rather,** inspections and evaluations performed in accordance with the Structures Monitoring (B.2.3.33) AMP will identify the presence of expansion and cracking due to reaction with aggregates should it occur.

SLRA Section 3.5.2.2.2.3 Item 2, on page 3.5-31, is revised as follows:

2. Group 6 structures at PSL are designed and constructed in accordance with ACI 318-63 (Unit 1) and ACI 318-71 (Unit 2) using ingredients/materials conforming to ACI and ASTM standards. The aggregates used in the concrete of the PSL concrete components did not come from a region known to yield aggregates suspected of or known to cause aggregate reactions. All aggregates used at PSL conform to all the requirements of “Specification of Concrete Aggregates” ASTM-C33-67 and –71, to ensure that the hardness, weight, strength, durability, reactivity, and gradation fall within the limits established for a good grade of concrete. PSL concrete components were constructed using aggregates from approved sources whose acceptability was based on established industry standards and ASTM tests~~]~~. The concrete mix uses Type II Portland cement conforming to ASTM C150. Also, for Unit 2, the cement contains no more than 0.60 percent by weight of total alkalis, which prevents harmful expansion due to alkali aggregate reaction. Materials for concrete used in PSL concrete SSCs were specifically investigated, tested, and examined in accordance with pertinent ASTM standards at the time of construction. However, this testing may not fully conform to ASTM C295 specified in NUREG 2192, and, therefore, cracking due to expansion and reaction with aggregates, including alkali silicate reactions ~~(ASR)~~ and alkali carbonate reactions, is a potential aging effect in below grade inaccessible concrete areas for PSL Group 6 structures and will be managed by the PSL Structures Monitoring (B.2.3.33) AMP. The Structures Monitoring (B.2.3.33) AMP **includes opportunistic inspection of inaccessible concrete locations and**

~~will~~has been ~~enhanced~~refined, based on industry/fleet information, to include visual examination for patterned cracking, darkened crack edges, water ingress and misalignment that would be indicative of reaction with aggregates**, such as alkali silica reaction (ASR) and alkali carbonate reaction (ACR), and includes opportunistic inspection of inaccessible concrete locations**. As such, **a plant specific program or plant specific enhancement of the Structures Monitoring (B.2.3.33) AMP is not required to manage this aging effect; rather,** inspections and evaluations performed in accordance with the Structures Monitoring (B.2.3.33) AMP will identify the presence of cracking due to expansion and cracking due to reaction with aggregates should it occur.

SLRA Appendix A1, Section 19.4, Commitment No. 36 portion of Table 19-3 on page A1-103, amended by Attachment 5 to this letter, is further revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
36	Structures Monitoring (19.2.2.33)	XI.S6	Continue the existing PSL Structures Monitoring AMP, including enhancements to: a) Monitor and inspect steel edge supports on masonry walls. b) Specify the use of high-strength bolt storage requirements discussed in Section 2 of the Research Council for Structural Connections publication, “Specification for Structural Joints Using High-Strength Bolts,” for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. c) Inspect concrete structures for increase in porosity and permeability, loss of strength, ~~**indications of cracking and expansion due to reaction with aggregates,**~~ and reduction in concrete anchor capacity due to local concrete degradation. d) Inspect elastomers for loss of material and **cracking**. e) Inspect stainless steel and aluminum components for pitting and crevice corrosion, and evidence of cracking due to SCC. f) Include monitoring and trending of leakage volumes and chemistry for signs of concrete or steel reinforcement degradation if active through-wall leakage or groundwater infiltration is identified. **g) Specify that all bolting is monitored for loss of material, loose bolts, missing or loose nuts, and other conditions indicative of loss of preload.**	No later than 6 months prior to the SPEO, i.e.: PSL1: 09/01/2035

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			h) Include tactile inspection in addition to visual inspection of elastomeric elements to detect hardening. i) Include evidence of water in-leakage as a finding requiring further evaluation. This may include engineering evaluation, more frequent inspections, or destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels. When leakage volumes allow, assessment may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the water. j) **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			k) Require inspections of the Condensate Storage Tank (CST) and Auxiliary Feedwater (AFW) Structures and Piping Inspections in the Trenches every third refueling outage, which will ensure that these inspections are performed at least once per 5 years. **l**) **Include stainless steel ASME Class 1, 2, or 3 support members, welds, bolted connections, or anchorage in the engineering evaluation of acceptance criteria, expansion criteria, and examination frequency if cracking due to SCC in the uncontrolled indoor and outdoor air at PSL is detected for stainless steel mechanical or non-ASME structural components.**	

SLRA Appendix A2, Section 19.4, Commitment No. 36 portion of Table 19-3 on page A2-103, amended by Attachment 5 to this letter, is further revised as follows:

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
36	Structures Monitoring (19.2.2.33)	XI.S6	Continue the existing PSL Structures Monitoring AMP, including enhancement to: a) Monitor and inspect steel edge supports on masonry walls. b) Specify the use of high-strength bolt storage requirements discussed in Section 2 of the Research Council for Structural Connections publication, "Specification for Structural Joints Using High-Strength Bolts," for structural bolting consisting of ASTM A325, ASTM A490, and equivalent bolts. c) Inspect concrete structures for increase in porosity and permeability, loss of strength, ~~**indications of cracking and expansion due to reaction with aggregates,**~~ and reduction in concrete anchor capacity due to local concrete degradation. d) Inspect elastomers for loss of material and **cracking**. e) Inspect stainless steel and aluminum components for pitting and crevice corrosion, and evidence of cracking due to SCC. f) Include monitoring and trending of leakage volumes and chemistry for signs of concrete or steel reinforcement degradation if active through-wall leakage or groundwater infiltration is identified. **g) Specify that all bolting is monitored for loss of material, loose bolts, missing or loose nuts, and other conditions indicative of loss of preload.**	No later than 6 months prior to the SPEO, i.e.: PSL2: 10/06/2042

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			h) Include tactile inspection in addition to visual inspection of elastomeric elements to detect hardening. i) Include evidence of water in-leakage as a finding requiring further evaluation. This may include engineering evaluation, more frequent inspections, or destructive testing of affected concrete to validate existing concrete properties, including concrete pH levels. When leakage volumes allow, assessment may include analysis of the leakage pH, along with mineral, chloride, sulfate, and iron content in the water. j) **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

Table 19-3
List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			k) Require inspections of the Condensate Storage Tank (CST) and Auxiliary Feedwater (AFW) Structures and Piping Inspections in the Trenches every third refueling outage, which will ensure that these inspections are performed at least once per 5 years. **l**) **Include stainless steel ASME Class 1, 2, or 3 support members, welds, bolted connections, or anchorage in the engineering evaluation of acceptance criteria, expansion criteria, and examination frequency if cracking due to SCC in the uncontrolled indoor and outdoor air at PSL is detected for stainless steel mechanical or non-ASME structural components.**	

SLRA Appendix B, Section B.2.3.33, element 3 portion of Enhancements Table on page B-248, is revised as follows:

Element Affected	Enhancement
3. Parameters Monitored or Inspected	Update the governing AMP procedure and other applicable procedures to inspect concrete structures for increase in porosity and permeability, loss of strength, and reduction in concrete anchor capacity due to local concrete degradation.
~~**3. Parameters Monitored or Inspected**~~	~~**Update the governing AMP procedure and other applicable procedures to inspect concrete structures for patterned cracking, darkened crack edges, water ingress and misalignment that would be indicative of reaction with aggregates.**~~
3. Parameters Monitored or Inspected	Update the governing AMP procedure and other applicable procedures to inspect elastomers for loss of material and loss of strength.
3. Parameters Monitored or Inspected	Update the governing AMP procedure and other applicable procedures to inspect SS and aluminum components for pitting and crevice corrosion, and evidence of cracking due to SCC.
3. Parameters Monitored or Inspected	Update the governing AMP procedure and other applicable procedures to include monitoring and trending of leakage volumes and chemistry for signs of concrete or steel reinforcement degradation if active through-wall leakage or groundwater infiltration is identified.

Associated Enclosures:

None.

Water Chemistry AMP – Aging Management of the Refueling Cavity and Spent Fuel Pool

RAI 3.5.2.2.2.4-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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:

Cracking due to stress corrosion cracking (SCC) and loss of material due to pitting and crevice corrosion could occur in stainless steel (SS) and aluminum alloy support members, welds, bolted connections, or support anchorage to building structure exposed to air or condensation per the Standard Review Plan for Subsequent License Renewal (SRP-SLR) guidance. The related SRP-SLR Section 3.5.2.2.2.4 associated with SRP-SLR Table 3.5-1 items 3.5.1-099 and 3.5.1-100 recommends further evaluation to determine if the plant-specific air or condensation environments are aggressive enough to result in loss of material or cracking after prolonged exposure.

Subsequent License Renewal Application (SLRA) Section 3.5.2.2.2.4, as amended by Supplement 1 dated April 7, 2022 (ADAMS Accession No. ML22097A202), states that cracking due to SCC and loss of material due to pitting and crevice corrosion is a potentially applicable aging effect for plant structures for SS and aluminum and is managed with the Structures Monitoring Aging Management Program (AMP), the ASME Section XI, Subsection IWF AMP, and the Fire Protection AMP.

SLRA Table 3.5-1 item 3.5.1-100, as amended by Supplement 1, states that the Structures Monitoring AMP is credited with managing loss of material and cracking of nonsafety-related aluminum and SS component supports, anchorage embedment, electrical and instrument panel and enclosures, conduits and cable trays exposed to air, and the Fire Protection AMP is credited with managing loss of material and cracking in aluminum and SS fire barrier penetrations and radiant energy shields.

SLRA Table 3.5.2-1 and Table 3.5.2-7, as amended by Supplement 1, list SS reactor cavity and fuel pool components associated with Table 3.5-1 item 3.5.1-100, with generic note E and plant-specific note 12 and 6, respectively. PSL credits the Water Chemistry AMP and monitoring cavity or pool water levels to manage the aging effects for the pressure boundary

components of reactor cavity seal ring and fuel pool gates exposed to an uncontrolled indoor air environment. Also, for the pressure boundary components of refueling cavity liner plate and pool liner plates exposed to an uncontrolled indoor air environment, PSL credits the Water Chemistry AMP and monitoring cavity or pool water levels and leakage from the leak chase channels to manage the aging effects.

Issue:

It is unclear to the staff how the Water Chemistry AMP is capable of managing the aging effects of SCC and loss of material for the above stated SS pressure boundary components that are exposed to an uncontrolled indoor air environment, given that the PSL Water Chemistry AMP is a mitigation program that relies on monitoring and control of reactor water chemistry, and it does not provide for detection of any aging effects of concern for the in-scope components, as stated in the PSL Water Chemistry AMP basis document.

It appears that evidence of cracking and loss of material for pressure boundary intended function components depends on monitoring the refueling cavity or spent fuel pool water level to detect leakage. However, it is unclear how water leakage monitoring can be used as an indicator of cracking and material loss in components exposed to air rather than water. It is also unclear if there are any provisions in procedures associated with these monitoring activities to ensure cracking and material loss of associated components exposed to indoor air will be detected prior to loss of pressure boundary intended function.

SLRA Table 3.5-1 item 3.5.1-100, as amended by Supplement 1, does not include or provide information in its discussion column about the Water Chemistry AMP and monitoring of the refueling cavity or spent fuel pool water level to detect leakage for associated pressure boundary components.

SLRA Section 3.5.2.2.2.4, as amended by Supplement 1, does not provide further evaluation of the Water Chemistry AMP and monitoring of both the refueling cavity and spent fuel pool water levels to manage the aging effects for the pressure boundary components exposed to air associated with SLRA Table 3.5-1 item 3.5.1-100.

Request:

1. Describe how the Water Chemistry AMP is capable of and adequate (scope, inspection methods and frequency) for managing the aging effects for the SS pressure boundary components in SLRA Tables 3.5.2-1 and 3.5.2-7, associated with SLRA Table 3.5-1 Item 3.5.1-100, that are exposed to an uncontrolled indoor air environment.

2. Explain how water leakage monitoring can be used as an indicator of cracking and material loss in components exposed to indoor air rather than water. Clarify whether there are any provisions in the procedures to ensure cracking and material loss of associated components exposed to indoor air will be detected prior to loss of pressure boundary intended function. Describe the provisions if any. If not, enhance relevant procedures to include appropriate provisions.

3. If the Water Chemistry AMP is determined to be an appropriate AMP, update SLRA Table 3.5-1 item 3.5.1-100 to include the Water Chemistry AMP and monitoring the refueling cavity or spent fuel pool water level to detect leakage for the associated Table 2 components.

4. Update SLRA Section 3.5.2.2.2.4 accordingly based on the responses above to provide further evaluation of the Water Chemistry AMP and its scope, monitoring method/interval to manage the aging effects of cracking and loss of material for the components exposed to indoor air associated with SLRA Table 3.5-1 item 3.5.1-100.

PSL Response:

The intended function for each of the subject stainless steel component types is to maintain a pressure boundary containing treated borated water within the refueling cavity or spent fuel pool. Entries in SLRA Tables 3.5.2-1 and 3.5.2-7 associated with SLRA Table 3.5-1 Item 3.5.1-078 represent the wetted surfaces of the subject SS component types.

For completeness, initial entries in SLRA Tables 3.5.2-1 and 3.5.2-7 associated with SLRA Table 3.5-1 Item 3.5.1-100 were included to represent the following portions of the subject stainless steel component types that are normally dry:

- Reactor cavity seal ring (external surfaces exposed to containment atmosphere);
- Refueling cavity liner plate (fully submerged during refueling; otherwise exposed to containment atmosphere);
- Fuel pool gates (external surfaces exposed to fuel handling building atmosphere); and
- Pool liner plates (surfaces above spent fuel pool level exposed to fuel handling building atmosphere).

Upon further review, these entries were determined to be unnecessary as the normally dry portions of the subject component types do not perform the pressure boundary function. Accordingly, entries for the four SS component types listed above are removed from SLRA Tables 3.5.2-1 and 3.5.2-7. This determination is consistent with treatment of the equivalent component types in a prior SLRA accepted by the NRC staff (References 1 and 2).

The numbered responses below correspond to the numbered requests above.

1. The Water Chemistry AMP is no longer credited to manage aging for portions of the four stainless steel component types that are exposed to an uncontrolled indoor air environment, and therefore are removed from SLRA Tables 3.5.2-1 and 3.5.2-7. Management of aging effects for the wetted surfaces provides reasonable assurance of detecting degradation prior to loss of intended function for these component types. Implementing procedures for the Water Chemistry AMP specify sampling requirements for the refueling cavity and spent fuel pool. During normal operations at PSL, water in the spent fuel pool is sampled weekly. Prior to flooding the refueling cavity above the reactor flange for refueling operations, the refueling cavity and spent fuel pool are sampled. The refueling cavity and spent fuel pool are sampled daily thereafter during Mode 6. Sample analysis is required for both locations within 24 hours of moving the

first fuel assembly or control element assembly. The chemistry supervision/program owner is notified of any out of specification result. Out of specification conditions are documented and evaluated in accordance with the corrective action program.

2. Water leakage monitoring is no longer credited to manage aging for portions of the four stainless steel component types that are exposed to an uncontrolled indoor air environment, and therefore are removed from in SLRA Tables 3.5.2-1 and 3.5.2-7. The wetted surfaces of the same component types are represented by entries associated with Table 3.5-1 Item 3.5.1-078 in the same two tables. Water leakage monitoring is identified as an indicator of cracking and material loss for such components in NUREG-2191 Chapter III Table A5 (Item III.A5.T-14). At PSL, the flooded refueling cavity and spent fuel pool level are monitored by control room personnel as appropriate during fuel movements. Verification of spent fuel pool level is required within 8 hours prior to core alterations. Refueling cavity water level and spent fuel pool level are verified within 2 hours prior to core alterations. Refueling cavity water level is also verified daily. Any time an acceptance criterion is not met, the unit supervisor/shift manager is immediately notified and takes appropriate actions per Technical Specifications. When an operating check is performed and reveals undesirable conditions, the operating check is repeated for verification of the condition (when practical) and a CR generated.

3. Entries for the associated Table 2 component types are removed from SLRA Tables 3.5.2-1 and 3.5.2-7; thus, text in the ‘Discussion’ column of SLRA Table 3.5-1 for item 3.5.1-100 is unchanged.

4. Entries for the associated Table 2 component types are removed from SLRA Tables 3.5.2-1 and 3.5.2-7; thus, further evaluation of the Water Chemistry AMP is not added to SLRA Section 3.5.2.2.2.4.

Note that the above portions of the PSL SLRA Revision 1 were also impacted by SLRA Supplement 1 (ADAMS Accession No. ML22097A202), Attachment 6, as described in the ‘Background’ discussion for this RAI. The update provided in SLRA Supplement 1 is also reflected in the SLRA revision below for clarity.

References:

1. Surry Power Station Units 1 and 2 Application for Subsequent License Renewal, October 2018 (ADAMS Accession No. ML18291A828) – Tables 3.5.2-1, 3.5.2-5

2. Final Safety Evaluation Report for the Subsequent License Renewal Application Review for Surry Power Station, Units 1 and 2 (ADAMS Accession No. ML20052F523) – Section 3.5.2.2.2.4

Associated SLRA Revisions:

SLRA Table 3.5.2-1, page 3.5-81, is revised as follows:

Table 3.5.2-1: Containment Building Structures – Summary of Aging Management Evaluation								
Component Type	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-2191 Item**	**Table 1 Item**	**Notes**
~~Reactor cavity seal ring~~	~~Pressure boundary~~	~~Stainless steel~~	~~Air – indoor uncontrolled~~	~~Cracking~~ ~~Loss of material~~	~~**Water Chemistry**~~ ~~(B.2.3.**2**)~~ ~~**and monitoring cavity, pool water levels**~~	~~III.B2.T-37b~~	~~3.5-1, 100~~	~~**E, 12**~~

SLRA Table 3.5.2-1, page 3.5-82, is revised as follows:

Table 3.5.2-1: Containment Building Structures – Summary of Aging Management Evaluation								
Component Type	**Intended Function**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-2191 Item**	**Table 1 Item**	**Notes**
~~Refueling cavity liner plate~~	~~Pressure boundary~~	~~Stainless steel~~	~~Air – indoor uncontrolled~~	~~Cracking~~ ~~Loss of material~~	~~**Water Chemistry**~~ ~~(B.2.3.**2**)~~ ~~**and monitoring cavity, pool water levels and leakage from the leak chase channels**~~	~~III.B5.T-37b~~	~~3.5-1, 100~~	~~**E, 12**~~

SLRA Table 3.5.2-1, page 3.5-86 (notes for Table 3.5.2-1), is revised as follows:

General Notes

A. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.
B. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
C. Component is different, but consistent with material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.
D. Component is different, but consistent with material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
~~**E. Consistent with NUREG-2191 material, environment, and aging effect but a different AMP is credited or NUREG-2191 identifies a plant-specific AMP.**~~
F. Material not in NUREG-2191 for this component.
H. Aging effect not in NUREG-2191 for this component, material, and environment combination.
I. Aging effect in NUREG-2191 for this component, material and environment combination is not applicable.

Plant Specific Notes

1. <u>**Deleted.**</u>
2. Manway at the top of the Containment Vessel is a permanently sealed (welded) shut part of the Containment Vessel.
3. <u>**Deleted.**</u>
4. <u>**Deleted.**</u>
5. Insulation for main steam and feedwater penetrations <u>**is**</u> fully encased in the multiple flued head and guard pipes and there are no plausible moisture, contaminants, or exposures that could degrade the (calcium silicate) insulation.
6. <u>**The s**</u>tainless steel <u>**fuel transfer tube and flange**</u> penetration assemblies include stainless steel pressure-retaining bolting that is managed by the ASME Section XI, Subsection IWE AMP.
7. Irradiation of the concrete primary shield wall is addressed in Section 3.5.2.2.2.6 and is managed by the Structures Monitoring (B.2.3.33) AMP.
8. The loss of fracture toughness aging effect due to irradiation embrittlement of the steel reactor vessel supports and bolting is addressed in Section 3.5.2.2.2.7 and is managed by the ASME Section XI, Subsection IWF (B.2.3.30) AMP**.**
9. Consistent with SLR-ISG-2021-03-STRUCTURES, the Structures Monitoring AMP manages cracking of concrete due to reaction with aggregates.
10. Bottom portion and bottom head of the Containment Vessel are completely encased in concrete fill inside the (Group 1) Shield Building. The operating floor inside the Containment Vessel is this concrete fill.
11. <u>**Component also provides a fire barrier function as evaluated in the Fire Protection Program Design Document that is physically equivalent to the structural functions managed under the associated Containment structural programs or other applicable AMPs.**</u>

12. ~~**Evidence of cracking due to SCC and loss of material due to pitting and crevice corrosion (i.e., leakage) would be readily detected during refueling operations prior to loss of intended function for these components.**~~**Deleted.**
13. **Insulation for Type III penetrations is fully encased in the multiple flued head and guard pipes and there are no plausible moisture, contaminants, or exposures that could degrade the (calcium silicate) insulation.**

SLRA Table 3.5.2-7, page 3.5-105, is revised as follows:

Table 3.5.2-7: Fuel Handling Buildings – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	Table 1 Item	Notes
~~Fuel pool gates~~	~~Pressure boundary~~	~~Stainless steel~~	~~Air – indoor uncontrolled~~	~~Cracking~~ ~~Loss of material~~	~~**Water Chemistry**~~ ~~(B.2.3.~~**~~2~~**~~)~~ ~~**and monitoring cavity, pool water levels**~~	~~III.B5.T-37b~~	~~3.5-1, 100~~	~~**E, 6**~~

SLRA Table 3.5.2-7, page 3.5-106, is revised as follows:

Table 3.5.2-7: Fuel Handling Buildings – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-2191 Item	Table 1 Item	Notes
~~Pool liner plates~~	~~Pressure boundary~~	~~Stainless steel~~	~~Air – indoor uncontrolled~~	~~Cracking~~ ~~Loss of material~~	~~**Water Chemistry**~~ ~~(B.2.3.~~**~~2~~**~~)~~ ~~**and monitoring cavity, pool water levels and leakage from the leak chase channels**~~	~~III.B5.T-37b~~	~~3.5-1, 100~~	~~**E, 6**~~

SLRA Table 3.5.2-7, page 3.5-108 (notes for Table 3.5.2-7), is revised as follows:

General Notes

A. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.
B. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
C. Component is different, but consistent with material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.
D. Component is different, but consistent with material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
~~E. Consistent with NUREG-2191 material, environment, and aging effect but a different AMP is credited or NUREG-2191 identifies a plant-specific AMP.~~

Plant Specific Notes

1. Whereas the NUREG-2191/2192 item calls for a plant-specific AMP, PSL credits an existing AMP based on SLR-ISG-2021-03-STRUCTURE, “Updated Aging Management Criteria for Structures Portions of Subsequent License Renewal Guidance.”
2. Groundwater is considered to be water-flowing.
3. Conservatively, spent fuel pool leakage is identified as water-flowing.
4. There are no leak chase channels to monitor for the fuel pool gates.
5. **Deleted.**
6. ~~**Evidence of cracking due to SCC and loss of material due to pitting and crevice corrosion (i.e., leakage) would be readily detected during refueling operations prior to loss of intended function for these components.**~~**Deleted.**

Associated Enclosures:

None.

Containment Fatigue Waiver

RAI 4.6-1

Regulatory Basis:

10 CFR § 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).

Background:

SRP-SLR Section 4.6.1.1 states, in part: “The ASME Code contains explicit requirements for fatigue parameter evaluations (fatigue analyses or fatigue waivers), which are TLAAs.” SRP-SLR Section 4.6.2 and 4.6.3 provide acceptance criteria and review procedures for fatigue parameter evaluations.

SRP-SLR Section 3.5.2.2.1.5 (as modified by SLR-ISG-2021-03-STRUCTURES (ADAMS Accession No. ML2018A381)) provides guidance for further evaluation of cumulative fatigue damage of containment pressure-retaining boundary components subject to cyclic loading.

SLRA Section 4.6, as amended by Supplement 1 dated April 7, 2022 (ADAMS Accession No. ML22097A202), under subtitle “Metal Containment Fatigue,” states, in part:

> The fatigue waiver evaluation includes the steel containments vessel shells, containment penetration nozzles, personnel air locks, and equipment hatches. ……Fatigue waivers that consider transient cycles which occur over the life of the plant constitute TLAAs.

Similar information as above is indicated in related SLRA Section 3.5.2.2.1.5, SLRA Table 3.5-1, items 009 and 027, SLRA Table 3.5.2-1, and SLRA Sections 19.3.5 in Appendix A1 and Appendix A2, all as amended by SLRA Supplement 1 dated April 7, 2022.

The “TLAA Evaluation” section under “Metal Containment Fatigue” in SLRA Section 4.6 states, in part: “The fatigue waiver for the PSL Unit 1 and Unit 2 containment vessels remain valid through the 80-year SPEO since the service loading of the vessels meet all of the following six fatigue waiver conditions of Section III of the ASME Code.”

The corresponding TLAA Disposition in SLRA Section 4.6 for Metal Containment Fatigue is 10 CFR 54.21(c)(1)(i) indicating that the existing analysis remains valid for the subsequent period of extended operation.

SLRA Table 3.5.2-1, as amended by Supplement 1 dated April 7, 2022, credits the SLRA Section 4.6 TLAA for the following components in addition to the containment vessel: Airlocks, maintenance hatch and accessories; construction hatch and cover; containment vessel nozzle (electrical); containment vessel nozzle (fuel transfer); containment vessel nozzle (mechanical); and containment vessel nozzle of Nickel alloy material (mechanical).

Based on the audit of original stress reports and other calculations related to the containment vessel, the staff noted that fatigue waiver evaluations, in accordance with the ASME code, for the steel containment vessel for PSL Unit 1 and Unit 2 were sufficiently documented only in calculation PSL-1FSC-01-020 “Steel Containment Fatigue Evaluation, PSL Unit 1,” Revision 0 (dated 09/18/2001), and calculation PSL-2FSC-01-021 “Steel Containment Fatigue Evaluation, PSL Unit 2,” Revision 0 (dated 09/18/2001) , respectively. Based on audit of these calculations, the staff noted that the fatigue waiver evaluations were specific to the steel containment vessel shell and based on containment plate material SA516, Grade 70. The staff further noted that the above referenced calculations do not make explicit mention of containment penetration nozzles, personnel airlocks, and equipment hatches as being included in the fatigue waiver evaluations.

Issue:

1) Based on the audit and information provided in SLRA Sections 4.6 and 3.5.2.2.1.5, as amended by Supplement 1 dated April 7, 2022, the staff does not have sufficient information to verify that fatigue waiver analyses exist for containment penetration nozzles, personnel airlocks, equipment hatches, and other credited components in SLRA Table 3.5.2-1 as amended, or justified as being included in or bounded by the fatigue waiver evaluations of the steel containment vessel shell, given that material, geometry and stress conditions of these components may be different from that of the containment vessel shell.

2) SLRA Section 4.6, under TLAA Evaluation for “Metal Containment Fatigue,” does not state the material based on which the summarized evaluation fatigue waiver evaluation for the containment vessel shell was performed.

Request:

1. Describe in sufficient technical detail the basis of how the fatigue waiver evaluation of the containment penetration nozzles, personnel airlocks, equipment hatches, and other stated credited components in SLRA Table 3.5.2-1 as amended, are included in or bounded by the fatigue waiver evaluation of the PSL Unit 1 and Unit 2 containment vessel shell summarized in SLRA Section 4.6, as amended by Supplement 1, considering that material, geometry and stress conditions may differ for these components from that of the containment vessel shell.

   OR,

   Provide fatigue waiver analyses (including material and cyclic inputs, and fatigue parameter evaluations) that demonstrates how the six criteria in the ASME Code, Section III edition of record, for not requiring analysis for cyclic operation are met for the PSL Unit 1 and Unit 2 containment penetration nozzles, personnel airlocks, equipment hatches, and other stated credited components in SLRA Table 3.5.2-1 as amended, for which they are credited in the SLRA, as amended by Supplement 1 dated April 7, 2022.

2. In the summary of results of the fatigue waiver evaluation for the PSL Unit 1 and Unit 2 containments vessel shell provided in the current SLRA Section 4.6 under “TLAA

Evaluation" for "Metal Containment Fatigue," state and justify the material based on which the evaluation was performed.

3. Update applicable SLRA Sections (e.g., 4.6, 3.5.2.2.1.5, Table 3.5-1 and Table 3.5.2-1, related TLAA UFSAR supplements and disposition), as necessary consistent with responses to requests above and fatigue-waiver related calculations of record

PSL Response:

The numbered responses below correspond to the numbered requests above.

1. A new fatigue waiver evaluation, which includes material and cyclic inputs and fatigue parameter evaluations, has been prepared for the PSL Units 1 and 2 containment vessels which demonstrates that the six criteria in the ASME Codes of record have been satisfied. Specifically, the PSL Unit 1 fatigue waiver evaluation has been performed in accordance with Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, Section N-415, "Analysis for Cyclic Operation" and the PSL Unit 2 fatigue waiver evaluation was performed in accordance with the 1971 Edition of ASME Section III, Nuclear Vessels, Section NB-3222.4, "Analysis for Cyclic Operation." These are the applicable codes of record for the PSL Units 1 and 2 containment vessels.

   The design conditions for the PSL Units 1 and 2 containment vessels that are used in the fatigue waiver evaluations are included in Section 3.8.2 of each units UFSAR and are summarized below.

Parameter	PSL Unit 1	PSL Unit 2
Design Pressure	39.6 psig	44 psig
Design Temperature	264 ºF	264 ºF

   The containment vessel materials included in the new fatigue waiver evaluations include those materials specified in Tables 3.8-5 and 3.8-10 of the PSL Units 1 and 2 UFSARs, respectively. These tables are provided below. Note that the new containment vessel fatigue waiver evaluations do not include the bolting material specified in the UFSAR tables as the fatigue analysis requirements for bolting are included in Section N-416.2 of the 1968 Edition of ASME Section III, Nuclear Vessels and in Section NB-3232.3 of the 1971 Edition of ASME Section III, Nuclear Vessels. These code sections state that if the vessel on which the bolts are installed meets all the conditions of N-415-1 for PSL Unit 1 and NB-3222(d) for PSL Unit 2, no fatigue analysis is required for bolting.

PSL-1 Containment Vessel Materials

Material	Specification	Design Stress Intensity (psi)
Plate	SA 516, Gr 70 to SA 300	17,500
Forgings	SA 350, Gr LF2	17,500
	SA 182, F304	13,750
Pipe	SA 333, Gr 1	13,750
	SB 167	18,200
	SB 166	18,800
	SA 516, Gr 70 to SA 300	17,500
Castings	SA 352, Gr LCB	16,250
	SA 351, Gr B8	13,750

PSL-2 Containment Vessel Materials

Material	Specification	Design Stress Intensity (psi)
Plate	SA 516, Gr 70	19,300
	SA 240, Type 304	17,300 (Note 1)
Forgings	SA 350, Gr LF 1	16,500
	SA 350, Gr LF 2	19,300
	SA 182, F304	18,300
Pipe	SA 333, Gr 6	16,500
	SB 168	20,700
	SB 167	20,700
	SB 166	20,700
	SA 516, Gr 70	19,300
Castings	SA 352, Gr LC1	16,200
	SA 351, Gr CF8	16,000

Note 1 – there is no design stress intensity value for the SA 240, Type 304 material in Table 3.8-10 of the PSL Unit 2 UFSAR. However, page AA-14 of the Reference 4 CBI stress report does include the specified value for that material.

In accordance with the respective ASME code sections on fatigue waiver evaluations for the PSL Units 1 and 2 containment vessels, the six criteria to be met for allowing a fatigue waiver were evaluated for all of the materials specified in the tables above. For each criterion, the limiting material(s) and corresponding condition(s) were found to be less than the applicable limits and are summarized below.

Criterion (1) from Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, and Section N-415 of the 1971 Edition of the ASME Code, Section III describes the specified number of times (including startup and shutdown) that the component pressure will be cycled from atmospheric pressure to operating pressure and back to atmospheric pressure. The number of allowed cycles at Sa = 3 times the design stress intensity S_m is compared to the number of applied cycles.

For this condition, 500 operating pressure cycles consistent with the design number of heatup and cooldown cycles for PSL Units 1 and 2 are considered. The limiting material for this criterion based on the lowest number of allowable operating pressure cycles for PSL Unit 1 is the SB 166 nickel alloy pipe. For PSL Unit 2, the limiting materials are the SA 516, Grade 70 carbon steel plate and the SA 350 Grade LF2 carbon steel forgings. The design stress intensity (S_m) for SB 166 is 18.8 ksi and for SA 516 Grade 70 and SA 350 Grade LF2 is 19.3 ksi. For PSL Unit 1, for a value of S_a equal to $3S_m$, or 56.4 ksi, the allowable number of cycles (N) is 3000 [Reference 3, Figure N-415(A)]. For PSL Unit 2, for a value of S_a equal to $3S_m$, or 57.9 ksi, the allowable number of cycles (N) is 2500 [Reference 4, Figure I.9-1]. As 500 operating pressure cycles are less than 3000 cycles for PSL Unit 1 and 2500 cycles for PSL-2, Criterion 1 is satisfied for both units.

Criterion (2) from Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, and Section N-415 of the 1971 Edition of the ASME Code, Section III describes that the full range of pressure fluctuations during normal operation shall not exceed the quantity of (1/3) x design pressure x (S_a/S_m). The Reference 1 and 2 fatigue waivers evaluate two cases. The first case considers the number of containment pressure cycles during normal operation to be greater than or equal to 10^6 cycles. The References 5 and 6 fatigue waiver evaluations assume the containment vessels are normally at atmospheric pressure with little or no fluctuation, or essentially 0 psi. Since the value of (1/3) x Design Pressure x (S_a/S_m) is greater than 0 psi for all PSL Units 1 and 2 containment vessel materials, the first case of Criterion 2 is satisfied.

The second case considers the more significant pressure fluctuation cycles which occur when the containment vessels are subjected to periodic integrated leak rate tests (ILRT). Reference 1 and 2 conservatively assume 100 ILRTs which equals greater than one ILRT per year over the 80-year SPEO. The limiting material for the allowable pressure fluctuation for PSL Unit 1 is the SB 166 nickel alloy pipe, and for PSL Unit 2 the limiting materials are the A 516, Grade B carbon steel plate and the SA 350 Grade LF2 carbon steel forgings. For PSL Unit 1, the value of S_a for 100 cycles is 200 ksi [Reference 3, Figure N-415(A)]. For PSL Unit 2, the value of S_a for 100 cycles is 190 ksi [Reference 4,

Figure I.9-1]. The allowable pressure fluctuation for the second case of Criterion 2 for each unit is:

- PSL Unit 1 = (1/3) x 39.6 x (200/18.8) = 140.4 psi
- PSL Unit 2 = (1/3) x 44 x (190/19.3) = 144.4 psi

Both of these allowable pressure fluctuations are greater than the maximum containment vessel ILRT test pressure of 42.8 psi for PSL Unit 1 and 43.8 psi for PSL Unit 2. Therefore, Criterion 2 is satisfied for both units.

Criterion (3) from Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, and Section N-415 of the 1971 Edition of the ASME Code, Section III describes the allowable temperature difference during normal operation startup and shutdown and is based on a mean operating temperature and S_a. Consistent with the References 1 and 2 fatigue waiver evaluations, the maximum operating range for the PSL Units 1 and 2 containment vessels is conservatively assumed to be 110ºF and should not exceed the value $S_a/2E\alpha$, where E is the material modulus of elasticity and α is the coefficient of thermal expansion.

The maximum allowable number of heatup and cooldown cycles for PSL Units 1 and 2 is 500. The limiting materials that result in the smallest allowable temperature difference for PSL Unit 1 for this criterion are the SA 182 F304 stainless steel forgings and the SA 351 Grade CF8 stainless steel castings. For PSL Unit 2, the SA 240 Type 304 stainless steel plate, SA 182 F304 stainless steel pipe, and SA 351 Grade CF8 stainless steel castings are the limiting materials for this criterion. For PSL Unit 1, the value of S_a for the limiting stainless steel material for 500 cycles is 110 ksi [Reference 3, Figure N-415(A)]. For PSL Unit 2, the value of S_a for the limiting stainless steel material for 500 cycles is 140 ksi [Reference 4, Figure I.9-2]. The allowable temperature difference for Criterion 3 for each unit is:

- PSL Unit 1, $S_{a/}(2E\acute{\alpha})$ = (110,000)/(2 x 29.2 x 9.21) = 204.5°F
- PSL Unit 2, $S_{a/}(2E\acute{\alpha})$ = (140,000)/(2 x 28.3 x 9.21) = 268.6°F

Both of these allowable temperature values exceed the conservative startup and shutdown temperature difference of 110°F. Therefore, Criterion 3 is satisfied for both units.

Criterion (4) from Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, and Section N-415 of the 1971 Edition of the ASME Code, Section III describes that a temperature-difference fluctuation shall be considered to be significant if its total algebraic range exceeds the quantity $S/(2E\acute{\alpha})$ where S is the value of S_a obtained from the applicable design fatigue curve for 10^6 cycles. From Reference 1 and 2, the normal operating temperature of containment is approximately 110°F to 125°F, therefore the temperature difference for normal service is 15°F for both units. The limiting materials for PSL Unit 1 that result in the smallest temperature difference are the SA 182 F304 stainless steel forgings and the SA 351 Grade CF8 stainless steel castings. For PSL Unit 2, the SA 333 Grade 6 carbon steel pipe is the limiting material. For PSL Unit 1, the

value of S_a for the limiting SA 182 F304 and SA 351 Grade CF8 stainless steel materials for 10^6 cycles is 13 ksi [Reference 3, Figure N-415(A)]. For PSL Unit 2, the value of S_a for the limiting SA 333 Grade 6 carbon steel material for 10^6 cycles is also 13 ksi [Reference 4, Figure I.9-1]. The allowable pressure fluctuation for Criterion 4 for each unit is:

- PSL Unit 1, $S_a/(2E\acute{\alpha})$ = (13,000)/(2 x 29.2 x 9.21) = 24.2°F
- PSL Unit 2, $S_a/(2E\acute{\alpha})$ = (13,000)/(2 x 30 x 6.57) = 33°F

Both of these allowable differences exceed the normal service temperature difference of 15°F. Therefore, Criterion 4 is satisfied for both units.

Criterion (5) from Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, and Section N-415 of the 1971 Edition of the ASME Code, Section III describes the allowable temperature difference during normal operation for dissimilar materials. A significant temperature difference is defined as a value greater than Sa/2(E1α1 - E2α2), where Sa is the allowed stress for the weaker material. The dissimilar metal combinations for the PSL Units 1 and 2 containment vessels are 1) carbon steel and stainless steel, 2) carbon steel and nickel alloy, and 3) stainless steel and nickel alloy. The limiting dissimilar material combination based on the lowest allowable temperature difference for PSL Unit 1 and 2 for this criterion is carbon steel and stainless steel. The allowable temperature difference for each unit is:

- PSL Unit 1, $S_a/[2(E_1\acute{\alpha}_1-E_2\acute{\alpha}_2)]$, = 13,000/[(2 (29.2 x 9.21) – (29.9 x 6.44))] = 85.1°F
- PSL Unit 2, $S_a/[2(E_1\acute{\alpha}_1-E_2\acute{\alpha}_2)]$, = 13,000/[(2 (28.3 x 9.21) – (29.9 x 6.2))] = 86.4°F

Both of these allowable temperature differences exceed the normal operating temperature difference of 15°F. Therefore, Criterion 5 is satisfied for both units.

Criterion (6) from Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, and Section N-415 of the 1971 Edition of the ASME Code, Section III describes that the specified full range of mechanical loads, excluding pressure but including pipe reaction loads, does not result in load stresses whose range exceeds S_a. In accordance with References 1 and 2, the only mechanical load fluctuation on the PSL Units 1 and 2 containment vessels during normal operation occurs at the piping penetrations. The process piping for the majority of the PSL Unit 1 piping penetrations is designed in accordance with ANSI B31.7 Class 2 requirements and the process piping for all PSL Unit 2 piping penetrations is designed in accordance with ASME Section III Class 2 requirements. The fatigue evaluation for this process piping is included in SLRA Section 4.3.2, Metal Fatigue of Non-Class 1 Components, and assumes a total of 7000 thermal cycles. The exception to the design requirements above are the four PSL Unit 1 safety injection piping penetrations which are designed in accordance with ANSI B31. 7 Class 1 requirements. The fatigue evaluation for this process piping is included in SLRA Section 4.3.1, Metal Fatigue of Class 1 Components, and utilized design cycles less than 7000.

The limiting material based on the lowest stress difference (S_a minus the maximum membrane stress) for PSL Unit 1 for this criterion is the SB 166 nickel alloy pipe. For Unit 2 the SA 516, Grade 70 carbon steel plate and the SA 350 Grade LF2 carbon steel forgings are the limiting materials. For PSL Unit 1, the value of S_a at 7000 cycles for the SB 166 nickel alloy material is 41 ksi [Reference 3, Figure N-415(B)]. For PSL Unit 2, the value of S_a for 7000 cycles for the limiting SA 516, Grade 70 and the SA 350 Grade LF2 carbon steel materials is 43 ksi [Reference 4, Figure I.9-1]. From Reference 1 and 2, the maximum allowable membrane stress for the penetrations is 1.5 x S_m and are the following for each unit:

- PSL Unit 1, SB 166 material alloy pipe = 1.5 x 18.8 ksi = 28.2 ksi
- PSL Unit 2, SA 516, Grade 70 and SA 350 Grade LF2 materials = 1.5 x 19.3 ksi = 28.95 ksi

Both of these allowable membrane stress values are less than the S_a values of 41 ksi and 43 ksi for PSL Units 1 and 2, respectively. Therefore, Criterion 6 is satisfied for both units.

In summary, the new fatigue waiver evaluations described above for the PSL Units 1 and 2 containment vessel materials of construction demonstrate that all six criteria defined in Article 4, Section N-415 of the 1968 Edition of the ASME Code, Section III, Nuclear Vessels and in Section NB-3232.3 of the 1971 Edition of the ASME Code, Section III are met. Therefore, the following components included in SLRA Table 3.5.2-1, as amended by Supplement 1 dated April 7, 2022, credit the new containment vessel fatigue waiver evaluations and are included in the revised SLRA revisions below:

- airlocks, maintenance hatches and accessories
- construction hatches and covers
- containment vessels
- containment vessel nozzles (electrical)
- containment vessel nozzles (fuel transfer)
- containment vessel nozzles (mechanical)
- nickel alloy containment vessel nozzles (mechanical).

Note that this new fatigue waiver evaluation is also applicable to the current 60-year PEO.

2. SLRA Section 4.6 is revised to state that the new PSL Units 1 and 2 fatigue waiver evaluations include the containment vessel carbon steel, stainless steel, and nickel alloy materials. A summary of the six fatigue waiver criteria is provided for the limiting PSL Units 1 and 2 containment materials.

3. SLRA Sections 3.5.2.2.1.5, 4.6, and Table 19-3 (Appendices A1 and A2) are revised and included below.

References:

1. Calculation PSL-1FSC-01-020 “Steel Containment Fatigue Evaluation, PSL Unit 1,” Revision 0, dated September 18, 2001

2. Calculation PSL-2FSC-01-021 “Steel Containment Fatigue Evaluation, PSL Unit 2,” Revision 0, dated September 18, 2001

3. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Power Plant Component, 1968

4. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Component, 1971 edition

Associated SLRA Revisions:

SLRA Section 3.5.2.2.1.5, page 3.5-24 is revised as follows:

The PSL Units 1 and 2 Containment Vessels are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. For the current renewed licenses, the Containment Vessels' compliance with these design requirements precludes cyclic fatigue cracking that may result in leakage. However, compliance with leakage design criteria to ensure containment integrity is verified through periodic visual examination and testing in accordance with ASME Section XI, Subsection IWE, Inservice Inspection AMP and 10 CFR Part 50, Appendix J AMP.

~~The CLB~~ <u>Fatigue waiver evaluations were prepared</u> for the PSL Units 1 and 2 containment vessels <u>for the current 60-year plant life. However, these fatigue waiver evaluations only addressed the PSL Units 1 and 2 containment vessel plate material fabricated from SA 516, Grade 70 carbon steel. New fatigue waiver evaluations have been prepared for the PSL Units 1 and 2 containment vessels for the 80-year plant life. These new fatigue waiver evaluations eliminate</u> ~~includes fatigue waiver evaluations that preclude~~ the need for detailed containment fatigue analyses. ~~The fatigue waiver evaluations are included in the containment vessel stress reports provided by the supplier. Components included in t~~<u>T</u>he <u>new fatigue waiver evaluations</u> ~~stress reports~~ include the <u>carbon steel, stainless steel, and nickel alloy materials for the PSL Units 1 and 2</u> steel containment vessels, containment vessel penetration nozzles, equipment hatches, and personnel air locks. Fatigue analyses of the mechanical penetration assembly stainless steel expansion bellows were performed by the original equipment supplier. Computation of the bellows cyclic life is included in Appendix 3G of the PSL Unit 1 UFSAR. Note that a previous engineering evaluation concluded this computation of the bellows cyclic life is applicable to the PSL Unit 2 mechanical penetrations. The fatigue analyses for the mechanical penetration process piping are described in SLRA Sections 4.3.1, "Metal Fatigue of Class 1 Components" and 4.3.2 "Metal Fatigue of Non-Class 1 Components". Additional discussion of the fatigue analyses for these containment components is included in SLRA Section 4.6 "Containment Liner Plate, Metal Containments and Penetrations Fatigue."

SLRA Table 3.5-1, page 3.5-51 is revised as follows:

Table 3.5-1: Summary of Aging Management Evaluations for the Containments, Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
3.5-1, 009	Metal liner, metal plate, personnel airlock, equipment hatch, control rod drive (CRD) hatch, penetration sleeves; penetration bellows, steel elements: torus; vent line; vent header; vent line bellows; downcomers, suppression pool shell; unbraced downcomers, steel elements: vent header; downcomers	Cumulative fatigue damage **due to cyclic loading (Only if CLB fatigue analysis exists)**	TLAA, SRP-SLR "Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis"	Yes (SRP-SLR Section 3.5.2.2.1.5)	Consistent with NUREG-2191**.** **~~The CLB~~ Fatigue waiver evaluations were prepared for the PSL Units 1 and 2 containment vessels for the current 60-year PEO. However, these fatigue waiver evaluations only addressed the PSL Units 1 and 2 containment vessel plate material fabricated from SA 516, Grade 70 carbon steel. New fatigue waiver evaluations have been prepared for the PSL Units 1 and 2 containment vessels for the 80-year plant life and eliminate ~~includes fatigue waiver evaluations that preclude~~ the need for detailed containment fatigue analyses. ~~The fatigue waiver evaluations are included in the containment vessel stress reports provided by the supplier. Components included in the~~The new fatigue waiver evaluations ~~stress reports~~ include the carbon steel, stainless steel, and nickel alloy materials for the PSL Units 1 and 2 steel containment vessels, containment vessel penetration nozzles, equipment hatches, and personnel air locks.** **Fatigue analyses of the mechanical penetration assembly stainless steel expansion bellows were performed by the original equipment supplier.** **The fatigue analyses for the mechanical penetration process piping are described in SLRA Sections 4.3.1 and 4.3.2.** **Additional discussion of the fatigue**

Table 3.5-1: Summary of Aging Management Evaluations for the Containments, Structures and Component Supports					
Item Number	Component	Aging Effect/Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
					analyses for these containment components is included in SLRA Section 4.6 "Containment Liner Plate, Metal Containments and Penetrations Fatigue." Further evaluation is documented in Section 3.5.2.2.1.5.

SLRA Table 4.1.5-3, page 4.1-8 is revised as follows:

Table 4.1.5-3
Summary of Results – PSL TLAAs

TLAA Description	Resolution 10 CFR 54.21(c)(1) Section	Section
CONTAINMENT LINER PLATE, METAL CONTAINMENTS AND PENETRATIONS FATIGUE	(ii) projected to the end of ~~remains valid for~~ the SPEO	4.6
OTHER PLANT-SPECIFIC TLAA		4.7
Leak-Before-Break of Reactor Coolant System Loop Piping	(ii) projected to the end of the SPEO	4.7.1
Alloy 600 Instrument Nozzle Repairs	(i) remains valid for the SPEO	4.7.2
Unit 1 Core Support Barrel Repairs	(ii) projected to the end of the SPEO	4.7.3
Reactor Coolant Pump Flywheel Fatigue Crack Growth	(i) remains valid for the SPEO	4.7.4
Reactor Coolant Pump Code Case N-481	(i) remains valid for the SPEO	4.7.5
Crane Load Cycle Limits	(i) remains valid for the SPEO	4.7.6
Flaw Tolerance Evaluation for CASS RCS Piping Components	(ii) projected to the end of the SPEO	4.7.7
Unit 2 Structural Weld Overlay PWSCC Crack Growth Analyses	(iii) the effects of aging on the intended function will be adequately managed for the SPEO	4.7.8

SLRA Section 4.6, pages 4.6-1 through 4.6-3 are revised as follows:

4.6 CONTAINMENT LINER PLATE, METAL CONTAINMENTS AND PENETRATIONS FATIGUE

Metal Containment Fatigue

TLAA Description

As stated in Sections 3.8.2.1.5 and 3.8.2.6.1 of the PSL Unit 1 and 2 UFSARs, respectively, the containment vessels are fabricated from welded ASME-SA 516 Grade 70 steel plates to provide an essentially leak-tight barrier. Design criteria applied to the steel containment vessels assure that the specified leak rate is not exceeded under the design basis accident conditions. The PSL Unit 1 containment vessel is designed in accordance with the 1968 Edition of ASME Boiler and Pressure Vessel Code, Section III (Reference 4.8.20) and the PSL Unit 2 containment vessel is designed in accordance with the 1971 Edition of ASME Boiler and Pressure Vessel Code, Section III (Reference 4.8.21). ~~The CLB~~ **Fatigue waiver evaluations were prepared** for the PSL Unit 1 and 2 containment vessels **for the current 60-year plant life. However, these fatigue waiver evaluations only addressed the PSL Units 1 and 2 containment vessel plate material fabricated from SA 516, Grade 70 carbon steel. New fatigue waiver evaluations have been prepared for the PSL Units 1 and 2 containment vessels for the 80-year plant life. These new fatigue waiver evaluations eliminate** ~~includes fatigue waiver evaluations that preclude~~ the need for a detailed **containment** fatigue analysis. **The fatigue waiver** ~~**analysis**~~ **evaluations for the 80-year plant life includes** ~~s~~ **carbon steel, stainless steel, and nickel alloy materials for the PSL Units 1 and 2 steel containment**~~**s**~~ **vessel shells, containment penetration nozzles, personnel air locks, and equipment hatches. The** ~~**PSL Unit 1**~~ **fatigue waiver evaluation for PSL Unit 1 was performed in accordance with Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, Section N-415, "Analysis for Cyclic Operation". The fatigue waiver evaluation for PSL Unit 2 was performed in accordance with the 1971 Edition of ASME Section III, Nuclear Vessels, Section NB-3222.4, "Analysis for Cyclic Operation."** Fatigue waivers that consider transient cycles which occur over the life of the plant constitute TLAAs.

TLAA Evaluation

~~The~~ **For the 80-year plant life, new fatigue waiver evaluations have been prepared for the** ~~original~~ PSL Unit**s** 1 and 2 ~~demonstrated that the fatigue waiver was applicable to both~~ containment vessels. The fatigue waiver **evaluations prepared** for the ~~PSL Unit 1 and 2 containment vessels remain valid through the~~ 80-year **plant life** ~~SPEO~~ **include the following containment vessel materials as specified in Tables 3.8-5 and 3.8-10 of the PSL Unit 1 and 2 UFSARs, respectively**~~since the service loading of the vessels meet all the following six fatigue waiver conditions of Section III of the ASME Code~~.

PSL-1 Containment Vessel Materials

Material	Specification	Allowable Tensile Strength (psi)
Plate	SA 516, Gr 70 to SA 300	17,500
Forgings	SA 350, Gr LF2	17,500
	SA 182, F304	13,750
Pipe	SA 333, Gr 1	13,750
	SB 167	18,200
	SB 166	18,800
	SA 516, Gr 70 to SA 300	17,500
Castings	SA 352, Gr LCB	16,250
	SA 351, Gr B8	13,750

PSL-2 Containment Vessel Materials

Material	Specification	Design Stress Intensity (psi)
Plate	SA 516, Gr 70	19,300
	SA 240, Type 304	17,300 (Note 1)
Forgings	SA 350, Gr LF 1	16,500
	SA 350, Gr LF 2	19,300
	SA 182, F304	18,300
Pipe	SA 333, Gr 6	16,500
	SB 168	20,700
	SB 167	20,700
	SB 166	20,700
	SA 516, Gr 70	19,300
Castings	SA 352, Gr LC1	16,200
	SA 351, Gr CF8	16,000

Note 1 – there is no design stress intensity value for the SA 240, Type 304 material in Table 3.8-10 of the PSL Unit 2 UFSAR. However, the original containment vessel stress report does include the specified value for that material.

A summary of the results addressing the six (6) fatigue waiver criteria for the limiting PSL Units 1 and 2 containment vessel materials is summarized below.

Criterion 1) Atmospheric to operating pressure cycles

The limiting material for this criterion based on the lowest number of allowable cycles for PSL Unit 1 is the SB 166 nickel alloy pipe. For PSL Unit 2 the limiting materials are the SA 516, Grade 70 carbon steel plate and the SA 350 Grade LF2 carbon steel forgings.

For the limiting material for PSL Unit 1, the value of S_a is equal to $3S_m$, or 56.4 ksi, and the allowable number of cycles (N) is 3000. For the limiting materials for PSL Unit 2, the value of S_a is equal to $3S_m$, or 57.9 ksi and the allowable number of cycles (N) is 2500.

As 500 operating pressure cycles are less than both 3000 cycles for PSL Unit 1 and 2500 cycles for PSL-2, Criterion 1 is satisfied for both units.

~~The Unit 2 containment vessel allowable stress is:~~

- ~~Sa= 3Sm = 3 x 19,300 psi = 57,900 psi~~

~~which corresponds to approximately 3000 allowable cycles (Reference 4.8.21) and is bounding for both units. As stated in Section 4.3.1, the maximum allowable number of heatup/cooldown cycles for the PSL Unit 1 and 2 RCS is 500. Therefore, this fatigue waiver requirement is satisfied for both PSL Unit 1 and 2 for the SPEO.~~

Criterion 2) Normal service pressure fluctuations

~~The allowable pressure fluctuation (design pressure x (1/3)(S/Sm)) is:~~

- ~~39.6 psig x (1/3)(13,000 psi /17,500 psi) = 9.81 psi for the Unit 1 containment vessel and,~~
- ~~44 psig x (1/3)(13000 psi / 19,300 psi) = 9.88 psi for the Unit 2 containment vessel~~

~~The normal operating pressure of both containment vessels is essentially constant with little and no fluctuation. Thus, the normal service pressure fluctuation requirement is satisfied.~~

The significant pressure fluctuations **for the PSL Units 1 and 2 containment vessels** are from integrated leak rate tests (ILRT**s**) ~~of the containment vessels which are typically performed every 15 years~~. **The limiting material for the ILRT pressure fluctuation for PSL Unit 1 is the SB 166 nickel alloy pipe. The limiting materials for PSL Unit 2 are the A 516, Grade B carbon steel plate and the SA 350 Grade LF2 carbon steel forgings. The values of S_a for the limiting materials for 100 cycles is 200 ksi for PSL Unit 1 and 190 ksi for PSL Unit 2.**

The allowable pressure fluctuation is equal to the (design pressure x (1/3)(S_a/Sm)) and for each unit is:

- **PSL Unit 1 = (1/3) x 39.6 x (200/18.8) = 140.4 psi**
- **PSL Unit 2 = (1/3) x 44 x (190/19.3) = 144.4 psi**

Both of these allowable pressure fluctuations are greater than the maximum containment vessel ILRT test pressure of 42.8 psi for PSL Unit 1 and 43.8 psi for PSL Unit 2. Therefore, Criterion 2 is satisfied for both units.

~~Conservatively assuming 100 ILRTs for the 80-year SPEO, the bounding PSL Unit 2 allowable pressure fluctuation is:~~

- ~~44 psig x (1/3)(190,000 psi / 19,300 psi) = 144 psi > 44 psig~~

~~Therefore, this fatigue waiver requirement is satisfied for both PSL Unit 1 and 2 for the SPEO.~~

Criterion 3) Temperature difference - startup and shutdown

The maximum temperature range for startup and shutdown is conservatively assumed to be 40°F to 150°F, or 110°F and shall not exceed Sa/(2Eα). The limiting materials that result in the smallest allowable temperature difference for PSL Unit 1 for this criterion are the SA 182 F304 stainless steel forgings and the SA 351 Grade CF8 stainless steel castings. For PSL Unit 2 the limiting materials are SA 240 Type 304, SA 182 F304 and SA 351 Gr CF8. The value of S_a for 500 cycles for PSL Unit 1 is 110 ksi and the value for PSL Unit 2 is also 110 ksi. The allowable temperature difference for Criterion 3 for each unit is: ~~, which is~~:

- PSL Unit 1, S_a/(2Eά) = (110,000)/(2 x 29.2 x 9.21) = 204.5°F ~~110,000 / ((2)(30 x 10^6)(6.44 x 10^{-6})) = 284°F for Unit 1, and~~
- PSL Unit 2, S_a/(2Eά) = (140,000)/(2 x 28.3 x 9.21) =268.6°F ~~110,000 / ((2)(30 x 10^6)(6.57 x 10^{-6})) = 279°F for Unit 2~~

Both of these allowable temperature values exceed the conservative startup and shutdown temperature difference of 110°F. Therefore, Criterion 3 is satisfied for both units. ~~Since the PSL maximum temperature range of 110°F (150°F to 40°F) is smaller than Sa/(2Eα), this fatigue waiver requirement is satisfied for both PSL Unit 1 and 2 for the SPEO~~.

Criterion 4) Temperature difference - normal service

The normal operating temperature of PSL containment vessels is approximately 110°F to 125°F, or ~~. This temperature fluctuation of~~ 15ºF for both units ~~is considered negligible~~. The limiting materials for PSL Unit 1 that result in the smallest temperature difference are the SA 182 F304 stainless steel forgings and the SA 351 Grade CF8 stainless steel castings. For PSL Unit 2 the SA 333 Grade 6 carbon steel pipe is the limiting material. For PSL Unit 1, the value of S_a for 10^6 cycles is 13 ksi and for PSL Unit 2 is also 13 ksi. The allowable pressure fluctuation for Criterion 4 for each unit is:

- PSL Unit 1, S_a/(2Eά) = (13,000)/(2 x 29.2 x 9.21) = 24.2°F
- PSL Unit 2, S_a/(2Eά) = (13,000)/(2 x 30 x 6.57) = 33°F

Both of these allowable differences exceed the normal service temperature difference of 15°F. Therefore, Criterion 4 is satisfied for both units ~~Thus, this fatigue waiver requirement is satisfied for both PSL Unit 1 and 2 for the SPEO~~.

Criterion 5) Temperature difference - dissimilar materials

The dissimilar metal combinations for the PSL Units 1 and 2 containment vessels are 1) carbon steel and stainless steel, 2) carbon steel and nickel alloy, and 3) stainless steel and nickel alloy. During normal operation, components fabricated from materials of differing elastic moduli~~us~~ and/or coefficient of thermal expansion may not fluctuate more than (Sa)/[2(E1α1 - E2α2)]. The limiting dissimilar metal combination for both the PSL containment vessels is

carbon steel and SS. ~~This range is therefore:~~ **The allowable temperature difference for each unit is:**

- **PSL Unit 1, $S_a/[2(E_1\acute{\alpha}_1-E_2\acute{\alpha}_2)]$, = 13,000/[(2 (29.2 x 9.21) – (29.9 x 6.44))] = 85.1 °F**
- **PSL Unit 2, $S_a/[2(E_1\acute{\alpha}_1-E_2\acute{\alpha}_2)]$, = 13,000/[(2 (28.3 x 9.21) – (29.9 x 6.2))] = 86.4 °F**

Both of these allowable temperature differences exceed the normal operating temperature difference of 15°F. Therefore, Criterion 5 is satisfied for both units.

- ~~13,000 / ((2)(30.5)(7.3 x 10^-6) – (27.4)(6.25 x 10^-6)) = 126.5 °F~~

~~This is greater than the 15ºF range of temperature fluctuation during normal operation of units. Thus, this fatigue waiver requirement is satisfied for both PSL Unit 1 and 2 for the SPEO.~~

Criterion 6) Mechanical loads

The only mechanical load fluctuation on the containment vessels associated with normal operation occurs at the piping penetrations. The PSL piping penetrations are qualified for 7000 cycles (Appendix 3G of the Unit 1 UFSAR and Section 3.8 of the Unit 2 UFSAR). **The limiting material for PSL Unit 1based on the lowest stress difference (S_a minus the maximum membrane stress) is the SB 166 nickel alloy pipe. For Unit 2 the limiting materials are the SA 516, Grade 70 carbon steel plate and the SA 350 Grade LF2 carbon steel forgings. The value of S_a at 7000 cycles for PSL Unit 1 is 41 ksi for PSL Unit 2 the value is 43 ksi. The maximum allowable membrane stress for the penetrations is 1.5 x S_m and are:**

- **PSL Unit 1, 1.5 x 18.8 ksi = 28.2 ksi**
- **PSL Unit 2, 1.5 x 19.3 ksi = 28.95 ksi**

Both of these allowable membrane stress values are less than the S_a values of 41 ksi and 43 ksi for PSL Units 1 and 2, respectively. Therefore, Criterion 6 is satisfied for both units ~~The Sa at 7000 cycles is approximately 43,000 psi (Reference 4.8.20, Fig. N-415 (a) and Reference 4.8.21, Fig. I-9.0). This value is much greater than allowable membrane stress values (17,500 psi for the Unit 1 containment vessel and 19,300 psi for the Unit 2 containment vessel). Therefore, this fatigue waiver requirement is satisfied for both PSL Unit 1 and 2 for the SPEO~~.

TLAA Disposition: 10 CFR 54.21(c)(1)(i1)

Based on the above ~~evaluation~~, the **new** fatigue waiver **evaluations** for the PSL Unit 1 and 2 containment vessels **have been projected to the end of** ~~remain valid through~~ the SPEO in accordance with 10 CFR 54.21(c)(1)(i**i**).

SLRA Section A-1, page A1-55 is revised as follows:

19.3.5 Containment Liner Plate, Metal Containments, and Penetrations Fatigue

Metal Containment Fatigue

The PSL Unit 1 containment vessel is fabricated from welded ASME-SA 516 Grade 70 steel plates to provide an essentially leak-tight barrier. Design criteria applied to the steel containment vessels assure that the specified leak rate is not exceeded under the design basis accident conditions. The PSL Unit 1 containment vessel is designed in accordance with the 1968 Edition of ASME Boiler and Pressure Vessel Code, Section III. ~~The CLB~~ **A new fatigue waiver evaluation has been prepared** for the PSL Unit 1 containment vessel **for the 80-year plant life. This new fatigue waiver evaluation** ~~includes a fatigue waiver evaluation~~ **was** **performed in accordance with Article 4 of the 1968 Edition of ASME Section III, Nuclear Vessels, Section N-415, "Analysis for Cyclic Operation"** ~~that~~ **and** precludes the need for a detailed **containment** fatigue analysis. **The fatigue waiver evaluation prepared for the 80-year plant life includes the fabrication materials for the steel containment~~s~~ vessel shell, containment penetration nozzles, personnel air locks, and equipment hatches.**

~~A common~~ **The new fatigue waiver** evaluation **for** ~~of~~ the PSL Unit~~s~~ 1 ~~and 2~~ containment vessel ~~fatigue waivers was performed for SLR and~~ concludes~~d~~ that ~~the fatigue waivers remain valid through the 80-year SPEO since the service loading of the vessels met all~~ the following six fatigue waiver conditions of the applicable ASME Codes **have been satisfied**:

1) Atmospheric to operating pressure cycles
2) Normal service pressure fluctuations
3) Temperature difference - startup and shutdown
4) Temperature difference - normal service
5) Temperature difference - dissimilar materials
6) Mechanical loads

Therefore, the **new** fatigue waiver **evaluation** for the PSL Unit 1 containment vessel **has been projected to the end of** ~~remains valid through~~ the SPEO in accordance with 10 CFR 54.21(c)(1)(i**i**).

SLRA Section A-2, pages A2-56 through A2-57 are revised as follows:

19.3.5 Containment Liner Plate, Metal Containments and Penetrations Fatigue

Metal Containment Fatigue

The PSL Unit 2 containment vessel is fabricated from welded ASME-SA 516 Grade 70 steel plates to provide an essentially leak-tight barrier. Design criteria applied to the steel containment vessels assure that the specified leak rate is not exceeded under the design basis accident conditions. The PSL Unit 2 containment vessel is

designed in accordance with the 1971 Edition of ASME Boiler and Pressure Vessel Code, Section III. ~~The CLB~~ **A new fatigue waiver evaluation has been prepared** for the PSL Unit 2 containment vessel **for the 80-year plant life. This new fatigue waiver evaluation** ~~includes a fatigue waiver evaluation~~ **was performed in accordance with the 1971 Edition of ASME Section III, Nuclear Vessels, Section NB-3222.4, “Analysis for Cyclic Operation”** ~~that~~ **and** precludes the need for a detailed **containment** fatigue analysis. **The fatigue waiver evaluation prepared for the 80-year plant life includes the fabrication materials for the steel containments vessel shell, containment penetration nozzles, personnel air locks, and equipment hatches.**

~~A common~~ **The new fatigue waiver evaluation for** ~~of~~ the PSL Unit~~s 1 and~~ 2 containment vessel ~~fatigue waivers was performed for SLR and~~ conclude**s**~~d~~ that ~~the fatigue waivers remain valid through the 80-year SPEO since the service loading of the vessels meet all~~ the following six fatigue waiver conditions of the applicable ASME Codes **have been satisfied**:

1) Atmospheric to operating pressure cycles
2) Normal service pressure fluctuations
3) Temperature difference - startup and shutdown
4) Temperature difference - normal service
5) Temperature difference - dissimilar materials
6) Mechanical loads

Therefore, the **new** fatigue waiver **analysis** for the PSL Unit 2 containment vessel **has been projected to the end of** ~~remains valid through~~ the SPEO in accordance with 10 CFR 54.21(c)(1)(i**i**).

Associated Enclosures:

None.

Further Evaluation Section 3.5.2.2.2.6 – Neutron Fluence Uncertainty and Conservatism

RAI 3.5.2.2.2.6-1

<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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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>

Subsequent License Renewal Application (SLRA) Section 3.5.2.2.2.6 (and similarly for SLRA Section 3.5.2.2.2.7), as amended by Supplement 2 dated April 13, 2022 (ADAMS Accession No. ML22103A014) describes the applicant's further evaluation for reduction of concrete strength (and equally applicable to SLRA Section 3.5.2.2.2.7 for loss of fracture toughness due to irradiation embrittlement of the reactor pressure vessel (RPV) structural steel support assemblies) of the St. Lucie Nuclear Plant (PSL) units. Specifically, SLRA Page Numbers: 3.5-35, 3.5-38, 3.5-39, 3.5-40, B-251, and the discussion contained in L-2022-044 Attachment 1 Page 3 of 15 address analytical uncertainties associated with fast neutron (E > 1.0 MeV) fluence at various locations. This is applicable to both steel components and concrete structures.

<u>Issue</u>

Specifically, the statement "As part of this analysis, numerous parameters that were identified as having a potentially significant contribution to the core neutron source, reactor geometry, coolant temperature, discretization, and modeling approximation uncertainties at the RPV inner and outer surfaces were evaluated," is not detailed enough for the staff to conclude that were acceptably evaluated. The estimate of 20% (or 25% for some components) fluence uncertainty for the primary shield wall and RPV structural components is a conservate estimate based this evaluation. It is not clear to the staff how this evaluation makes the safety conclusion that 20% (or 25% for some components) is sufficiently conservative when projecting fluence estimates to the end of the renewed operating licenses.

Request

1. Clarify how this evaluation makes the safety conclusion that 20% (or 25% for some components) is sufficiently conservative when projecting fluence estimates to the end of the renewed operating licenses.
2. Identify necessary updates of pertinent areas of the SLRA to reflect this clarification or justify your position for not updating.

PSL Response

An analytical uncertainty analysis associated with the neutron fluence and iron atom displacement (dpa) results for the RPV support structure was not performed for the St. Lucie SLRA. Therefore, a conservative estimate of the uncertainty associated with these results was established using the RPV extended beltline uncertainty analysis described in WCAP-18124-NP-A Revision 0 Supplement 1-P (Reference 1). Note that the level of detail in the model used for the extended beltline uncertainty analysis is commensurate with the plant-specific model for St. Lucie. For example, the mesh sizes, treatment of anisotropic scattering, angular quadrature, modeling of internals structures, etc., are similar.

The RPV extended beltline analysis described in (Reference 1) quantified the analytical uncertainty associated with calculated fast neutron (E > 1.0 MeV) fluence rates at the RPV inner and outer surfaces at various elevations above and below the active fuel. As part of this analysis, numerous parameters that were identified as having a potentially significant contribution to the core neutron source, reactor geometry, coolant temperature, discretization, and modeling approximation uncertainties at the RPV inner and outer surfaces were evaluated. More detailed information regarding these parameters may be found in Section 3 of (Reference 1) and Section 4.4 of WCAP-18124-NP-A (Reference 2).

Each parameter identified was evaluated on an individual basis by determining the maximum relative change in the base-case fluence rate that occurred as the magnitude of that parameter was varied over a bounding range of values. The net analytical uncertainty associated with a given RPV location was then determined by taking the root sum of squares of the individual parameter uncertainty values determined at that location. Given the parameters considered, the magnitudes of the parameter variations evaluated, and the relative proximity of the RPV outer surface to the RPV support structure, the extended beltline uncertainty analysis results for the RPV outer surface were judged to provide a reasonable basis for estimating the analytical uncertainty associated with the RPV support structure neutron exposures.

The maximum neutron fluence and iron atom displacement projections at the RPV support columns and horizontal support bottoms occur at elevations that are within 3 feet of the core midplane. However, since the extended beltline uncertainty analysis was, by design, focused on the RPV extended beltline region only, it did not consider axial elevations within 3 feet of the core midplane; the elevations nearest the midplane considered were 30 cm above the top and 30 cm below the bottom of the active fuel. Therefore, the extended beltline uncertainty analysis results determined at the RPV outer surface 30 cm above the top of the active fuel were used as the starting point for estimating the uncertainty associated with the RPV support columns and horizontal support bottoms. This is conservative because (1) analytical uncertainties at the top of the active fuel are greater than the ones near the core midplane, and (2) analytical uncertainties increase with axial distance above the top of the active fuel.

The maximum neutron fluence and iron atom displacement projections at the top of the 6-inch plate under the RPV nozzle foot occur at an elevation that is less than 2 ft above the top of the active fuel. Therefore, the extended beltline uncertainty analysis results determined at the RPV outer surface 90 cm above the top of the active fuel were used for the top of the 6-inch plate under the RPV nozzle foot. This is conservative because analytical uncertainties increase with axial distance above the top of the active fuel.

In addition to using these bounding RPV locations as starting points, the concrete composition parameter uncertainty values determined at these locations were increased by a factor of 2. These values were increased because they were associated with the one parameter evaluated in the RPV extended beltline analysis whose uncertainty was judged to be potentially impacted in a non-negligible manner if a detailed uncertainty analysis for the RPV support structure were performed. Note that the standard concrete composition from the BUGLE-96 documentation (Reference 3) was used for the primary shield wall (PSW) in both the extended beltline analytical uncertainty analysis base-case calculations and the St. Lucie neutron fluence and iron atom displacement calculations.

Following this process, the analytical uncertainty associated with the neutron fluence and iron atom displacement results for the RPV support columns and horizontal support bottoms was conservatively estimated to be 20%; the analytical uncertainty for the top of the 6-inch plate under the RPV nozzle foot was estimated to be 25%. Applying the estimated analytical uncertainties in the limiting direction to the maximum exposures determined for the RPV support columns, horizontal support bottoms, and top of the 6-inch plate under the RPV nozzle foot is considered to be sufficiently conservative when projecting RPV support structure neutron exposures to the end of the renewed operating licenses for the following reasons:

- The guidance provided in the NUREG-0933 (Reference 4) GSI-15 resolution is to utilize Figure 3-1 of NUREG-1509 [5] to calculate the change in RPV support structure nil-ductility transition temperature (ΔNDTT) based on iron atom displacements from neutrons with energies greater than 0.1 MeV. However, the RPV support iron atom displacement exposures listed in Table 3.5.2.2-3 and Table 3.5.2.2-4 of Section 3.5.2.2.2.7 of Enclosure 3 of the St. Lucie SLRA (Reference 6) include the contribution from neutrons with energies less than 0.1 MeV. Excluding the contribution from these lower energy neutrons would:
    - reduce the iron atom displacement values for the support columns and horizontal support bottoms by approximately 8%, and
    - reduce the iron atom displacement values for the top of the 6-inch plate under the RPV nozzle foot by approximately 16%.
- The fracture mechanics analysis summarized in WCAP-18623-P/NP (Reference 7) included an iron atom displacement exposure uncertainty of 25% in the embrittlement calculation. However, given the elevations at which they were determined (i.e., elevations within 3 feet of the core midplane), the embrittlement calculations performed for the support columns and horizontal support bottoms only needed to include an iron atom displacement exposure uncertainty of 20%.

- Neutron exposure projections for the St. Lucie Unit 1 RPV support structure were based on the core power distributions and operating conditions of Unit 1 Cycle 29 but included a +10% bias on the peripheral and re-entrant corner assembly relative powers; projections for Unit 2 were based on Cycle 24 and also included a +10% bias on the peripheral and re-entrant corner assembly relative powers. A +10% bias applied to the peripheral and re-entrant corner assembly relative powers of a given cycle increases the calculated exposure rates for that cycle by approximately 10%. While the biases used for the projection cycles were intended to account for changes in future-cycle exposure rates that are expected to occur as a result of normal variations in future-cycle core designs, they also provide a significant source of margin in the 72-EFPY exposures determined for the SLRA. This is because future cycle designs at St. Lucie Unit 1 would not be expected to consistently result in exposure rates 10% greater than the ones determined for Cycle 29, and future cycle designs at St. Lucie Unit 2 would not be expected to consistently result in exposure rates 10% greater than the ones determined for Cycle 24. This conclusion is supported by the RPV neutron exposure rates reported in LTR-REA-21-1-NP, Revision 1 (Reference 8) for Unit 1 Cycles 25–30 and LTR-REA-21-2-NP (Reference 9) for Unit 2 Cycles 20–25, which show normal cycle-to-cycle variations, but no long-term trend in either the limiting (increasing) or non-limiting (decreasing) direction.

Finally, consistent with the guidance of ASTM E2956-21 (Reference 10), any changes in fuel design, fuel management, plant operating conditions, or plant configuration that could consistently alter the neutron exposures of the RPV (and as a result, neutron exposures of the RPV support structure) will be evaluated prior to implementation. Examples of these types of changes include switching from low-leakage to out-in core loading patterns, transitioning from 18-month to 24-month cycles, performing a power uprate, and modifying the reactor internals geometry. Similarly, the neutron exposures of the RPV will be updated whenever new measurement data are available. For example, the exposures will be updated whenever a surveillance capsule is withdrawn from the reactor for analysis as part of the reactor vessel material surveillance program described in Section B.2.3.19 of Enclosure 3 of Reference 6.

References:

1. Westinghouse Letter LTR-NRC-20-69, Revision 0, "Submittal of WCAP-18124-NP-A Revision 0 Supplement 1-P and WCAP-18124-NP-A Revision 0 Supplement 1-NP, 'Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials,' Revision 0 (Proprietary/Non-Proprietary)," December 2020. (ADAMS Accession No. ML20344A386)
2. Westinghouse Report WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018. (ADAMS Accession No. ML18204A010)
3. RSICC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Shielding Information Computational Center, Oak Ridge National Laboratory (ORNL), July 1999.
4. USNRC Report NUREG-0933, "Resolution of Generic Safety Issues (Formerly entitled "A Prioritization of Generic Safety Issues)," Main Report with Supplements 1–34. Generic Issue No: 15, "Radiation Effects on Reactor Vessel Supports (Rev. 3)."
5. USNRC Report NUREG-1509, "Radiation Effects on Reactor Pressure Vessel Supports," May 1996. (ADAMS Accession No. ML20112B249)

6. Florida Power & Light Company Letter L-2021-142, “Application for Subsequent Renewed Facility Operating License,” August 2021. (ADAMS Accession Package No. ML21215A314)
7. Westinghouse Report WCAP-18623-P/NP, Revision 1, “St. Lucie Units 1 & 2 Subsequent License Renewal: Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports,” December 2021.
8. Westinghouse Letter LTR-REA-21-1-NP, Revision 1, “St. Lucie Units 1 & 2 Subsequent License Renewal: Unit 1 Reactor Vessel, Vessel Support, and Bioshield Concrete Exposure Data,” May 2021. (ADAMS Accession No. ML21215A320)
9. Westinghouse Letter LTR-REA-21-2-NP, Revision 1, “St. Lucie Units 1 & 2 Subsequent License Renewal: Unit 2 Reactor Vessel, Vessel Support, and Bioshield Concrete Exposure Data,” June 2021. (ADAMS Accession No. ML21215A320)
10. ASTM International Standard E2956, 2021, “Standard Guide for Monitoring the Neutron Exposure of LWR Reactor Pressure Vessels,” ASTM International, West Conshohocken, PA, 2021, DOI: 10.1520/E2956-21, www.astm.org.
11. Westinghouse Letter, LTR-SDA-21-021-P/NP, Revision 2, “St. Lucie Units 1&2 Subsequent License Renewal: Reactor Pressure Vessel Supports Assessment,” September 2021.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Further Evaluation Section 3.5.2.2.2.6 – RPV Sliding Plates and Lubrication

RAI 3.5.2.2.2.6-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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) Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022 (ADAMS Accession No. ML22103A014) describes the applicant's further evaluation for loss of fracture toughness due to irradiation embrittlement of the reactor pressure vessel (RPV) structural steel support assemblies of the St. Lucie Nuclear Plant (PSL) units. Specifically, SLRA Page Numbers: 3.5-44, 3.5-45, 3.5-46, and the discussion contained in L-2022-044 Attachment 2 Page 1 of 10 addresses impacts to the RPV sliding plates.

Issue

The RPV sliding plates are identified as ASTM B-22 Alloy E, a manganese bronze alloy, lubricated with Lubrite type AE-2 lubricant. There is not an identified radiation embrittlement basis provided for this manganese bronze alloy component. Additionally, the basis for inclusion of gamma ray heating that could increase the temperature for the assembly, of this manganese bronze alloy component and for its Lubrite lubricant is not clearly identified as it relates to the 300°F.

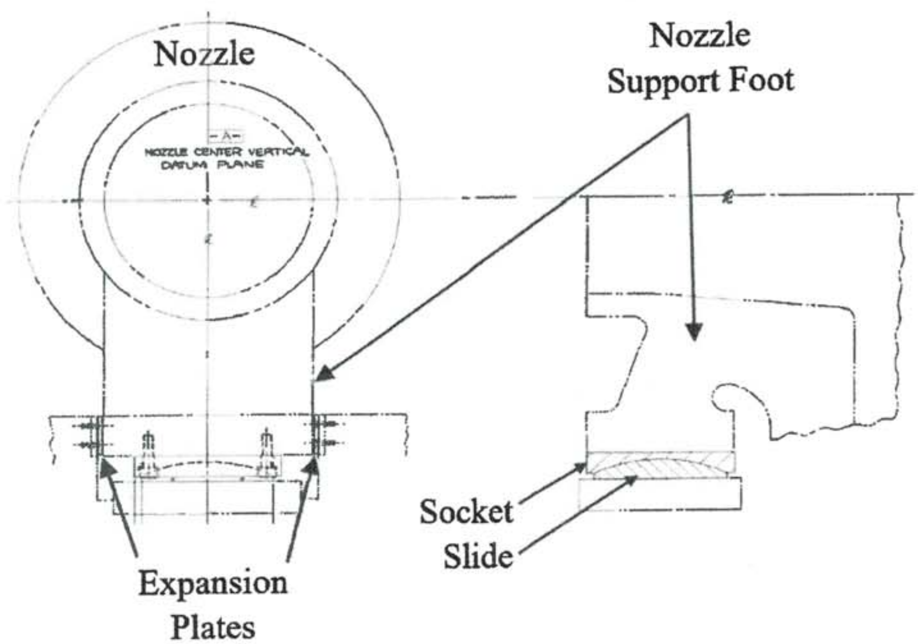
Request

1. Identify the radiation embrittlement basis provided for this manganese bronze alloy component.
2. Clarify whether the manganese bronze component and its Lubrite lubricant has been evaluated for inclusion of gamma ray heating.
3. Identify necessary updates of pertinent areas of the SLRA reflecting this RAI input or justify PSL position for not updating.

PSL Response

The numbered responses below correspond to the numbered requests above.

1. The RPV slide (circular convex plate) and expansion plates are ASTM B-22 Alloy E, a manganese bronze alloy. The socket and slide design allows the thermal expansion and contraction of the RPV during heatup and cooldown. As illustrated in Figure 1 below, the slide experiences compressive loads during normal operating conditions, and friction forces at the socket-slide interface. Similarly, the expansion plates on the side only experience compressive loads and friction forces during thermal expansion of the RPV. One of the key factors for stress corrosion cracking (SCC) to occur is a high level of sustained tensile stress. Since the slide and expansion plates are mainly in compression, SCC or any other crack growth mechanism is unlikely to occur. Even if a crack were to occur in the slide or expansion plates, they will likely remain in place and continue to perform their intended function. Therefore, an analysis of the radiation embrittlement of the manganese bronze alloy for the slide and expansion plates is unnecessary.



- Not to Scale -

Figure 1: St. Lucie Units 1 and 2 RPV Support Shoe Socket and Slide at Nozzle Support

2. Three-dimensional finite element thermal analyses were performed for the St. Lucie Units 1 and 2 reactor cavities under EPU conditions with gamma heating. Detailed gamma heat fluxes were calculated using explicit SORCERY and TORT calculations. These gamma heating fluxes were read into the ANSYS simulation as local, spatially dependent heat sources. The lubricant film was not explicitly modeled. The thin film of

lubricant has negligible effect on heat transfer analysis. The analysis results confirmed the base of lubricated slide plate remained below the required 300°F limit.

3. As discussed in responses 1 and 2, an evaluation of the radiation embrittlement for the manganese bronze components is unnecessary, and thermal analysis with gamma heating confirmed the 300°F limit was not exceeded. Therefore, no update to the SLRA is needed regarding RAI 3.5.2.2.2.6-2.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Further Evaluation Section 3.5.2.2.2.7 – RPV Support Partial Penetration Weldments

RAI 3.5.2.2.2.7-1

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022 (ADAMS Accession No. ML22103A014) describes the applicant's further evaluation for loss of fracture toughness due to irradiation embrittlement of the reactor pressure vessel (RPV) structural steel support assemblies of the St. Lucie Nuclear Plant (PSL) units. Specifically, SLRA Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022, refers to the Westinghouse report, LTR-SDA-21-021-NP, Revision 2 (Enclosure 4, Attachment 3 to the SLRA, ADAMS Accession No. ML21285A112) for a qualitative assessment of the RPV structural steel support assemblies and presents the results of this assessment at the most limiting locations in the assemblies shown in SLRA Table 3.5.2.2-5, as amended by Supplement 2 dated April 13, 2022. Sections 6.2 and 7.2 of LTR-SDA-21-021-NP, Revision 2 discusses the stresses determined through finite element analysis (FEA) and used for the results of the qualitative assessment. SLRA Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022, also refers to WCAP-18623-NP, "St. Lucie Units 1 & 2 Subsequent License Renewal: Fracture Mechanics Assessment of Reactor Pressure Vessel, Structural Steel Supports," Revision 1 (ADAMS Accession No. ML22103A133) for the plant-specific fracture mechanics assessment of the RPV structural steel support assemblies of the PSL units. Section 3 of WCAP-18623-NP, Revision 1 states that some of the weldments that join the plates in the RPV structural steel support assemblies of the PSL units are partial penetration weldments.

Issue

Based on its audit and the review of SLRA Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022, Sections 6.2 and 7.2 of LTR-SDA-21-021-NP, Revision 2, and WCAP-18623-NP, Revision 1, the staff noted that these documents did not include information regarding the potential impact of partial penetration weldments (i.e., fillet or groove welds) in the RPV structural steel assemblies on the stresses used for the results of the qualitative assessment at the limiting locations shown in SLRA Table 3.5.2.2-5, as amended by Supplement 2 dated April 13, 2022.

Request

Discuss the potential impact of partial penetration weldments (i.e., fillet or groove welds) in the RPV structural steel assemblies on the stresses used for the results of the qualitative assessment at the limiting locations shown in SLRA Table 3.5.2.2-5, as amended by Supplement 2 dated April 13, 2022. If the potential impact was determined to be small or none, explain how this determination was made (for example, explain any sensitivity studies or test cases in the FEA to determine the impact) and update the SLRA as necessary.

PSL Response

When constructing the finite element (FE) model for the St. Lucie reactor vessel supports, all partial penetration groove weld connection locations were modelled as fully bonded connections, essentially modelling full penetration welds at those locations. Use of a fully bonded connection is desirable because it lowers the complexity and runtime of the models and mitigates some aspects of the existence of singularities at the re-entrant corners that are common to such connections. This modelling decision was acceptable for the following reasons.

1. Early in the development of the model, a test run was completed which replaced the bonded connections with line-to-line bonds at the edges of the interfacing plates. This would be equivalent to bonding only at the root locations of the welds, in one dimension, along the length of the welds. This test run showed no appreciable difference in the stiffness of the structure compared to the face-to-face fully bonded connection model. This indicates that distribution of loads throughout the model was not impacted by the fully bonded connection model.
2. The test run showed that the high stress areas were not appreciably impacted by the face-to-face fully bonded model. This was partially due to the fact that this component is characterized by significant load transmission via compressive contact between interfacing plates, i.e., the large compressive loads on the structure do not result in a lot of significant tensile loads on the plate-to-plate connections which would be distributed by the welds.
3. The test run required many non-linear connections to model the surface interaction between plate faces between the welds greatly increasing the complexity of the model as well as the runtimes of the model. Furthermore, the existence of the line-to-line contacts created significant areas of unrealistic results in the immediate vicinities of the connections, since the only pathway to transmit certain loads was through the one-dimensional line-to-line connection. While the face-to-face fully bonded connection model showed similar singularity effects at the corners and edges of the plates, they were slightly mitigated by load transfer capability between material next to the edges, which is more in agreement with the physical reality of tensile load distribution across the welded areas.

Considering the above points, the plate-to-plate connections were maintained as fully bonded in the FE model with the understanding that the particular method of connection had a negligible impact on the results, and the simplicity afforded to the model by the chosen connection method added to the model's value and useability.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Further Evaluation Section 3.5.2.2.2.7 – Embrittlement Analysis Uncertainties

RAI 3.5.2.2.2.7-2

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 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. 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 managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, 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

Note 1 in Tables 1 and 2 of SLRA Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022, states that the “critical flaw size is calculated using fracture toughness which considered +25% iron dpa [displacement per atom] to account for analytical uncertainties associated with the methodology used to calculate embrittlement.”

Issue

It is not clear from the quoted citation above whether the additional “+25% iron dpa to account for analytical uncertainties associated with the methodology used to calculate embrittlement” includes uncertainties associated with the neutron fluence calculations only or includes uncertainties associated with neutron fluence calculations and other parameters.

Request

Clarify whether the additional “+25% iron dpa to account for analytical uncertainties associated with the methodology used to calculate embrittlement” in the quoted citation above includes uncertainties associated with the neutron fluence calculations only or includes uncertainties associated with neutron fluence calculations and other parameters. If the latter, state the other parameters with which the uncertainties are associated.

PSL Response

The estimated analytical uncertainty of +25% is associated with the methodology used to calculate the iron atom displacement exposures of the RPV support structure only. It does not account for any additional uncertainties that may be associated with other parameters used to calculate RPV support structure embrittlement. Refer to response to RAI 3.5.2.2.2.6-1 (Attachment 19 to this letter) for the detailed discussion of the additional 25% iron dpa to account for fluence uncertainty.

Note 1 in Tables 1 and 2 of SLRA Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022 (ADAMS Accession No. ML22103A014) refers to the detailed fracture mechanics evaluation in WCAP-18623-P, Revision 1 (Reference 1). This report calculated the fracture toughness with the additional 25% iron dpa to account for analytical uncertainties associated with the methodology used to calculate embrittlement. The PSL RPV supports critical flaw sizes were found to be acceptable with adequate margin. As shown in Tables 1 and 2 of SLR Section 3.5.2.2.2.7, as amended by Supplement 2, the PSL detailed fracture evaluation in WCAP-18623-P supports the conclusion in the LTR-SDA-21-021-P assessment, i.e., the PBN RPV supports critical flaw sizes are more limiting than PSL, even with the additional +25% iron dpa to capture uncertainties in the fluence calculation.

References:

1. Westinghouse Report WCAP-18623-P/NP, Revision 1, “St. Lucie Units 1 & 2 Subsequent License Renewal: Fracture Mechanics Assessment of Reactor Pressure Vessel Structural Steel Supports,” December 2021.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.