June 13, 2022

L-2022-075
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 1A RESPONSE AND REQUEST FOR CONFIRMATION OF INFORMATION (RCI) SET 1 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 May 12, 2022, St. Lucie SLRA – Request for Additional Information Set #1 (FINAL) (ADAMS Accession Nos. ML22133A002, ML22133A003)
6. NRC Email and Attachment dated May 26, 2022, St. Lucie SLRA RCI Set 1 Final (ADAMS Accession Nos. ML22147A086, ML22147A087)

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). Attachments 1-17 to this letter provide the partial response to those information requests (i.e., RAI Set 1A). The response to the outstanding (Set 1B) RAIs will be provided by June 30, 2022. Based on the NRC's review of the SLRA, the NRC issued its Set 1 RCI to FPL (Reference 6). Attachment 18 to this letter provides the response to the RCI Set 1.

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 13th day of June 2022.

Sincerely,

William Maher

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18	B.2.3.3-1	Reactor Head Closure Stud Bolting Examination Results
END		

Reactor Head Closure Stud Bolting AMP

RAI B.2.3.3-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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. To complete its review and enable the staff to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

St. Lucie SLRA Section B.2.3.3 describes the applicant's aging management program (AMP) for the reactor head closure stud bolting (studs, nuts, washers, and threads-in-flange) of the St. Lucie units. By letter dated January 27, 2020 (ADAMS Accession No. ML20027B419), the applicant requested relief from the inspection schedule specified in the ASME Code, Section XI for examination of the studs, nuts, and washers (the threads-in-flange are not included in the relief) for the St. Lucie Unit 1 fifth 10-year inservice inspection (ISI) interval and for the St. Lucie Unit 2 fourth 10-year ISI interval The applicant stated that it had requested relief in order to accommodate an additional set of reactor head closure studs, nuts, and washers that are shared in rotation between the St. Lucie units. The staff issued the safety evaluation for this proposed relief by letter dated February 17, 2021 (ADAMS Accession No. ML21027A226). Associated with this proposed relief, the applicant is taking an exception to Element 4, "Detection of Aging Effects" of the AMP in GALL-SLR because the use of three sets of reactor vessel closure studs, nuts, and washers (instead of just two sets) does not make it feasible to maintain an inspection cycle that meets the ASME Code, Section XI requirements. Additionally, based on its audit of the AMP in the St. Lucie SLRA, the staff identified an exception to Element 5, "Monitoring and Trending" of the AMP in GALL-SLR, because this element also refers to the ASME Code, Section XI, inspection requirements for the reactor vessel closure studs, nuts, and washers.

Issue:

The staff noted that, per the safety evaluation dated February 17, 2021, the proposed alternative (referred to as "relief" in SLRA Section B.2.3.3) submitted pursuant to 10 CFR 50.55a(z)(1), was authorized only through the fifth 10-year ISI interval of Unit 1 and only through the fourth 10-year ISI interval of Unit 2. The staff noted that a separate alternative will need to be requested and submitted to the NRC for review and approval in order for the exception to Element 4 and staff-identified exception to Element 5 of the AMP in GALL-SLR to continue for the remainder of the subsequent period of extended operation after the fifth 10-year ISI interval of Unit 1 and after the fourth 10-year ISI interval of Unit 2. However, the staff is not clear whether relief similar to the one described in SLRA Section B.2.3.3 will be requested and submitted to the NRC for review and approval.

Request:

Clarify whether a proposed alternative (referred to as “relief” in SLRA Section B.2.3.3) similar to the one described in SLRA Section B.2.3.3 will be requested and submitted to the NRC for review and approval in order to continue the exception to Element 4 and staff-identified exception to Element 5 of the reactor head closure stud bolting AMP in GALL-SLR described above for the remainder of the subsequent period of extended operation prior to the end date of the fifth 10-year ISI interval of Unit 1 and after the fourth 10-year ISI interval of Unit 2. If an alternative will not be submitted for prior NRC review and approval, justify the exception to Element 4 and staff-identified exception to Element 5 of the reactor head closure stud bolting AMP in GALL-SLR for the remainder of the subsequent period of extended operation after the fifth 10-year ISI interval of Unit 1 and after the fourth 10-year ISI interval of Unit 2.

PSL Response:

A separate alternative (referred to as “relief”) for each unit similar to the one described in SLRA Section B.2.3.3 will be submitted pursuant to 10 CFR 50.55a(z)(1) to the NRC for review and approval in order to continue the exceptions to Elements 4 and 5 of the reactor head closure stud bolting AMP. This submittal will be made in conjunction with the submittal of the revised PSL Unit 1 and 2 Inservice Inspection (ISI) Program Plans for each new ten-year inspection interval through the remainder of the subsequent period of extended operation (SPEO). Accordingly, additional Enhancements to Elements 4 and 5 of the reactor head closure stud bolting AMP are added.

Section 19.4 (Table 19-3) of Appendix A1 and Appendix A2, as well as Section B.2.3.3 of the PSL SLRA Revision 1 are revised accordingly. Note that Section B.2.3.3 of the PSL SLRA Revision 1 was also impacted by SLRA Supplement 1 (Reference ML22097A202), Attachment 13, which added the exception to Element 5. The update provided in SLRA Supplement 1 is also reflected in the SLRA revision below for clarity.

References:

None.

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.4, Table 19-3, Commitment No. 6 on page A1-65 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
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.: PSL1: 09/01/2035

SLRA Appendix A2, Section 19.4, Table 19-3, Commitment No. 6 on pages A2-66 and A2-67 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
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

SLRA Appendix B Section B.2.3.3, Exceptions to NUREG-2191, Exception 2 on pages B-48 and B-49, is revised as follows:

2. For Element 4, Detection of Aging Effects, **and Element 5, Monitoring and Trending,** relief for the Unit 1 Fifth 10-Year ISI Interval and Unit 2 Fourth 10-Year ISI Interval has been requested 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. **This relief request was approved by the NRC on February 21, 2021 (ADAMS Accession Number ML21027A226).** Use of three sets of reactor vessel closure studs, nuts, and washers does not make it feasible to maintain an inspection cycle which meets the requirements of successive examinations per ASME Section XI, IWB-2420(a). Note that PSL Units 1 and 2 were granted similar relief for the Fourth and Third 10-Year ISI Intervals respectively. **In order to continue this exception, separate proposed alternatives for relief of this requirement will be submitted pursuant to 10 CFR 50.55a(z)(1) for approval in subsequent ISI intervals for both Units 1 and 2 through the remainder of the SPEO.**

SLRA Appendix B Section B.2.3.3, Enhancements Table, on page B-49, is revised as follows:

Element Affected	Enhancement
2. Preventive Actions	Revise the procurement requirements for reactor head closure stud material to ensure that the maximum yield strength of replacement material is limited to a measured yield strength less than 150 ksi and a maximum tensile strength of 170 ksi. In addition, revise maintenance procedures to preclude the use of molybdenum disulfide (MoS_2) lubricant for the reactor head closure stud bolting.
4. Detection of Aging Effects **5. Monitoring and Trending**	**Separate 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 will be submitted pursuant to 10 CFR 50.55a(z)(1) for approval in subsequent ISI intervals for both Units 1 and 2 through the remainder of the SPEO.**
7. Corrective Actions	Revise the procurement requirements for reactor head closure stud material to ensure that the maximum yield strength of replacement material is limited to a measured yield strength less than 150 ksi and a maximum tensile strength of 170 ksi.

Associated Enclosures:

None

Reactor Vessel Internals AMP – Fatigue Screening

RAI B.2.3.7-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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. To complete its review and enable the staff to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

In MRP-227, Revision 1-A, the EPRI MRP defines that the CSB flexure welds, LSS core support plates, and UIA fuel alignment plates in Combustion Engineering (CE)-designed pressurized water reactors (PWRs) are Primary category components for the MRP-227 basis being applied to CE-design PWR facilities. For aging management of these components, the EPRI MPR establishes the following aging management inspection and evaluation (I&E) bases for the components per the following inspection items defined in Table 4-2 of the MRP-227, Rev. 1-A report.

- Item C7 for the Primary category CSB flexure weld: Perform EVT-1 visual inspection of the weld no later than two refueling outages from the beginning of the license renewal period (with subsequent re-inspections to be performed at a 10-Year interval) if screening of the flexure weld for both fatigue and stress corrosion cracking (SCC) cannot be satisfied by plant-specific evaluation.

- Item C9 for the Primary category LSS core support plate: Perform EVT-1 visual inspection of the core support plate no later than two refueling outages from the beginning of the license renewal period (with subsequent re-inspections to be performed at a 10-Year interval) if screening of the core support plate for fatigue cannot be satisfied by plant-specific evaluation.

- Item C9 for the Primary category UIA fuel alignment plate: Perform EVT-1 visual inspection of the fuel alignment plate no later than two refueling outages from the beginning of the license renewal period (with subsequent re-inspections to be performed at a 10-Year interval) if screening of the fuel alignment plate for fatigue cannot be satisfied by plant-specific evaluation.

Issue:

An information gap exists on the programmatic I&E bases for these components because neither SLRA AMP B.2.3.7 nor SLRA Appendix C provides any information on whether the screening analysis assessments for these components have been performed, and if so, how they have been performed and whether the analyses qualify as time-limited aging analyses (TLAAs) for SLRA per the TLAA definition criteria in 10 CFR 54.3(a).

Additionally, for the CSB flexure welds and LSS core support plates, the RVI gap analysis in SLRA Appendix C did not alter the Primary Inspection category bases for the CSB flexure welds from those defined for the weld type in Item C7 of Table 4-2 in MRP-227, Rev. 1-A or the Primary category bases for the LSS core support plates from those defined for the plate types in Item C9 of Table 4-2 in MRP-227, Rev. 1-A. However, based on the updated assessment for the UIA fuel alignment plates in MRP 2018-022, the gap analysis basis adjusted the inspection category for the fuel alignment plates by making the plates as Expansion inspection category components for the program (as linked to Primary EVT-1 visual inspections that will be performed on the CSB cylinder middle girth welds [MGWs]). This differs from the 60-year I&E criteria for the UIA fuel alignment plates in Item C10 of Table 4-2 in MRP-2018-022 which maintains the UIA fuel alignment plates as Primary inspection category components if the plates cannot be screened out for fatigue. If the UIA fuel alignment plates were screening out for fatigue, the fuel alignment plates would be placed in “No Additional Measures [NAM] category.

<u>Request:</u>

1. CSB flexure weld bases. Consistent with Item C7 in Table 4-2 of MRP-227, Rev. 1-A, clarify whether the CSB flexure welds are being placed in the Primary inspection category for the AMP based on plans to perform primary EVT-1 inspections of the welds during the period of extended operation or whether the CSB flexure welds are being placed in the NAM category of the program based on performance of fatigue and SCC screening analysis and acceptable screening results of those analyses. If the CSB flexure welds are being placed into the NAM category based on applicable component-specific screening results, identify the type of analyses that were performed for the fatigue and SCC screening objectives of the flexure welds in the current licensing basis (CLB). As part of this, the staff requests that the fatigue and SCC screening analysis or analyses for the flexure welds be provided for the Reactor Vessel Internals AMP. Additionally, clarify whether the applicable component-specific screening analyses for fatigue and SCC need to be identified as TLAAs for the SLRA when assessed against the six criteria for defining TLAAs in 10 CFR 54.3(a).

2. LSS core support plate bases. Consistent with Item C9 in Table 4-2 of MRP-227, Rev. 1-A, clarify whether the LSS core support plates are being placed in the Primary inspection category for the AMP based on plans to perform primary EVT-1 inspections of the plates during the period of extended operation or whether the LSS core support plates are being placed in the NAM category of the program based on performance of a fatigue screening analysis and acceptable screening results of the analysis. If the LSS core support plates welds are being placed into the NAM category based on applicable component-specific screening results, identify the type of analysis that was performed for the fatigue screening objective of the plates in the CLB. As part of this, the staff requests that the applicable type of fatigue screening analysis for core support plates be provided for the Reactor Vessel Internals AMP. Additionally, clarify whether the applicable component-specific screening analysis for fatigue needs to be identified as TLAAs for the SLRA when assessed against the six criteria for defining TLAAs in 10 CFR 54.3(a).

3. UIA fuel alignment plate bases. Since Item C10 in Table 4-2 of MRP-227, Rev. 1-A either placed the UIA fuel alignment plates in either the NAM category or Primary inspection category of the AMP, explain and justify the change in the basis that now places the UIA fuel alignment plates in the Expansion category of the program versus the prior bases for the plates in Item C10 of Table 4-2 in the MRP-227, Rev. 1-A report. As part of this explanation,

clarify if the UIA fuel alignment plates were appropriately screened out for fatigue-type cracking mechanisms for the prior 60-year programmatic basis. If a fatigue screening analysis was performed for the UIA fuel alignment plates as part of the CLB, identify the type of analysis that was performed for the fatigue screening objective of the plates. As part of this, the staff requests that the applicable type of fatigue screening analysis of the fuel alignment plates be provided for the Reactor Vessel Internals AMP. Additionally, clarify whether the applicable component-specific screening analysis for fatigue needs to be identified as TLAAs for the SLRA when assessed against the six criteria for defining TLAAs in 10 CFR 54.3(a).

PSL Response:

The numbered responses below correspond to the numbered requests in the RAI.

1. The CSB flexure welds have been evaluated for fatigue in the CLB and were documented in the Extended Power Uprate (EPU) LAR submittals in FPL Letters L-2010-259 and L-2011-021 (References ML103560419, ML110730116). The analyses were performed in accordance with paragraph NG-3228.3 of ASME Section III and confirmed the cumulative usage factors (CUF) to be less than the ASME Code acceptance limit of 1.0. This demonstrated that the components are not susceptible to cracking due to fatigue through 60 years of operation and was confirmed as such with the resolution to RAI-MF6777/MF6778-EVIB-01 as cited in Section 3.1.2 of the NRC review of the license renewal commitment for the St. Lucie Unit 1 and 2 reactor vessel internals aging management plan (Reference ML18071A002). The fatigue analyses and screening objectives remain consistent with those defined in Attachment 1 to FPL Letter L-2016-040 (Reference ML16063A006).

   The TLAA evaluation in SLRA Section 4.3.1 demonstrates that the total number of design transients used to develop the CUF values for the current licensing basis will not be exceeded during an 80-year subsequent period of extended operation (SPEO). This TLAA will be managed by the Fatigue Monitoring AMP during the SPEO to ensure the CSB flexure welds continue to not be susceptible to cracking due to fatigue. SLRA Appendix B is revised to cite the SLRA Section 4.3.1 TLAA evaluation as the site-specific screening basis for fatigue. In addition, a component-specific fatigue screening evaluation for the CSB flexure has been developed and provides additional confirmation that the CLB CUF acceptance criteria of 1.0 will not be exceeded during the SPEO. A copy of the fatigue screening evaluation for the CSB flexure welds has been posted on the ePortal for reference.

   The current St. Lucie reactor vessel internals AMP does not credit an evaluation of SCC susceptibility and as such, continues to treat the CSB flexure welds as a Primary component subject to the recommended examination method, frequency, and coverage provided in MRP-227 Revision 1-A. Since the submittal of the St. Lucie SLRA, a SCC screening evaluation for the CSB flexure welds has been completed. The evaluation demonstrates that SCC that the CSB flexure welds can be appropriately managed as an expansion component with the CSB upper flange weld as the associated primary component. The evaluation considers the safety and economic consequences, the design similarities, the irradiation dose, the environmental conditions, and the normal operating stresses for each component. While the effects of SCC are not dispositioned as not applicable, SCC in the CSB flexure welds can be appropriately managed as an expansion component to the CSB upper flange weld during the initial period of extended operation and will continue to be managed as such through the SPEO. As the SCC evaluation does not rely on any time dependent inputs, it does not constitute a TLAA.

A copy of the expected changed entries to the St. Lucie RVI AMP showing the Attachment 3 Primary Component table entry for the CSB upper flange weld, Attachment 4 Expansion Component table entry for the CSB flexure weld, and Attachment 5 Examination Acceptance and Expansion Criteria table entry for the CSB upper flange weld are provided below. A copy of the SCC evaluation has been posted on the ePortal for reference. The SLRA is revised to recognize that the RVI AMP manages cracking due to SCC in the CSB flexure welds as an expansion component.

2. The LSS core support plates have been evaluated for fatigue in the CLB and are screened as not susceptible to cracking due to fatigue through 60 years of operation. This evaluation is maintained as a TLAA in SLRA Section 4.3.1 as discussed above. Similar to the CSB flexure welds above, a component-specific fatigue screening evaluation for the CSB flexure welds has also been developed and provides additional confirmation that the CLB CUF acceptance criteria of 1.0 will not be exceeded during the SPEO. A copy of the fatigue screening evaluation for the LSS core support plates has been posted on the ePortal for reference.

   As such, the LSS core support plates are grouped as a No Additional Measures component. SLRA Appendix B is revised to cite the SLRA Section 4.3.1 TLAA evaluation as the site-specific screening basis for fatigue. SLRA Table 3.1.2-2 is revised to recognize the core support plate as a No Additional Measures component including removal of a previously new entry for the core support plate which was added in Attachment 15 of FPL Letter L-2022-043 (Reference ML22097A202). Note that while specific entries for the core support plate are removed from SLRA Table 3.1.2-2, it remains managed by the generic component type entries which address “No Additional Measures” components and “Reactor vessel internal components with a fatigue analysis”.

3. Item C10 in Table 4-2 of MRP-227, Revision 1-A is not applicable to St. Lucie Unit 1 and 2 as the applicability for the UIA fuel alignment plate is limited to only plants with welded core shrouds assembled with full-height shroud plates. This conclusion is consistent with the information regarding the welded core shroud design presented in Section 3.1.1 of the NRC review of the license renewal commitment for the St. Lucie Unit 1 and 2 reactor vessel internals aging management plan (ML18071A002). As such, for the current period of operation, the fuel alignment plates will continue to be managed as a No Additional Measures component. For the SPEO, the UIA fuel alignment plates are appropriately managed as an expansion component per SLRA Appendix C consistent with the guidance in MRP 2018-022.

ATTACHMENT 3

CE PLANTS – PRIMARY COMPONENTS

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C5. Core Support Barrel Assembly Upper flange weld (UFW)	All plants **Applicable for PSL**	Cracking (SCC)	C5.2. Upper Girth Weld (UGW) C5.1. Lower Girth/Flange Weld (LGW/LFW) C5.3. Upper Axial Welds (UAW) C5.4 Lower core support beams C5.5 CSB Flexure Weld (CSBFW)	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of one side of the UFW and ¾" of adjacent base metal shall be examined. (Note 4) See Figure 4-29 of MRP-227.

Notes:

1. Examination acceptance criteria and expansion criteria are in Attachment 5.
4. Examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined.

ATTACHMENT 4

CE PLANTS – EXPANSION COMPONENTS

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
C.5.5. Core Support Barrel Assembly CSB Flexure Weld (CSBFW)	All plants with welded core shrouds	Cracking (SCC)	C5. Upper flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible weld length of one side of the CSBFW and ¾” of adjacent base metal shall be examined.

Note:

1. Examination acceptance criteria and expansion criteria are in Attachment 5.

ATTACHMENT 5

CE PLANTS – EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
C5. Core Support Barrel Assembly Upper Flange Weld (UFW)	All plants **Applicable to PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	C5.2. Upper Girth Weld (UGW) C5.1. Lower Girth/Flange Weld (LGW/LFW) C5.3. Upper Axial Welds (UAW) C5.4 Lower core support beams C5.5 CSB Flexure Weld (CSBFW)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the UFW shall require that the inspection be expanded to include the UGW, LGW, UAW, and CSBFW by the completion of the next refueling outage. b. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the UFW shall require inspection of the lower core support beams within the next three refueling outages.	The specific relevant condition is a detectable crack-like surface indication.

Note:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s)

References:

None

Associated SLRA Revisions:

SLRA Section 3.1.3, Table 3.1.2-2, page 3.1-61, is revised as follows including changes in response to RAI B.2.3.7-2:

Table 3.1.2-2: Reactor Vessel Internals – 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
Core support barrel expandable plugs and patches (Unit 1)	Structural support	Stainless steel	Reactor coolant Neutron flux	Cracking	Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)	~~IV.B3.RP-423~~ **IV.B3.R-423**	3.1-1, 118	A
Core support barrel expandable plugs and patches (Unit 1)	Structural support	Stainless steel	Reactor coolant Neutron flux	Loss of preload	TLAA – Section 4.7.3, Unit 1 Core Support Barrel Repair Plug Preload Relaxation	~~IV.B3.RP-424~~ **IV.B3.R-424**	3.1-1, 119	A
Core support barrel expandable plugs and patches (Unit 1)	**Structural support**	**Stainless steel**	**Reactor coolant** **Neutron flux**	**Loss of fracture toughness**	**Reactor Vessel Internals (B.2.3.7)**	**IV.B3.R-424**	**3.1-1, 119**	**A**
Core support barrel flexure weld	Structural support	Stainless steel	Reactor coolant Neutron flux	Cracking	Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)	~~IV.B3.RP-328~~ **IV.B3.RP-333**	~~3.1-1, 052a~~ **3.1-1, 052b**	~~A~~ **C**
Core support columns	Structural support	Stainless steel	Reactor coolant Neutron flux	Cracking	Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)	IV.B3.RP-363	3.1.1, 052b	A
Core support columns	Structural support	Stainless steel	Reactor coolant Neutron flux	Loss of fracture toughness	Reactor Vessel Internals (B.2.3.7)	IV.B3.RP-343	3.1-1, 056b	A
~~Core support plate~~	~~Structural support~~ ~~Flow distribution~~	~~Stainless steel~~	~~Reactor coolant~~ ~~Neutron flux~~	~~Cracking~~	~~Reactor Vessel Internals (B.2.3.7)~~	~~IV.B3.RP-343~~	~~3.1-1, 056b~~	~~A~~

Table 3.1.2-2: Reactor Vessel Internals – 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
~~Core support plate~~	~~Structural support~~ ~~Flow distribution~~	~~Stainless steel~~	~~Reactor coolant~~ ~~Neutron flux~~	~~Loss of fracture toughness~~	~~Reactor Vessel Internals (B.2.3.7)~~	~~IV.B3.RP-365~~	~~3.1-1, 056a~~	~~A~~
~~Core support plate~~	**~~Structural support~~** **~~Flow distribution~~**	**~~Stainless steel~~**	**~~Reactor coolant~~** **~~Neutron flux~~**	**~~Changes in dimensions~~** **~~Loss of material~~**	**~~Reactor Vessel Internals (B.2.3.7)~~**	**~~IV.B3.R-424~~**	**~~3.1-1, 119~~**	**~~A, 3~~**

SLRA Page 3.1-63 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.

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.

Plant Specific Notes

1. Per Appendix C these components are added to the Primary inspection category in the Reactor Vessel Internals AMP.

2. Per Appendix C the fuel alignment plate is added to the Expansion category in the Reactor Vessel Internals AMP.

~~3. Per Appendix C, the newly screened in aging effects are managed by the Reactor Vessel Internals AMP.~~

SLRA Appendix B, Section B.2.3.7, page B-66, is revised as follows:

The PSL Reactor Vessel Internals AMP applies the guidance in MRP-227 Revision 1-A (as supplemented by a gap analysis) for inspecting and evaluating reactor vessel internal components at PSL. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the SPEO. This AMP includes expanding periodic examinations and other inspections if the extent of the degradation identified exceeds the expected levels.

MRP-227 Revision 1-A provides guidance for selecting reactor vessel internal components for inclusion in the inspection sample. Through this process, the reactor vessel internals were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures. Definitions of each group are provided in MRP-227 Revision 1-A.

A set of Primary reactor vessel internals component locations are inspected because they are highly susceptible to the effects of at least one of the eight aging mechanisms identified above. Another set of Expansion reactor vessel internals component locations are specified to expand the inspection sample should the Primary Component indications be more severe than anticipated.

A third set of reactor vessel internals component locations, Existing Programs components, are susceptible to the effects of at least one of the eight aging mechanisms and are deemed to be adequately managed by Existing Programs, such as American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Examination Category B-N-3, examinations of core support structures.

A fourth set of reactor vessel internals component locations are deemed to require No Additional Measures, for which the effects of all eight aging mechanisms are below the screening criteria as demonstrated in MRP-191 Revision 2.

MRP-227 Revision 1-A includes three components which require site-specific screening to determine the appropriate group; core support barrel (CSB) flexure welds, lower support structure (LSS) core support plates, and upper internals assembly (UIA) fuel alignment plate. The PSL site-specific disposition of these components are outlined below.

The MRP-227 Revision 1-A guidance for the CSB flexure welds is to manage the component as a part of the Primary components group unless they can be shown to be below the screening threshold for fatigue and have an evaluation to disposition the effects of stress corrosion cracking (SCC). The PSL Reactor Vessel Internals AMP demonstrates through fatigue screening that the CSB flexure welds are below the screening threshold. This analysis is shown to be bounding for the SPEO in Section 4.3.1. The PSL Reactor Vessel Internals AMP includes an evaluation of SCC in the CSB flexure welds and concludes it can be appropriately managed as an expansion component to the CSB upper flange weld.

The MRP-227 Revision 1-A guidance for the LSS core support plates is to manage the component as a part of the Primary components group unless they can be shown to be below the screening threshold for fatigue. The PSL Reactor Vessel Internals AMP demonstrates through fatigue screening that the LSS core support plates are below the screening threshold. This analysis is shown to be bounding for the SPEO in Section 4.3.1. As such, the LSS core support plates are dispositioned as No Additional Measures components.

The MRP-227 Revision 1-A guidance for the UIA fuel alignment plates does not apply to PSL Unit 1 and 2 as it is only applicable for sites with welded core shrouds assembled with full-height shroud plates and the PSL design uses partial height shroud plates. As such, the fuel alignment plate is not a part of the Primary Components inspection category, however, the fuel alignment plate is subject to further evaluation for the 80-year operating period.

Based on the results of the gap analysis for the 60- to 80-year operating period, two reactor vessel internals component locations are added to the Primary Components inspection category in addition to those identified in MRP-227 Revision 1-A. In addition, one component is added to the Expansion Components inspection category.

The two reactor vessel internals component locations added to the Primary Components inspection category are the core shroud tie rods and core stabilizing lugs, shims, and bolts. The reactor vessel internals component added to the Expansion Components is the fuel alignment plate. These additions are consistent with the MRP 2018-022 guidance.

Associated Enclosures:

None

Reactor Vessel Internals AMP – Core Support Barrel Plugs and Patches

RAI B.2.3.7-2

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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. To complete its review and enable the staff to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

SLRA Table 3.1.2-2 includes two AMR Items for the Unit 1 CSB expandable plugs and patches: (1) a GALL-SLR-based AMR item (based on GALL-SLR AMR Item IV.B4.R-423, as updated in in NRC Interim Staff Guidance No. SLR-ISG-2021-01-PWRVI) on cracking of the plugs and patches which credits the Reactor Vessel Internals Program as the basis for aging management, and (2) a GALL-SLR-based AMR item (based on GALL-SLR AMR Item IV.B4.R-424, as updated in NRC Interim Staff Guidance No. SLR-ISG-2021-01-PWRVI) on loss of preload in the CSB expandable plugs and patches that credits the time limited aging analysis (TLAA) in SLRA Section 4.7.3 as the basis for aging management.

Issue:

The staff acknowledges the validity of using GALL-SLR AMR Item IV.B4.R-423 as the basis for the AMR line item on cracking of the Unit 1-specific CSB expandable plugs and patches, as given on SLRA page 3.1-61. However, since the scope of the criteria in MRP-227, Rev. 1-A do not bound or include any inspection and evaluation (I&E) criteria for these types of components, aging management for cracking of the Unit 1 CSB expandable plugs and patches should be being done on a St Lucie Unit 1 plant-specific basis. Yet the RVI gap analysis tables in SLRA Appendix C does not include any line item for the Unit 1-specific CSB expandable plugs and patches.

Request:

Provide the basis for why the table entries in SLRA Appendix C do not include any line item entry or entries for the St. Lucie Unit 1-specific CSB expandable plugs and patches that include(s) the following information: (1) inspection category for the components, and the corresponding Primary or Expansion category component if the CSB patches and plugs are identified as Expansion or Primary components under the program, (2) applicable aging effects or mechanisms, (3) component applicability basis, (4) examination method, frequency and coverage criteria, and (5) applicable examination Expansion criteria and examination acceptance criteria.

PSL Response:

The St. Lucie Unit 1 CSB expandable plugs and patches are in the scope of subsequent license renewal consistent with the initial license renewal reactor vessel internals (RVI) aging management program (AMP) (Reference 1). Copies of Attachments 3 and 5 of the current St. Lucie RVI AMP showing the Primary Components table entries for the CSB expandable plugs and patches and the relevant table entries for the Examination and Expansion Criteria are provided below. A copy of the current PSL RVI AMP has also been posted on the ePortal for reference.

As stated in SLRA Section 4.7.3, the St. Lucie Unit 1 CSB expandable plugs and patches were visually inspected in 1984, 1986, 1996, 2008, 2018, and 2019 and no anomalies were identified. Therefore, the current RVI AMP has been effective in managing the aging effect of cracking for the CSB expandable plugs and patches and these visual inspections will continue through the SPEO. The RVI AMP will recognize any CSB expandable plug and patch degradation and be enhanced, as necessary, to ensure continued safe operation through the SPEO.

In addition to the degradation mechanisms identified for the current operating period, the CSB expandable plugs and patches are susceptible to loss of fracture toughness due to neutron irradiation embrittlement consistent with the increased cumulative fluence associated with an 80-year operating period. This degradation mechanism is identified for the SPEO and is appropriately managed by the current inspection method, frequency, and coverage.

SLRA Appendix C is revised below to address the CSB expandable plugs and patches aging management considerations for the SPEO.

ATTACHMENT 3
CE PLANTS – PRIMARY COMPONENTS

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Expandable Plugs and patches (Note 8)	PSL Unit 1 Only	Cracking (IASCC, SCC, Fatigue)	None	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval	Repair region of core support barrel

Note:
1. Examination acceptance criteria and expansion criteria are in Attachment 5.
8. Categorized as primary component due to thermal shield failure earlier in plant life.

ATTACHMENT 5
CE PLANTS – EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Barrel Assembly Expandable plugs and patches (Note 3)	PSL Unit 1 Only	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A

Note:
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
3. These items added as part of RAIs during the license renewal application review and approval.

References:

1. St. Lucie Plant, Unit Nos. 1 and 2 - Review of License Renewal Commitment for Reactor Vessel Internals Aging Management Plan, dated May 2, 2018 (ADAMS Accession No. ML18071A002)

Associated SLRA Revisions:

SLRA Section 3.1.3, Table 3.1.2-2, page 3.1-61, is revised as follows including changes in response to RAI B.2.3.7-1:

Table 3.1.2-2: Reactor Vessel Internals – 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**
Core support barrel expandable plugs and patches (Unit 1)	Structural support	Stainless steel	Reactor coolant Neutron flux	Cracking	Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)	~~IV.B3.RP-423~~ **IV.B3.R-423**	3.1-1, 118	A
Core support barrel expandable plugs and patches (Unit 1)	Structural support	Stainless steel	Reactor coolant Neutron flux	Loss of preload	TLAA – Section 4.7.3, Unit 1 Core Support Barrel Repair Plug Preload Relaxation	~~IV.B3.RP-424~~ **IV.B3.R-424**	3.1-1, 119	A
Core support barrel expandable plugs and patches (Unit 1)	**Structural support**	**Stainless steel**	**Reactor coolant Neutron flux**	**Loss of fracture toughness**	**Reactor Vessel Internals (B.2.3.7)**	**IV.B3.R-424**	**3.1-1, 119**	**A**
Core support barrel flexure weld	Structural support	Stainless steel	Reactor coolant Neutron flux	Cracking	Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)	~~IV.B3.RP-328~~ **IV.B3.RP-333**	~~3.1-1, 052a~~ **3.1-1, 052b**	~~A~~ **C**
Core support columns	Structural support	Stainless steel	Reactor coolant Neutron flux	Cracking	Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)	IV.B3.RP-363	3.1.1, 052b	A
Core support columns	Structural support	Stainless steel	Reactor coolant Neutron flux	Loss of fracture toughness	Reactor Vessel Internals (B.2.3.7)	IV.B3.RP-343	3.1-1, 056b	A
~~Core support plate~~	~~Structural support Flow distribution~~	~~Stainless steel~~	~~Reactor coolant Neutron flux~~	~~Cracking~~	~~Reactor Vessel Internals (B.2.3.7)~~	~~IV.B3.RP-343~~	~~3.1-1, 056b~~	~~A~~

Table 3.1.2-2: Reactor Vessel Internals – 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
~~Core support plate~~	~~Structural support~~ ~~Flow distribution~~	~~Stainless steel~~	~~Reactor coolant~~ ~~Neutron flux~~	~~Loss of fracture toughness~~	~~Reactor Vessel Internals (B.2.3.7)~~	~~IV.B3.RP-365~~	~~3.1-1, 056a~~	~~A~~
~~Core support plate~~	**~~Structural support~~** **~~Flow distribution~~**	**~~Stainless steel~~**	**~~Reactor coolant~~** **~~Neutron flux~~**	**~~Changes in dimensions~~** **~~Loss of material~~**	**~~Reactor Vessel Internals (B.2.3.7)~~**	**~~IV.B3.R-424~~**	**~~3.1-1, 119~~**	**~~A, 3~~**

SLRA Page 3.1-63 is revised as follows:

General Notes

B. Consistent with component, material, environment, aging effect, and AMP listed for NUREG-2191 line item. AMP is consistent with 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.

Plant Specific Notes

1. Per Appendix C these components are added to the Primary inspection category in the Reactor Vessel Internals AMP.

2. Per Appendix C the fuel alignment plate is added to the Expansion category in the Reactor Vessel Internals AMP.

~~3. Per Appendix C, the newly screened in aging effects are managed by the Reactor Vessel Internals AMP.~~

SLRA Appendix B, Section B.2.3.7, page B-66, is revised to add the following new paragraph after the 7th paragraph:

> **The PSL unit 1 core support barrel expandable plugs and patches were evaluated to address potential for degradation mechanisms which may become applicable due to the increased fluence accumulation during 80 years of operation. The aging effect of loss of fracture toughness due to neutron irradiation embrittlement is recognized as applicable for the SPEO. Appendix C concludes that the current inspection scope and schedule remains adequate through the SPEO.**

SLRA Appendix C Section C.2.0, page C-4, is revised as follows:

(d) The unit listings of functional components have been confirmed to include the components and material class as listed in the latest revision of MRP-191.

MRP 2018-022 addresses increases in neutron irradiation dose at 80-years through calculations specifically for representative Combustion Engineering-designed plants. To obtain representative dose projections with a reasonable amount of added conservatism, dose projections were generated using a model for one specific plant at 72 EFPY. To account for variations in axial and radial power shapes, two different dose projections were generated:

- A flat axial power shape that produced conservative results above and below the active fuel, and
- A best-estimate power shape that with 30 percent margin added was more limiting in the radial direction.

A composite dose map was generated using the maximum value of the two dose projections above for each mesh cell in the neutron transport calculation. The above assumptions (a) through (d) were validated for PSL Units 1 and 2 in the NRC ~~SE~~ **review** of the initial license renewal **commitment for the** reactor vessel internals aging management ~~program~~**plan** (Reference ML18071A002) and low leakage fuel management parameters are verified for every cycle. As such, the dose projection used is demonstrated to be applicable to PSL. ~~With respect to item (c), note that the Unit 1 core support barrel expandable plugs and patches, discussed in Section 4.7.3, were developed and the repair method was analyzed by the original vendor.~~ The PSL Units 1 and 2 site specific fluence model is consistent with the guidance of Regulatory Guide 1.190 and the methodology described in WCAP-18124-NP-A (Reference ML18204A010) that was approved by the NRC.

With respect to items (c) and (d), the PSL Unit 1 core support barrel expandable plugs and patches are site-specific components which are not addressed by the industry guidance. These components were developed, and the repair method was analyzed by the original vendor. As cited in the NRC review of the initial license renewal commitment for the RVI AMP (Reference ML18071A002), the components were dispositioned to be managed as a Primary category component with EVT-1 inspections of the repair region on a 10-year interval due to cracking from IASCC, SCC, and fatigue without any ties to expansion components. In addition, the expandable plugs and patches are discussed in Section 4.7.3 and evaluated through a TLAA to address loss of preload through the SPEO. To address potential for degradation mechanisms which may become applicable due to the increased fluence accumulation during 80 years of operation, the expandable plugs and patches will be exposed to the same cumulative fluence as the cylinder girth welds and axial welds. As such, the fluence related screening in MRP-191 Revision 2 for the cylinder girth welds and axial welds is applicable to the core support barrel expandable plugs and patches. This results in one additional applicable aging mechanism which is loss of fracture toughness due to neutron irradiation

embrittlement. While this mechanism will be recognized for the SPEO, there are no other changes necessary as the 60-year inspection method, frequency, and coverage is appropriate to continue through the SPEO.

Continuing the inspections currently performed through the SPEO, in combination with the TLAA evaluation for loss of preload, will appropriately manage the Unit 1 core support barrel expandable plugs and patches including any aging mechanisms introduced through the increased fluence for operation through 80 years.

Associated Enclosures:

None

Reactor Vessel Internals AMP – Core Support Barrel Component Expansion Criteria

RAI B.2.3.7-3

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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. To complete its review and enable the staff to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

On pages C-7 and C-8 of the gap analysis summary in SLRA Appendix C, the applicant identifies that the following three component types are Expansion category components for the Primary category CSB “lower cylinder girth welds” (i.e., CSB MGWs) that will be inspected during the subsequent period of extended operation: (1) CSB middle axial welds (MAWs), (2) CSB lower axial welds (LAWs), and (3) fuel alignment plates in the upper internals assemblies (UIAs).

Issue:

Per the footnotes of SLRA pages C-7 and C-8, the CSB MGWs are the “lower cylinder girth welds” of reference. In MRP 2018-022 (ADAMS Accession No. ML19081A061), the EPRI MRP identifies that the Primary category CSB MGWs should also link to a fourth Expansion category component, the lower support structure (LSS) core support columns. The LSS core support columns are also reflected as Expansion category components for the CSB MGWs in the newly submitted MRP-227, Revision 2 report. However, pages C-7 and C-8 in SLRA Appendix C do not cite or identify that the LSS core support columns are Expansion category components for the CSB “lower cylinder girth welds” (i.e., in addition to the CSB MAWs and LAWs, and the UIA fuel alignment plates as being Expansion category components for the CSB MGWs).

Request:

Provide the basis why the tabular line items and associated footnotes for CSB “lower cylinder girth welds” in SLRA Appendix C (i.e., on SLRA pages C-7 and C-8) only cite the “lower cylinder axial welds” (i.e., the CSB MAWs and LAWs as explained in the tabular footnote) and the UIA “fuel alignment plate” as the “Expansion Link(s)” components for the CSB “lower cylinder girth welds” (i.e., for the CSB MGWs) and do not identify the LSS core support columns as a fourth Expansion category component type for the Primary category CSB ”lower cylinder girth welds.” Additionally, provide the basis why the “Expansion Criteria” column entry of the line item for the CSB “lower cylinder girth welds” on SLRA page C-8 does not include any expansion criteria for the LSS core support columns and why the “Expansion Item Examination Acceptance Criteria” column entry of the same line item on SLRA page C-8 does not specifically define the relevant conditions for fuel alignment plates or LSS core support columns.

PSL Response:

The tables on SLRA pages C-7 and C-8 are transcriptions from MRP 2018-022 (Reference 1), Tables 5-11 and 5-12, including the note which outlines that the intent of the guidance is for a site to implement the MRP-227, Revision 1-A resolution regarding CSB lower cylinder girth welds, when available. The text following the transcription cites that the MRP-227, Revision 1-A resolution establishes the primary component as the middle girth weld with expansion links to the middle axial weld, lower axial weld, and core support columns. The text on SLRA pages C-7 and C-8 immediately following the MRP 2018-022 note identifies that the core support columns are an expansion link to the Primary Item middle girth weld. These sections are revised to clarify the intent of each section of text.

PSL implemented the guidance in MRP-227, Revision 1-A prior to submittal of the SLRA and as such, further references to the PSL actions regarding managing the “lower cylinder girth welds” are to continue implementing MRP-227, Revision 1-A. Transcriptions of Attachments 3, 4 and 5 of the current PSL Reactor Vessel Internals AMP showing the Primary Components table entry for the middle girth weld, the Expansion Components table entries for the middle axial weld, lower axial weld, and core support columns, and the relevant table entry for the Examination and Expansion Criteria are provided below for reference. A copy of the current PSL RVI AMP has been posted on the ePortal for reference.

In addition to continuing to implement MRP-227, Revision 1-A, PSL will add the fuel alignment plate as an expansion component to the middle girth welds beginning in the SPEO. The table on SLRA page C-8 containing the expansion criteria and expansion item examination acceptance criteria is intended to be a transcript of MRP 2018-022 Table 5-12, “*Revised Table Entry”*. This includes individual criteria for the lower cylinder axial welds and the fuel alignment plates designated by “a.” and “b.” Page C-8 is revised to show these character designators. The text following the transcription clarifies the appropriate expansion criteria and expansion acceptance criteria for the fuel alignment plates at PSL. SLRA pages C-6 through C-9 are revised to clarify the intent of the texts following each MRP 2018-022 transcription.

ATTACHMENT 3

CE PLANTS – PRIMARY COMPONENTS

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
C6. Core Support Barrel Assembly Middle Girth Weld (MGW)	All plants **Applicable for PSL**	Cracking (SCC, IASCC) Aging Management (IE)	C6.1. Middle Axial Weld (MAW) C6.2. Lower Cylinder Axial Weld (LAW) C6.3. Core Support Columns	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible weld length of the OD of the MGW and ¾” of adjacent base metal shall be examined. (Note 4) See Figure 4-29 of MRP-227.

Notes:

2. Examination acceptance criteria and expansion criteria are in Attachment 5.
4. Examination coverage requires a minimum of 75% of the length of either the ID or the OD of the weld being examined.

ATTACHMENT 4

CE PLANTS – EXPANSION COMPONENTS

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Support Barrel Assembly C.6.1 Middle Axial Weld (MAW) C.6.2 Lower Axial Weld (LAW)	All plants **Applicable for PSL**	Cracking (SCC, IASCC) Aging Management (IE)	C6. Middle Girth Weld (MGW)	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible weld length of the OD of the MAW and LAW ¾" of adjacent base metal surfaces shall be examined. (Note 5) See Figure 4-29 of MRP-227.

ATTACHMENT 4

CE PLANTS – EXPANSION COMPONENTS

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Support Structure C6.3. Core Support Columns	Plants with full-height bolted or half-weighted welded core shroud plates	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	C6. Middle Girth Weld (MGW)	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	Plants with full-height bolted core shroud plates: 25% of the total number of column assemblies (both visible and non-visible from above the core support plate) using a VT-3 examination from above the core support plate. The inspection coverage must be evenly distributed across the population of column assemblies. Plants with core shrouds assembled in two vertical sections: 25% of the accessible surfaces of the core support column welds, from the top side of the core support plate. The inspection coverage must be evenly distributed across the population of core support column welds. (Notes 3 and 4) See Figure 4-36 of MRP-227.

Notes:

2. Examination acceptance criteria and expansion criteria are in Attachment 5.

3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 of MRP-227 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.

4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies or accessible core support column welds in one quadrant of the core support plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column or weld across the entire plate.

5. Examination coverage requires a minimum of 75% of the weld length for either the ID or the OD of the weld being examined.

ATTACHMENT 5

CE PLANTS – EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
C6. Core Support Barrel Assembly Middle Girth Weld (MGW)	All plants **Applicable to PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	C6.1. Middle Axial Weld (MAW) C6.2. Lower Axial Weld (LAW) C6.3. Core Support Columns	The confirmed detection and sizing of a surface-breaking indication with a length >2 inches in the MGW shall require that the inspection be expanded to include the MAW and LAW by completion of the next refueling outage. The confirmed detection of a surface-breaking linear indication in the MGW shall require examination of 25% (of the total of both visible and non-visible as seen from above the core support plate) of the core support column assemblies by the completion of the next refueling outage. Plants with core shrouds assembled in two vertical sections: The confirmed detection of a relevant disruption of discontinuity in the surface of a core support column weld shall require examination of 100% of the accessible uninspected core support column welds from the top side of the core support plate (minimum of 75 of the total population of core support column welds) during the same outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication. The specific relevant condition for the core support column welds is a disruption or discontinuity in the surface of the weld. The specific relevant condition for the core support columns viewed from above the core support plate is missing or separated welds, or fractured, misaligned, or missing columns.

Note:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s)

References:

1. EPRI MRP 2018-022, Interim Guidance for the Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227-A, for Subsequent License Renewal-Westinghouse and Combustion Engineering-Designed Reactor Vessel Internals, (ADAMS Accession No. ML19081A061)

Associated SLRA Revisions:

SLRA Appendix C, pages C-6 through C-9 are revised as follows:

MRP 2018-022 Expected New Entries for Combustion Engineering Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Upper Internals Assembly Fuel alignment plate (Expansion only after entering SLR period)	All plants with welded core shrouds assembled in two vertical sections	Cracking (IASCC, Fatigue), Loss of material (Wear), Aging management (IE)	Core Support Barrel Assembly: Lower cylinder girth welds	Enhanced visual (EVT-1) examination. Subsequent examination on a ten-year interval*	100% of accessible surfaces.*

* The inspection requirement for the fuel alignment plate is analogous to the current requirement for the Westinghouse-design UCP in MRP-227-A. The inspection technique requirement for the UCP was reduced in MRP-227, Revision 1 and justified in responses to RAIs on MRP-227, Revision 1. Once the NRC safety evaluation is complete, the resulting reduced inspection requirements of the Westinghouse-design UCP can be substituted here for the fuel alignment plate.

Resolution of MRP 2018-022 Table Note for PSL SPEO

As the NRC safety evaluation has been completed, the recommendation is now to perform a visual (VT-3) examination with reinspection every 10 years following initial inspection. The examination coverage is required to be a minimum of 25 percent of core side surfaces if no significant indications are found. However, the examination acceptance criteria require that additional coverage must be achieved in the same outage if significant flaws are found.

Additionally, the primary link for this component is the middle girth weld instead of the lower cylinder girth weld. This is consistent with the changes to the primary and expansion components which occurred during the NRC review of MRP-227 Revision 1.

Fuel Alignment Plate

The addition of the fuel alignment plate in the Expansion Components inspection category is unique to MRP 2018-022 and has not previously appeared in NRC approved guidance. Adding the fuel alignment plate as an expansion component is applicable to PSL for the subsequent period of extended operation. The examination method and examination coverage should be made consistent with the Westinghouse-design UCP.

PSL Actions

This guidance is applicable to the subsequent period of extended operation. PSL will incorporate the guidance, consistent with the examination method, examination coverage, and examination acceptance criteria presented for the Westinghouse-design UCP in MRP-227 Revision 1-A. This augmentation of the guidance presented in MRP 2018-022 is consistent with the note provided.

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MRP 2018-022 Expected Revised Entries for Combustion Engineering Primary Components

Primary Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method / Frequency	Examination Coverage
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by measurable separation between the upper and lower core shroud segments or by shifting of the segments relative to one another or the core support plate Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the horizontal seam between the upper and lower core shroud segments. 100% of the seam between the lower core shroud segment and the core support plate.
Core Support Barrel Assembly Lower cylinder girth welds*	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Lower cylinder axial welds Fuel alignment plate (Plant designs with core shrouds assembled in two vertical sections only)	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the first license renewal period. Subsequent examinations on a ten-year interval	100% of the accessible surfaces of the lower cylinder welds.

* Under MRP-227, Revision 1, this component would be the Core Support Barrel Assembly Middle Girth Weld (MGW) with expansions to the middle axial weld (MAW) and lower axial weld (LAW). Per the responses to NRC RAIs on MRP-227, Revision 1, the core support columns could also become an expansion to the MGW. Once MRP-227, Revision 1 has received a safety evaluation with acceptance of these changes, revisions to the naming of the Primary and Expansion components for the MRP-227-A "lower cylinder girth welds" provided in the approved version of MRP-227, Revision 1 should be substituted here.

Resolution of MRP 2018-022 Table Note for PSL SPEO

As the NRC safety evaluation has been completed, the middle girth weld is the appropriate primary component with expansion links to the middle axial weld, lower axial weld, and core support columns. Consistent with the MRP 2018-022 recommendation to include the fuel alignment plate as an expansion component, it is also linked to the middle girth weld. The examination method, frequency, and coverage defined in MRP-227 Revision 1-A for the middle girth weld is also appropriate.

Core Shroud Assembly

The revisions to the core shroud assembly in the Primary Components inspection category is not unique to MRP 2018-022. These revisions represent predictions of changes which were ultimately made in MRP-227 Revision 1-A. The guidance in MRP-227 Revision 1-A is appropriate to implement.

PSL Actions

PSL will continue to implement the NRC approved changes within MRP-227 Revision 1-A.

Lower Cylinder Girth Welds

The addition of the fuel alignment plate as an Expansion component to the lower cylinder girth welds is unique to MRP 2018-022 and has not previously appeared in NRC approved guidance. Adding the fuel alignment plate as an expansion component is applicable to PSL for the subsequent period of extended operation. Consistent with the note in MRP 2018-022, the primary and expansion components for this link should be consistent with the guidance in MRP-227 Revision 1-A.

PSL Actions

PSL will continue to implement the NRC approved changes within MRP-227 Revision 1-A and add the fuel alignment plate to the expansion link.

MRP 2018-022 Expected Revised Entries for Combustion Engineering Examination Acceptance and Expansion Criteria

Primary Item	Applicability	Examination Acceptance Criteria	Expansion Link(s)	Expansion Criteria	Expansion Item Examination Acceptance Criteria
Core Support Barrel Assembly Lower cylinder girth welds*	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Lower cylinder axial welds b. Fuel alignment plate (Plant designs with core shrouds assembled in two vertical sections only)	a. The confirmed detection and sizing of a surface-breaking indication >2 inches in length in a lower cylinder girth weld shall require an EVT-1 examination of all accessible lower cylinder axial welds by the completion of the next refueling outage. b. (Applicable only after entering the SLR period) The confirmed detection and sizing of a surface-breaking indication >2 inches in length in a lower cylinder girth weld shall require an EVT-1 of the fuel alignment plate within the next three refueling outages.	a. The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication. b. The specific relevant condition is a detectable crack-like surface indication.

* Under MRP-227, Revision 1, this component would be the Core Support Barrel Assembly Middle Girth Weld (MGW) with expansions to the middle axial weld (MAW) and lower axial weld (LAW). Per the responses to NRC RAIs on MRP-227, Revision 1, the core support columns could also become an expansion to the MGW. Once MRP-227, Revision 1 has received a safety evaluation with acceptance of these changes, revisions to the naming of the Primary and Expansion components for the MRP-227-A "lower cylinder girth welds" provided in the approved version of MRP-227, Revision 1 should be substituted here.

Resolution of MRP 2018-022 Table Note for PSL SPEO

As the NRC safety evaluation has been completed, the middle girth weld is the appropriate primary component with expansion links to the middle axial weld, lower axial weld, and core support columns. Consistent with the MRP 2018-022 recommendation to include the fuel alignment plate as an expansion component, it is also linked to the middle girth weld.

The expansion criteria for the fuel alignment plate are consistent with those identified for the Westinghouse-design UCP in MRP-227 Revision 1-A as follows;

The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the middle girth weld shall require inspection of the fuel alignment plate within the next three refueling outages. If an indication is found in this inspection of the fuel alignment plate, the examination coverage shall be expanded to 100 percent of

the accessible surface of the core-side surface of the fuel alignment plate during the same outage.

The expansion acceptance criteria for the fuel alignment plate are consistent with those identified for the Westinghouse-design UCP in MRP-227 Revision 1-A as follows;

The specific relevant conditions for the inspection of the fuel alignment plate are broken or missing parts of the plate.

Lower Cylinder Girth Welds

The addition of the fuel alignment plate as an Expansion component to the lower cylinder girth welds is unique to MRP 2018-022 and has not previously appeared in NRC approved guidance. Adding the fuel alignment plate as an expansion component is applicable to PSL for the subsequent period of extended operation.

Consistent with the note in MRP 2018-022, the primary item, expansion links, expansion criteria, and expansion item examination criteria should be consistent with the guidance in MRP-227 Revision 1-A. In the case of the newly added fuel alignment plate, the expansion criteria and expansion item examination acceptance criteria should be consistent with the Westinghouse-design UCP.

PSL Actions

PSL will continue to implement the NRC approved changes within MRP-227 Revision 1-A and add the fuel alignment plate as an expansion link for the middle girth weld using the expansion criteria and expansion item examination acceptance criteria from the Westinghouse-design UCP.

Fatigue Monitoring AMP

RAI B.2.2.1-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One finding that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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 "monitoring and trending" program element of GALL-SLR Report AMP X.M1, Fatigue Monitoring Program, indicates that the program provides for revisions to the fatigue analyses or other corrective actions (e.g., revising augmented inspection frequencies) on an as-needed basis if the values assumed for fatigue parameters are approached or transient counts exceed the design or assumed quantities.

SLRA Section B.2.3.44 addresses the ASME Code, Section XI, Appendix L flaw tolerance analysis for the pressurizer surge line. The section indicates that the projected 80-year fatigue cycles, as opposed to the design cycles, were used to establish an estimate of the average number of cycles per year for calculating fatigue crack growth. Specifically, the following reference describes the 80-year projected transient cycles that are assumed in the flaw tolerance analysis (Reference: Table 1 of Structural Integrity Report No. 2001262.401, Revision 1, "Flaw Tolerance Evaluation of St. Lucie Units 1 and 2 Surge Line Using ASME Code, Section XI, Appendix L for Subsequent License Renewal").

Issue:

SLRA Section B.2.2.1 addresses the Fatigue Monitoring Program. However, the SLRA section does not clearly describe whether the Fatigue Monitoring Program will monitor the transient cycles, which are assumed in the Appendix L analysis for the pressurizer surge line, to ensure that the actual cycles do not exceed the assumed transient cycles.

Request:

Clarify whether the Fatigue Monitoring Program will monitor the transient cycles, which are assumed in the Appendix L flaw tolerance analysis for the pressurizer surge line, to ensure the validity of the cycles that are used in the flaw tolerance analysis. If some of the transients are not monitored for cycle counting, explain why cycle monitoring is not needed for those transients (e.g., demonstration of conservatism associated with the transient cycles assumed in the flaw tolerance analysis compared to the estimated 80-year cycles representing actual cycles).

PSL Response:

As discussed in the disposition of TLAA 4.3.3, the effects of aging for the PSL Units 1 and 2 pressurizer surge lines due to environmentally-assisted fatigue will continue to be managed by the Pressurizer Surge Line AMP through the SPEO in accordance with 10 CFR 54.21(c)(1)(iii). The Fatigue Monitoring Program AMP, and associated cycle counting, is not credited for managing the effects of aging of the pressurizer surge lines due to the significant amount of margin available in the ASME Code, Section XI, Appendix L flaw tolerance analysis for the pressurizer surge lines as described below.

Per SLRA Section B.2.3.44, the PSL Pressurizer Surge Line AMP is an existing site-specific AMP that incorporates an aging management inspection program that has been previously accepted by the NRC (Reference 1). Following the guidelines of Table L-3420-1 of Appendix L and IWB-2410 of ASME Code, Section XI, the successive volumetric inspection schedule for every surge line weld, including those for the pressurizer surge nozzles and hot leg surge nozzles, is 10 years from the time of the last inspection of those welds during the SPEO (refer to SLRA Table B-10).

The revised Appendix L flaw tolerance evaluation results are shown in SLRA Table B-9 and conclude that the allowable operating period of 47 years at the bounding surge line weld location is significantly greater than the 10 year inspection interval required by Appendix L. Furthermore, the Appendix L flaw tolerance evaluation implies that the allowable operating period for every surge line weld is 47 years from the time of the last inspection of those welds during the SPEO. This means that each pressurizer surge line weld in the Appendix L flaw tolerance evaluation is subjected to 47/80 of the projected fatigue cycles included in Table 1 of Structural Integrity Report No. 2001262.401, Revision 1 (Reference 2), during each 10 year inspection interval. Therefore, cycle counting of these transients for each 10 year inspection interval is not required.

References:

1. Letter from NRC to Mr. Mano Nazar, St. Lucie Plant, Unit Nos. 1 and 2 - Review of License Renewal Commitment for Pressurizer Surge Line Welds Inspection Program, dated October 13, 2016 (ADAMS Accession No. ML16235A138)

2. Structural Integrity Report No. 2001262.401, Revision 1, “Flaw Tolerance Evaluation of St. Lucie, Units 1 and 2 Surge Line Using ASME Code, Section XI, Appendix L for Subsequent License Renewal,” July 15, 2021

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Pressurizer Surge Line AMP

RAI B.2.3.44-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One finding that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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:

SLR Section B.2.3.44 addresses the Pressurizer Surge Line Program that is a plant-specific program for 80 years of operation.

Issue:

SLRA Section B.2.3.44 provides the overall program description and “operating experience” program element of the Pressurizer Surge Line Program. However, SLRA Section B.2.3.44 does not clearly describe the other program elements of the Pressurizer Surge Line Program even though this program is a plant-specific program that is not generically described in the GALL-SLR report.

Request:

Provide the program elements of the Pressurizer Surge Line Program other than the “operating experience” program element, consistent with SLR-SRP Section A.1.2.3, “Aging Management Program Elements.”

PSL Response:

The PSL Pressurizer Surge Line Inspection Program (Fatigue) is an existing Aging Management Program (AMP) that was originally developed to address the effects of environmentally assisted fatigue (EAF) for the PSL pressurizer surge line welds during the initial period of extended operation (PEO). The description of the proposed Pressurizer Surge Line Inspection Program (Fatigue), including the details of the ten program elements, was submitted for NRC approval on October 29, 2015 (ADAMS Accession No. ML15314A160) and approved October 13, 2016 (ADAMS Accession No. ML16235A138).

The AMP is being carried forward for SLR. The additional program elements of the Pressurizer Surge Line AMP other than the “operating experience” program element are provided in the

SLRA markups below, which depict consistency with SLR-SRP Section A.1.2.3, “Aging Management Program Elements.”

References:

None.

Associated SLRA Revisions:

SLRA Appendix B Section B.2.3.44, on pages B-305 through B-310, is revised as follows:

Program Description

The PSL Pressurizer Surge Line AMP is an existing site-specific AMP that formerly was the PSL Pressurizer Surge Line Inspection Program (Fatigue). This AMP assesses fatigue based on the approach documented in the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components, Non-Mandatory Appendix L Operating Plant Fatigue Assessment." This AMP incorporates an aging management inspection program that has been approved by the NRC. A flaw tolerance evaluation was performed specifically for PSL to assess the operability of the surge lines by using ASME Code Section XI Appendix L methodology and to determine the successive inspection schedule for the surge line welds with a postulated surface flaw. Two bounding locations applicable to both Units were evaluated in detail.

The two bounding weld locations of concern are the hot leg surge nozzle elbow-to-pipe weld and the adjacent elbow base material, which is a CASS material. Based on a comparison of geometry, material properties and applicable loads, the results of the detailed evaluation of the two bounding locations are also applicable to all other in-between pipe locations on the surge lines for both units.

The technical analysis supporting the postulated flaw tolerance evaluation for original License Renewal was provided by NRC letter to FPL (Reference ML16235A138), “PSL Plant, Unit Nos. 1 and 2 – Review of License Renewal Commitment for Pressurizer Surge Line Welds Inspection Program (CAC Nos. MF7026 and MF7027)”. The results of the circumferential crack growth for the hot leg surge nozzle elbow and weld are presented in the Table B-8 below.

Table B-8
Hot Leg Surge Nozzle Elbow and Weld Circumferential Crack Growth Results

Location	Stress Path	Crack Growth Results: Final Flaw Depth [a/t]	Final Flaw Depth (in.)	Final Half Flaw Length (in.)	Final Half Flaw Length (θ/π)	Allowable Operating Period (months)
Base Metal	1	0.7481	0.9815	2.9445	0.163	432
	2	0.7241	0.9500	2.8500	0.159	624
	3	0.7327	0.9613	2.8839	0.160	384
	4	0.7394	0.9701	2.9103	0.162	252
Weld Metal	9	0.2808	0.3684	1.1052	0.062	720
	10	0.2608	0.3422	1.0266	0.057	720
	11	0.3285	0.4311	1.2933	0.072	720
	12	0.2807	0.3682	1.1046	0.061	720

Notes for Table B-8

1. The postulated initial flaw depth is 20 percent of the thickness (i.e., 0.201 inches) and the initial flaw length is 6 times its depth (i.e., 1.206 inches) per Appendix L guidelines.

2. A constant aspect ratio (a/l) of 1/6 is used in the crack growth analysis.

3. Flaw length based on Inner Diameter (ID)

4. The axial stresses are bounding hence only circumferential flaws were analyzed.

5. Per Appendix L, if the allowable operating period is equal or greater than 10 years, the successive inspection schedule shall be equal to the examination interval listed in the PSL ASME Section XI schedule of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Section B.2.3.1) AMP for the component.

Considering the allowable operating periods listed in Table B-8 above are greater than 10 years, per the guidelines of ASME Code Section XI Appendix L, Table L-3420-1, the applicable surge line welds for both units listed in Table B-9 below shall be examined by the end of each inspection interval listed in the schedule of inspection programs in IWB-2410 for the PEO. Note that welds with structural weld overlays (SWOL) have been screened out from the scope of analysis and inspections.

For SLR , the original flaw tolerance evaluation was revised and provided in the SIA Engineering Report No. 2001262.401 (Reference 1.6.47) to address eighty years of plant operation (end of the SPEO) using the ASME Code, Section XI, Appendix L methodology in the 2007 Edition with 2008 Addenda, which is the Code edition specified for PSL Units 1 and 2. The elbow adjacent to the hot leg surge nozzle was identified as the sentinel location for the surge line for both Units 1 and 2. As such, the revised Appendix L evaluation for SPEO was performed for the same bounding surge line location as the previous evaluation for PEO.

The revised Appendix L evaluation for SLR uses the same design inputs (i.e. surge line geometry, design transients, piping loads, etc.) and stress analysis as the previous Appendix L evaluation with the following technical changes:

1. For SLR, a separate evaluation of the CASS surge line base metal components was performed in support of the PSL Thermal Aging Embrittlement of Cast Austenitic Stainless Steel AMP (Section B.2.3.6). As such, the scope of the revised Appendix L flaw tolerance evaluation for SLR is only for the weld metal of the surge line in support of PSL ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP (Section B.2.3.1). The PSL ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP (Section B.2.3.1) does not inspect the CASS base metal, and thus only the surge line welds are inspected for Appendix L.

2. For fatigue crack growth in the weld metal, the initial flaw depth was determined from the applicable in-service inspection acceptance standard in ASME Code Section XI Table IWB-3410-1 per the Appendix L methodology [L-3212].

3. Projected 80-year fatigue cycles for PSL Units 1 and 2 were used to establish an estimate of the average number of cycles per year for calculating fatigue crack growth.

4. The latest crack growth curves for Type 304 and Type 316 stainless steels from ASME Code Case N-809 were used for fatigue crack growth. ASME Code Case N-809 has been approved by ASME and has been used in previous Appendix L evaluations for LR and SLR.

The revised Appendix L evaluation for SLR addressed the same stress paths in the weld of the bounding elbow adjacent to the hot leg surge nozzle as the previous Appendix L evaluation for the PEO. The revised evaluation results are shown in Table B-9 below and concluded that the allowable operating period was 47 years at the bounding surge line weld location (i.e., stress path 12 in the weld metal of the elbow adjacent to the hot leg surge nozzle). Furthermore, the bounding evaluation indicates that the allowable operating period for every surge line weld is 47 years from the time of the last inspection of that weld.

Table B-9
Crack Growth Results for Revised Appendix L Evaluation – 80 Years

Analysis Section Number (ASN) (Note 1)	Flaw Configuration (Note 2)	Appendix L Calculated Aspect Ratio	Initial Flaw Size, Acceptable Standards Flaw Size Table Section XI Table IWB-3410-1 (a/t)	Final Flaw Size (a/t)	Maximum Allowable End-of-Evaluation Flaw Size (a/t)	Allowable Operating Period (years)
Stress Path P9	360-Degree Circumferential Flaw	N/A	0.1097	0.2195	0.2214	55
Stress Path P10	360-Degree Circumferential Flaw	N/A	0.1097	0.2194	0.2214	73
Stress Path P11	360-Degree Circumferential Flaw	N/A	0.1097	0.2178	0.2214	51
Stress Path P12	360-Degree Circumferential Flaw	N/A	0.1097	0.2172	0.2214	47

<u>Notes for Table B-9</u>

1. Stress paths are in the weld of the elbow adjacent to the hot leg surge nozzle. The location has been identified as the sentinel location for the surge line of both Units 1 and 2.

2. A 360-degree circumferential flaw bounds a semi-elliptical axial flaw and a semi-elliptical circumferential flaw.

Following the guidelines of Table L-3420-1 of Appendix L and IWB-2410 of ASME Code, Section XI, the successive inspection schedule for every surge line weld, including those for the pressurizer surge nozzle and hot leg surge nozzle at PSL Units 1 and 2, is 10 years from the time of the last inspection of that weld for SLR (refer to Table B-10 below). Therefore, for the SPEO, the effects of EAF for the PSL pressurizer surge line welds will continue to be managed by an inspection program consistent with the AMP approved by the NRC for the initial PEO.

<u>Scope of Program: Element 1</u>

<u>PSL contains twenty one pressurizer surge line weld locations subject to the effects of EAF (ten welds in Unit 1 and eleven welds in Unit 2) that are listed in Table B-10 below. These welds will be examined in accordance with ASME Section XI, IWB for Class 1 piping welds as modified by the requirements of 10 CFR 50.55a. The aging effect managed with these inspections is cracking due to EAF. In each 10-year ISI interval during the SPEO, in scope surge line welds will be inspected in accordance with IWB-2410 and the PSL ASME Section XI schedule of ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP (Section B.2.3.1) under Augmented Programs within</u>

the ISI Program Plans. Note that welds with structural weld overlays (SWOL) have been screened out from the scope of analysis and inspections. The SWOL welds are inspected under ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP (Section B.2.3.1) requirements.

Table B-10
Pressurizer Surge Line Welds Subject to EAF Inspections for SLR

Unit	Weld Number	Inspection Type and Frequency
Unit 1	RC-6-509 (12-inch branch to Safe End)	Weld with SWOL - AMP Not Applicable
	RC-108-FW-3 (Safe-End to Elbow)	Weld with SWOL - AMP Not Applicable
	RC-1-505-A (Elbow to Pipe)	Volumetric Once in 10-Year
	RC-1-505-B (Pipe to Pipe)	Volumetric Once in 10-Year
	RC-1-505-C (Pipe to Elbow)	Volumetric Once in 10-Year
	RC-108-FW-2 (Elbow to Pipe)	Volumetric Once in 10-Year
	RC-2-505-C (Pipe to Pipe)	Volumetric Once in 10-Year
	RC-108-FW-2000 (Pipe to Elbow)	Volumetric Once in 10-Year
	RC-108-FW-2001 (Elbow to Safe-End)	Volumetric Once in 10-Year
	S/C 004 (Surge line Nozzle to Safe-End Weld)	Volumetric Once in 10-Year
Unit 2	RC-301-771 (Nozzle to Safe End)	Weld with SWOL - AMP Not Applicable
	RC-108-FW-3 (Safe End to Elbow)	Weld with SWOL - AMP Not Applicable
	RC-106-751 (Elbow to Pipe)	Volumetric Once in 10-Year
	RC-113-751 (Pipe to Pipe)	Volumetric Once in 10-Year
	RC-107-751 (Pipe to Elbow)	Volumetric Once in 10-Year
	RC-108-FW-2 (Elbow to Pipe)	Volumetric Once in 10-Year
	RC-112-751 (Pipe to Pipe)	Volumetric Once in 10-Year
	RC-101-751 (Pipe to Elbow)	Volumetric Once in 10-Year
	RC-102-751 (Elbow to Pipe)	Volumetric Once in 10-Year
	RC-108-FW-1 (Pipe to Safe End)	Weld with SWOL - AMP Not Applicable
	RC-514-671 (Safe End to Nozzle)	Weld with SWOL - AMP Not Applicable

Based on postulated flaw tolerance analysis in SIA Engineering Report No. 2001262.401 (Reference 1.6.47), and per the guidelines of ASME Code, Section XI, Appendix L, Table L-3420-1, the successive inspection schedule is determined to be ten years. This inspection interval will be used for all surge line piping welds in scope. Examination results are evaluated by qualified individuals in accordance with ASME Section XI acceptance criteria. Components with indications that do not exceed the acceptance criteria are considered acceptable for continued service.

Preventive Actions: Element 2

There are no specific preventive actions under this AMP to prevent the effects of aging.

Parameters Monitored or Inspected: Element 3

Volumetric in-service examinations will be performed for the surge line welds indicated in Table B-10.

Detection of Aging Effects: Element 4

The degradation of surge line welds is determined by volumetric examination in accordance with the requirements of the PSL ISI Program Plans. The frequency and scope of examination are sufficient to ensure that the aging effects are detected before the integrity of the surge line welds would be compromised.

Monitoring and Trending: Element 5

The frequency and scope of the examinations are sufficient to ensure that the cracking aging effect is detected before the intended function of these welds would be compromised. Examinations will be performed in accordance with the inspection intervals based on the results of the postulated flaw evaluation performed in accordance with the ASME Code Section XI, Appendix L methodology provided in SIA Engineering Report No. 2001262.401 (Reference 1.6.47).

Acceptance Criteria: Element 6

Acceptance standards for the in-service inspections are identified in Subsection IWB for Class 1 components. Table IWB-2500-1 identifies references to acceptance standards listed in IWB-3500. Relevant indications found in the surge line welds that are revealed by the in-service inspections, may require additional evaluation per the requirements of ASME Section XI, Appendix L. Indications that exceed the acceptance criteria are documented and evaluated in accordance with the PSL CAP. Operability of the surge line welds will require an IWB-3600 evaluation for acceptance based on engineering evaluation, repair, replacement, or analytical evaluation. Repairs or replacements will be performed in accordance with ASME Section XI, Subsection IWA-4000.

Corrective Actions: Element 7

See Section B.1.3 for discussion on how Corrective Actions: Element 7 is addressed by this AMP.

Confirmation Process: Element 8

See Section B.1.3 for discussion on how Confirmation Process: Element 8 is addressed by this AMP.

Administrative Controls: Element 9

See Section B.1.3 for discussion on how Administrative Controls: Element 9 is addressed by this AMP.

NUREG-2191 Consistency

The PSL Pressurizer Surge Line AMP is consistent with the ten elements of an aging management program described in NUREG-2192, Branch Technical Position A.1.2.3.

Exceptions to NUREG-2191

None.

Enhancements

None.

Operating Experience: Element 10

Industry Operating Experience

PSL evaluates industry OE items for applicability per the FPL OE Program and takes corrective actions, when necessary. For example:

- NRC Information Notice (IN) 88-80: This IN provided Trojan plant experience regarding unexpected piping movement attributed to thermal stratification. No specific action or written response was required as a result of this IN.

- NRC Bulletin No. 88-11: Unexpected movement of the pressurizer surge line during inspections performed at the Trojan plant were observed at each refueling outage since 1982, when monitoring of the line movements began. During the most recent outage prior to this bulletin, the licensee found that in addition to unexpected gap closures in the pipe whip restraints, the piping contacted two restraints. It was also noted that the licensee for Beaver Valley 2 noticed unusual snubber movement and significantly larger-than-expected surge line displacement during power ascension. Unexpected piping movements are highly undesirable because of potential high piping stress that may exceed design limits for fatigue and stresses.

PSL completed all actions associated with this bulletin in conjunction with the Combustion Engineering Owners Group (CEOG). As noted in L-89-91 (Reference ML17222A738) and L-89-276 (ML17223A254), no gross discernable distress or structural damage was identified in the pressurizer surge line for either PSL Units 1 or 2. Therefore, no additional actions were required

Plant Specific Operating Experience

The Pressurizer Surge Line AMP does not have a program health report; however, in scope surge line welds are inspected in accordance with the PSL ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP under Augmented Programs within the ISI Program Plans. Therefore, the health reports for the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP were reviewed for the last five years (2015-2020). The AMP had an Acceptable (Green) status for the entire five years except for the third and fourth quarters of 2018, which were White. However, none of the contributing factors to the White status were related to the pressurizer surge line.

A sample of the surge line welds have been examined ultrasonically in both PSL Units 1 and 2 during the first three in-service inspection intervals in accordance with the requirements of ASME Section XI, Subsection IWB. All in scope welds in the Unit 1 pressurizer surge line were examined in the ISI 4th Interval of Unit 1 except for one weld. To complete the 4th Interval exams, the remaining Unit 1 weld is scheduled for examination during the refueling outage in the fall of 2025. All in scope welds in the Unit 2 pressurizer surge line were examined during the ISI 4th Interval of Unit 2 in 2017.

To date, no reportable indications have been found in the subject pressurizer surge line welds in either unit. The programmatic OE activities described in relevant station procedures ensure the adequate evaluation of OE on an ongoing basis to address age-related degradation and aging management for the PSL Pressurizer Surge Line AMP.

<u>Program Assessments and Evaluations</u>

2016 – Unit 1 Post-Approval Site Inspection for License Renewal, Inspection Report

> The NRC completed a post-approval site inspection for license renewal (LR) for Unit 1. The proposed program for managing EAF of the Unit 1 and 2 pressurizer surge lines was submitted to the NRC on October 29, 2015. The inspectors noted that the proposal detailed the intent of PSL to utilize the ASME, ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP to manage the recurring inspections, and the associated evaluations for any flaws noted. However, at the time of this inspection, a safety evaluation report accepting this program had not yet been issued by the NRC. This was resolved via issuance of the SER, which accepted the program for managing EAF of the pressurizer surge lines.

2017 – Focused Self-Assessment PSL Unit 2 License Renewal Implementation

> This self-assessment focused on the readiness for the NRC LR Implementation IP71003 Phase II inspections scheduled for October 2017. LR implementation was determined to be on track but identified some existing findings that needed to be closed prior to the August 28, 2017 NRC Phase II inspection; however, none of the findings involved the pressurizer surge line. With regard to the Pressurizer Surge Line AMP, the self-assessment stated that the NRC had approved the program via the SER issued 10/13/2016 and that all assignments had been completed.

2020 – License Renewal Effectiveness Review

AMP effectiveness will be assessed at least every five years per NEI 14-12. A 5-year effectiveness review was completed in January 2021 and no findings related to the PSL Pressurizer Surge Line AMP were identified.

The PSL Pressurizer Surge Line AMP is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry OE, including research and development, such that the effectiveness of the AMP is evaluated consistent with the discussion in NUREG-2191, Appendix B.

Associated Enclosures:

None

Metal Fatigue of Non-Class 1 Components TLAA – Sample Line Stress Analysis

RAI 4.3.2-1

Regulatory Basis:

Pursuant to 10 CFR 54.21(c), the SLRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background:

SLRA Section 4.3.2 addresses the implicit fatigue analysis and the associated 80-year cycle projections for non-Class 1 piping systems. Specifically, SLRA Table 4.3.2-2 indicates that the hot-leg sample line is subject to approximately 29,200 cycles for 80 years of operation. Therefore, the relevant stress range reduction factor for the sample line is 0.7, which corresponds to thermal cycles up to 45,000.

Issue:

However, the SLRA does not clearly discuss how the stress analysis for the sample line with the stress reduction factor (0.7) less than 1.0 meets a relevant acceptance criterion.

Request:

Clarify whether the thermal expansion stress (S_E) of the sample line meets the acceptance criteria of the stress analysis for each unit of the St. Lucie plant (e.g., the stress does not exceed the allowable stress range (S_A), as modified by applying the stress reduction factor of 0.7 for the piping). If not, provide justification for why the applicant's stress analysis results with the stress reduction factor less than 1.0 are acceptable, including relevant references (e.g., edition and provisions of a code).

PSL Response:

Revised stress analyses were prepared for the PSL Unit 1 and 2 hot leg sample lines for the 80-year subsequent period of extended operation (SPEO). These stress analysis revisions were required as the 29,200 projected thermal cycles for the hot leg sample lines during the SPEO exceed the 21,900 thermal cycle limit calculated for the current 60-year PEO. As stated in SLRA Section 4.3.2, this required the stress range reduction factor be reduced from a value of 0.8 to 0.7 for the SPEO.

The Unit 1 hot sample lines are designed to the requirements of ANSI B31.7 Class II (line numbers I-3/4-RC-143 and I-3/4-RC-208) and ANSI B31.1 (line number I-3/8-SS-635). The Unit 2 hot sample lines are designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 2 (line numbers I-3/4-RC-143 and I-3/4-RC-208) and ANSI B31.1 (line number I-3/8-SS-635). As discussed in SLRA Section 4.3.2, cyclic qualification for these codes is based on the number of equivalent full temperature cycles and corresponding stress range reduction factor as listed in SLRA Table 4.3.2-1.

For the SPEO, the calculated allowable stresses using a stress range reduction factor of 0.7 are presented in the table below for the Unit 1 and 2 hot leg sample lines. In all cases, the calculated maximum stresses, which includes thermal expansion stresses, are less that the allowable stress limits, thus meeting the acceptance criteria for the applicable codes and justifying up to 45,000 equivalent full temperature cycles for the Unit 1 and 2 hot leg sample lines.

Line Number	Unit 1		Unit 2	
	Maximum Stress (psi)	**Allowable Stress (psi)**	**Maximum Stress (psi)**	**Allowable Stress (psi)**
I-3/4-RC-143	25,523	35,132.5	28,465	35,132.5
I-3/4-RC-208	25,523	35,132.5	28,465	35,132.5
I-3/8-SS-635	12,381	16,100	9,472	16,100

The calculations presenting the above results have been posted on the ePortal.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Metal Fatigue of Non-Class 1 Components TLAA – Disposition Clasification

RAI 4.3.2-2

Regulatory Basis:

Pursuant to 10 CFR 54.21(c), the SLRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background:

SLRA Section 4.3.2 addresses the implicit fatigue analysis and the associated 80-year cycle projections for non-Class 1 piping systems. In addition, SLRA Appendix A1, Section 19.3.3.2 and Appendix A2, Section 19.3.3.2 provide the UFSAR supplement summarizing the implicit fatigue analysis for St. Lucie Units 1 and 2, respectively.

Issue:

The fatigue analysis in SLRA Section 4.3.2 is based on the 80-year cycle projections. However, SLRA Section 4.3.2 refers to the TLAA dispositions in accordance with both 10 CFR 54.21(c)(i), which indicates that the TLAA remains valid, and 10 CFR 54.21(c)(ii), which indicates that the TLAA has been projected. The same TLAA dispositions are listed in SLRA Table 4.1.5-3. In contrast, the UFSAR supplement descriptions in SLRA Appendix A1, Section 19.3.3.2 and Appendix A2, Section 19.3.3.2 refer to only the TLAA disposition in accordance with 10 CFR 54.21(c)(i).

Request:

Explain why SLRA Section 4.3.2 includes the TLAA disposition per 10 CFR 54.21(c)(i) even though the TLAA is based on the 80-year cycle projections. In addition, resolve the inconsistency among the TLAA dispositions described in SLRA Section 4.3.2, SLRA Table 4.1.5-3 and UFSAR summary descriptions (SLRA Appendix A1, Section 19.3.3.2 and Appendix A2, Section 19.3.3.2).

PSL Response:

SLRA Table 4.1.5-3 provides the correct resolution for TLAA 4.3.2, Metal Fatigue of Non-Class 1 Components. Specifically, the PSL Unit 1 and 2 non-Class 1 allowable stress calculations remain valid for the SPEO in accordance with 10 CFR 54.21(c)(1)(i) for all piping systems with the exception of the Unit 1 and 2 RCS hot leg sample lines. The Unit 1 and 2 RCS hot leg sample line allowable stress calculations results have been projected to the end of the SPEO in accordance with 10 CFR 54.21(c)(1)(ii).

SLRA Section 4.3.2 and UFSAR summary descriptions Appendix A1, Section 19.3.3.2 and Appendix A2, Section 19.3.3.2 have been revised to reflect the correct TLAA resolutions.

References:

None

Associated SLRA Revisions:

SLRA Section 4.3.2, page 4.3-17, is revised as follows:

> For the 80-year SPEO, daily hot leg samples would equate to 80 x 365 = 29,200 cycles and exceeds the 22,000 thermal cycles justified for PEO. Therefore, plant specific analyses were developed for the 80-year SPEO for PSL Unit 1 and 2 using a stress range reduction factor (f) of 0.7, which corresponded to the range 22,000-45,000 thermal cycles. Acceptable stress results were obtained with f = 0.7 and justifies a maximum number of cycles at 45,000 for the SPEO. This bounds the 29,200 thermal cycles projected for the 80-year PEO for the PSL Units 1 and 2 RCS hot leg sample lines.
>
> **TLAA Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(ii)**
>
> The PSL Unit 1 and 2 non-Class 1 allowable stress calculations remain valid for the SPEO **in accordance with 10 CFR 54.21(c)(1)(i)** for all piping systems with the exception of the Unit 1 and 2 RCS hot leg sample lines. The Unit 1 and 2 RCS hot leg sample line allowable stress calculations ~~results~~ have been projected to the end of the SPEO in accordance with ~~10 CFR 54.21(c)(1)(i) and~~ 10 CFR 54.21(c)(1)(ii).

SLRA Appendix A1 Section 19.3.3.2, page A1-53 is revised as follows:

> *represents a practical minimum exposure temperature for most plant systems.*
>
> Conservatively, based on this assessment, any system, or portions of systems with operating temperatures less than 220°F were excluded from further consideration. Any ANSI B31.7 and ANSI B31.1 piping systems or portions of systems with operating temperatures above 220°F are conservatively evaluated for fatigue. Once a system is established to operate at a temperature above 220°F, system operating characteristics are established, and a determination is made as to whether the system is expected to exceed 7000 full temperature cycles in 80 years of operation. In order to exceed 7000 cycles a system would be required to heatup and cooldown approximately once every four days. For the systems that are subjected to elevated temperatures above the fatigue threshold, an evaluation was performed to determine a conservative number of projected full temperature cycles for 80 years of plant operation. These projections conclude that 7000 thermal cycles will not be exceeded for 80 years of operation for all mechanical systems with the exception of the PSL Unit 1 RCS hot leg sample piping.
>
> For the 60-year license renewal PEO, the RCS hot leg sample lines were determined to be limiting as these samples are taken on a daily basis. For the 80-year SPEO, daily hot leg samples would equate to 80 x 365 = 29,200 cycles and exceeds the 7000 thermal cycle limit assuming a stress range reduction factor (f) of 1.0.

Therefore, a plant specific analysis was developed for the 80-year SPEO for PSL Unit 1 using a stress range reduction factor (f) of 0.7. Acceptable stress results were obtained with f = 0.7 and justifies a maximum number of cycles at 45,000 for the SPEO. This exceeds the 29,200 thermal cycles assumed for the 80-year SPEO for the PSL Unit 1 RCS hot leg sample line.

Therefore, the **PSL Unit 1** ANSI B31.7 and ANSI B31.1 allowable stress calculations remain valid for the SPEO in accordance with 10 CFR 54.21(c)(1)(i)~~.~~ **for all piping systems with the exception of the Unit 1 RCS hot leg sample line. The Unit 1 RCS hot leg sample line allowable stress calculation has been projected to the end of the SPEO in accordance with 10 CFR 54.21(c)(1)(ii).**

SLRA Appendix A2 Section 19.3.3.2, page A2-53 is revised as follows:

transients. Thus, carbon steel systems or portions of systems with operating temperatures less than 220°F and stainless steel systems or portions of systems with operating temperatures less than 270°F may generally be excluded from such concerns, since room temperature represents a practical minimum exposure temperature for most plant systems.

Conservatively, based on this assessment, any system, or portions of systems with operating temperatures less than 220°F were excluded from further consideration. Any ASME Section III Class 2, ASME Section III Class 3, and ANSI B31.1 piping systems or portions of systems with operating temperatures above 220°F are conservatively evaluated for fatigue. Once a system is established to operate at a temperature above 220°F, system operating characteristics are established, and a determination is made as to whether the system is expected to exceed 7000 full temperature cycles in 80 years of operation. In order to exceed 7000 cycles a system would be required to heatup and cooldown approximately once every four days. For the systems that are subjected to elevated temperatures above the fatigue threshold, an evaluation was performed to determine a conservative number of projected full temperature cycles for 80 years of plant operation. These projections conclude that 7000 thermal cycles will not be exceeded for 80 years of operation for all mechanical systems with the exception of the PSL Unit 2 RCS hot leg sample piping.

For the 60-year license renewal PEO, the RCS hot leg sample lines were determined to be limiting as these samples are taken on a daily basis. For the 80-year SPEO, daily hot leg samples would equate to 80 x 365 = 29,200 cycles and exceeds the 7000 thermal cycle limit assuming a stress range reduction factor (f) of 1.0. Therefore, a plant specific analysis was developed for the 80-year SPEO for PSL Unit 2 using a stress range reduction factor (f) of 0.7. Acceptable stress results were obtained with f = 0.7 and justifies a maximum number of cycles at 45,000 for the SPEO. This exceeds the 29,200 thermal cycles assumed for the 80-year SPEO for the PSL Unit 2 RCS hot leg sample line.

Therefore, the **PSL Unit 2** ASME Section III Class 2, ASME Section III Class 3, and ANSI B31.1 allowable stress calculations remain valid for the SPEO in accordance with 10 CFR 54.21(c)(1)(i)~~.~~ **for all piping systems with the exception of the Unit 2 RCS hot leg sample line. The Unit 2 RCS hot leg sample line allowable stress**

calculation has been projected to the end of the SPEO in accordance with 10 CFR 54.21(c)(1)(ii).

Associated Enclosures:

None.

Environmentally-Assisted Fatigue TLAA - Screening Evaluation

RAI 4.3.3-1

Regulatory Basis:

Pursuant to 10 CFR 54.21(c), the SLRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background:

SLRA Section 4.3.3 addresses the TLAA on the environmentally assisted fatigue (EAF) for reactor coolant pressure boundary components and piping. In relation to the EAF, SLRA Tables 3-1 and 3-2 provide the leading EAF locations (also called sentinel locations) for equipment and piping, respectively (Reference: Westinghouse Report LTR-SDA-II-20-31-NP, Revision 2, "St. Lucie Units 1 & 2 Subsequent License Renewal: Primary Equipment and Piping Environmentally Assisted Fatigue Evaluations").

Issue:

SLRA Section 4.3.3 does not clearly describe the approach for the EAF screening evaluation that was used to determine the leading EAF locations.

Request:

Describe the approach for EAF screening evaluation to determine the leading EAF locations. As part of the response, clarify the following: (1) whether the EAF screening evaluation calculates the environmental fatigue correction factor (F_{en}) and environmental fatigue usage factor (CUF_{en}) values in accordance with NUREG/CR-6909, Revision 1; (2) the criteria and their basis used to determine the leading EAF locations; (3) whether the leading EAF locations are determined based on the CUF_{en} values in each piping system or zone that is exposed to essentially the same thermal and pressure transients; and (4) whether the EAF screening is performed for each material of fabrication (e.g., carbon steel, stainless steel and nickel alloy).

PSL Response:

(1) Steps 3b and 3c of the EAF screening process in Section 3.1 of Enclosure 1 calculate the screening CUF_{en} values in accordance with the NUREG/CR-6909 Revision 1 fatigue curves and F_{en} formulas.

(2) The criteria to determine the leading EAF locations are described in Step 4 of the EAF screening process in Section 3.1 of Enclosure 1 and involve comparing locations within each system on the bases of common transients and stress analysis methods. The technical basis for the EAF screening process, including the transient and stress

analysis method comparisons, is summarized in Section 3.2 and further discussed in Section 4.1 of EPRI Report No. 3002018262 (Reference 1).

(3) Steps 2 and 3a of the EAF screening process in Section 3.1 of Enclosure 1 organize the equipment and piping locations into transient sections, which are defined as groups of locations that experience the same transients (i.e., thermal and related loadings). Step 3e calculates screening CUF_{en} values consistent with the transients experienced by each transient section. Step 4 compares locations within each transient section to identify the leading EAF locations. Note that Step 4 does not compare locations across transient sections to identify the leading EAF locations based on the discussion in Section 3.2.1.

(4) Step 3d of the EAF screening process in Section 3.1 of Enclosure 1 considers the material of fabrication (e.g., carbon steel, low alloy steel, stainless steel, and nickel alloy) in the calculation of the screening CUF_{en} value for each location. Step 4 compares locations of the same material within each transient section to identify the leading EAF locations. Note that Step 4 does not compare locations across materials to identify the leading EAF locations since NUREG/CR-6909 Revision 1 provides unique F_{en} formulas for each material of fabrication.

References:

1. Environmentally Assisted Fatigue Screening Methods (Revision 1). EPRI, Palo Alto, CA: 2020. 3002018262.

Associated SLRA Revisions:

None.

Associated Enclosures:

1. Westinghouse Document No. CSTLM-MC000-TM-CF-000001, Revision 1, St. Lucie Units 1 & 2 Subsequent License Renewal: Westinghouse Response to U.S. NRC RAI 4.3.3-1 Regarding EAF Screening, dated June 2, 2022 (9 pages total)

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DOCUMENT COVER SHEET

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PE SEAL (If required)

ALTERNATE DOCUMENT NUMBER: None

TITLE: St. Lucie Units 1 & 2 Subsequent License Renewal: Westinghouse Response to U.S. NRC RAI 4.3.3-1 Regarding EAF Screening

ATTACHMENTS:
See Section 5.0

Signature Responsibility	Name	SIGNATURE / DATE**(If processing electronic approval select option)**
Originator	Eric J. Matthews	**Electronically Approved*** **
Verifier	Robert Dunn	**Electronically Approved*** **
Reviewer	Mark A. Gray	**Electronically Approved*** **
Responsible Manager	John Klimek	**Electronically Approved*** **

**Approval signifies that the document and all required reviews are complete, the appropriate proprietary class has been assigned, electronic file has been provided to PRIME, and the document is released for use.*

******Electronically approved records are authenticated in the electronic document management system.***

*** This record was final approved on 6/2/2022, 1:01:52 PM. (This statement was added by the PRIME system upon its validation)

Table of Contents

1.0 Background and Purpose

This document provides Westinghouse responses to the Request for Additional Information (RAI) 4.3.3-1 from the United States (U.S.) Nuclear Regulatory Commission (NRC) for the St. Lucie Nuclear Plant (PSL) Units 1 and 2 Subsequent License Renewal Application (SLRA) [1] as related to environmentally assisted fatigue (EAF) screening. Section 2.0 presents RAI 4.3.3-1 from the NRC and Section 3.0 provides Westinghouse's response.

Revision 1 of this document addresses the comments in the attached "RAI 4.3.3-1 EAF - Att 9 Consolidated Comment Form_WEC signed.pdf" file. All changes from Revision 0 to Revision 1 are denoted by revision bars in the left margin.

2.0 Request for Additional Information 4.3.3-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the SLRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

SLRA Section 4.3.3 addresses the TLAA on the environmentally assisted fatigue (EAF) for reactor coolant pressure boundary components and piping. In relation to the EAF, SLRA Tables 3-1 and 3-2 provide the leading EAF locations (also called sentinel locations) for equipment and piping, respectively (Reference: Westinghouse Report LTR-SDA-II-20-31-NP, Revision 2, "St. Lucie Units 1 & 2 Subsequent License Renewal: Primary Equipment and Piping Environmentally Assisted Fatigue Evaluations").

Issue

SLRA Section 4.3.3 does not clearly describe the approach for the EAF screening evaluation that was used to determine the leading EAF locations.

Request

Describe the approach for EAF screening evaluation to determine the leading EAF locations. As part of the response, clarify the following: (1) whether the EAF screening evaluation calculates the environmental fatigue correction factor (F_{en}) and environmental fatigue usage factor (CUF_{en}) values in accordance with NUREG/CR-6909, Revision 1; (2) the criteria and their basis used to determine the leading EAF locations; (3) whether the leading EAF locations are determined based on the CUF_{en} values in each piping system or zone that is exposed to essentially the same thermal and pressure transients; and (4) whether the EAF screening is performed for each material of fabrication (e.g., carbon steel, stainless steel and nickel alloy).

3.0 Westinghouse Response

Westinghouse report LTR-SDA-II-20-31-NP [2] documents the results of EAF evaluations performed by Westinghouse for leading locations determined from an EAF screening process to support the SLRA. A separate Westinghouse report, LTR-SDA-II-20-30-NP [3], documents the results of the EAF screening evaluation performed to determine the leading EAF locations (also called sentinel locations) for the Safety Class 1 reactor coolant pressure boundary (RCPB) components with fatigue analyses in major equipment and piping that meet the six criteria for time-limited aging analyses (TLAAs) in 10 CFR 54.3(a), including the locations listed in NUREG/CR-6260 [4].

The Westinghouse EAF screening approach described in Section 4 of Electric Power Research Institute (EPRI) Report No. 3002018262 [5] was applied for PSL to determine the leading EAF locations presented in LTR-SDA-II-20-30-NP [3]. Sections 3.1 and 3.2 below describe the process elements and technical basis for this EAF screening approach and clarify each item requested in RAI 4.3.3-1 as described in the following:

(1) *Clarify whether the EAF screening evaluation calculates the environmental fatigue correction factor (F_{en}) and environmental fatigue usage factor (CUF_{en}) values in accordance with NUREG/CR-6909, Revision 1.*

Steps 3b and 3c of the EAF screening process in Section 3.1 calculate the screening CUF_{en} values in accordance with the NUREG/CR-6909 Revision 1 [6] fatigue curves and F_{en} formulas.

(2) *Clarify the criteria and their basis used to determine the leading EAF locations.*

The criteria to determine the leading EAF locations are described in Step 4 of the EAF screening process in Section 3.1 and involve comparing locations within each system on the bases of common transients and stress analysis methods. The technical basis for the EAF screening process, including the transient and stress analysis method comparisons, is summarized in Section 3.2 and further discussed in Section 4.1 of EPRI Report No. 3002018262 [5].

(3) *Clarify whether the leading EAF locations are determined based on the CUF_{en} values in each piping system or zone that is exposed to essentially the same thermal and pressure transients.*

Steps 2 and 3a of the EAF screening process in Section 3.1 organize the equipment and piping locations into transient sections, which are defined as groups of locations that experience the same transients (i.e., thermal and related loadings). Step 3e calculates screening CUF_{en} values consistent with the transients experienced by each transient section. Step 4 compares locations within each transient section to identify the leading EAF locations. Note that Step 4 does not compare locations across transient sections to identify the leading EAF locations based on the discussion in Section 3.2.1.

(4) *Clarify whether the EAF screening is performed for each material of fabrication (e.g., carbon steel, stainless steel and nickel alloy).*

Step 3d of the EAF screening process in Section 3.1 considers the material of fabrication (e.g., carbon steel, low alloy steel, stainless steel, and nickel alloy) in the calculation of the screening CUF_{en} value for each location. Step 4 compares locations of the same material within each transient section to identify the leading EAF locations. Note that Step 4 does not compare locations across materials to identify the leading EAF locations since NUREG/CR-6909 Revision 1 [6] provides unique F_{en} formulas for each material of fabrication.

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3.1 EAF Screening Process Steps

The following steps present the process elements of the screening method used to determine the list of sentinel locations. Note that this process excludes Steps 4d to 4f from Section 4.3 of EPRI Report No. 3002018262 [5] since they were not applied in the PSL EAF screening evaluation.

1. **Data Collection:** All of the pertinent inputs, including information on the applicable locations identified in NUREG/CR-6260 [4], must be collected. This includes all of the materials, drawings, and current licensing basis (CLB) fatigue evaluations, if they exist. Any location that was not part of the Class 1 RCPB should be removed from consideration. If the results of this task indicate design differences between comparable components within a unit, the information pertaining to the design differences is evaluated for consideration in the comparisons and then consolidated as part of Step 4. Locations are also excluded during this step based on the following criteria:
    a. Not in contact with primary coolant.
    b. Locations excluded from fatigue usage factor calculation based on fatigue waivers from ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB [7].
    c. Locations with a cumulative usage factor (CUF) of 0.000.
2. **Transient Section Definition:** For this step, the transient sections for all applicable piping systems and equipment included in the screening evaluation must be determined. Components within common transient sections are evaluated as a group. The transient sections are developed based on knowledge of the system function in relation to plant transients, system layouts and flow paths, and/or equipment configurations. This is typically determined from the fatigue analysis of record, since common transient local effects required for the analysis are defined for various groups of components. Refer to Section 3.2.1 for further details on the definition of transient sections.
3. **Screening Environmental Fatigue Multiplier Calculation:** In this step, the fatigue information collected in Step 1 is combined with the transient section definitions established in Step 2 to determine a screening CUF_{en} value for each location susceptible to the effects of the light water reactor environment. The result of this step is an initial list of leading locations that will be further examined in the subsequent steps to determine the plant specific list of sentinel locations.
    a. Organize the locations susceptible to EAF identified in Step 1 into the transient sections defined in Step 2.
    b. Adjust the CUF values by any applicable factors to correct for differences between the fatigue curves used in the source fatigue evaluation (e.g., Section III Appendix I of the ASME Code) and the fatigue curves applicable to the industry document used to determine the screening F_{en}, as required. This factor is represented by F_{adj} and the result of this calculation is CUF_{adj}. The impact of the NUREG/CR-6909 Revision 1 [6] fatigue curves on the component CUF values were considered per the provision in RG 1.207 [8]. Since the AORs were performed to earlier ASME Code editions, an adjustment factor was applied in the calculation of the screening CUF_{en} to account for the differences between the AOR fatigue curves and NUREG/CR-6909 Revision 1 [6] fatigue curves, when the NUREG/CR-6909 Revision 1 [6] fatigue curves produce more limiting CUF_{adj} values than the AOR fatigue curves.

c. Apply the maximum F_{en} of all materials to all components corresponding to the F_{en} formulas from NUREG/CR-6909 Revision 1 [6]. If the screening CUF_{en} is less than 1.0, the location can be removed from the potential sentinel location list. Refer to Section 4.2 of EPRI Report No. 3002018262 [5] for further details on the development of the applicable F_{en} factors.

d. For the remaining potential sentinel locations, determine the maximum F_{en} and F_{adj} for each component based on actual material. These material specific F_{en} and F_{adj} values are used to determine a screening CUF_{en} for each component (designated material F_{en}, F_{adj}, and CUF_{en}). Perform this calculation following the F_{en} formulas and design fatigue curves outlined in the appropriate industry EAF document for the application. If the CUF_{en} is less than 1.0, the location can be removed from the list of potential sentinel locations. Retain at least one location per transient section, for example a CUF_{en} close to 1.0 if none exceeds 1.0, for completeness at this stage. Further treatment of these locations is addressed in Step 4a.

e. As applicable, calculate reduced screening F_{en} factors for each component in each transient section simply based on the maximum temperature experienced in the section, in an effort to reduce the screening CUF_{en} from Step 3d to a value below 1.0 (designated temperature F_{en} and CUF_{en}).

4. **Sentinel Location Identification:** Step 4 establishes the stress basis comparison ranking for the detailed comparison between components and the corresponding down-selection of the leading locations for EAF for each transient section. The result of this step is the plant specific list of sentinel locations.

   a. Remove components with a material or temperature (Steps 3c through 3e) screening CUF_{en} of less than 1.0 from the potential sentinel location list. The screening CUF_{en} values for these locations are conservative based on the approach used to derive the screening F_{en} values. Therefore, a detailed evaluation would be expected to result in a lower screening CUF_{en} value, so further evaluations would not be required for locations with a screening CUF_{en} less than 1.0.

   b. Identify the locations with the maximum screening CUF_{en}, for each applicable material type, in each transient section.

   c. Determine the stress basis comparison ranking for each remaining component.

      i. Determine the level of technical rigor and qualification criteria for each component within the transient section.

      ii. Qualitatively determine the most limiting components in each transient section, using a consistent stress analysis method ranking basis for comparison. This ranking is based on the amount of conservatism considered in the analysis and is described in Section 3.2.2. Note that, while evaluating the stress basis comparison for a given location, it must be confirmed that the CUF value used corresponds to a location that is in contact with primary coolant, in accordance with Step 1a. In some instances, this information may not be known at Step 1a, but would become evident in the details needed to determine the stress analysis basis for comparison. If the location of interest corresponds to a surface not in contact with primary coolant, the corresponding or next most limiting surface in contact with primary coolant must be considered in the stress basis comparisons. This removes from consideration potential high-CUF value locations which are not impacted by environmental effects.

iii. For each material group in each transient section, systematically compare each location to the maximum screening CUF_{en} location considering the stress basis comparison ranking. Remove locations with both a lower screening CUF_{en} and lower analysis rank, until the minimum number of locations is established. The goal is to identify the minimum number of locations, at least one, in each material group in each transient section.

3.2 Technical Basis

EAF is quantified by a CUF_{en} value, which is the product of the component CUF determined using an air fatigue curve and application of an F_{en} multiplication factor to account for the light water reactor (LWR) environment. The value of CUF for a component is a function of the component stress variations and cycles. The stress variations are influenced by the component material, geometry, and transient loadings. The transient loadings are typically influenced by variations in temperature, pressure, and force and/or moment loadings. Experience has shown that fatigue and EAF in Nuclear Class 1 components are most limiting in components subjected to relatively severe thermal transients. Stress ranges are typically dominated by the effects of temperature shocks. Ranges in stress due to forces and moments are generally related to the ranges of the thermal transient temperatures. In the component fatigue analyses, the material, geometry, and fundamental transient loadings (i.e., design transients) are fixed. The variable aspects of the fatigue analysis include modeling and stress calculation methods, simplification of loading applications by bounding or grouping transients or their effects, and conservatism in assumptions that influence various factors in the stress calculation process (such as the elastic-plastic penalty factor, K_e). The F_{en} multiplication factor is influenced by component material, temperature, strain rate, and dissolved oxygen (DO) content in the reactor water. The temperature and strain rate are also influenced by the transients and stress calculation methods discussed above. Therefore, consistent comparison of components that are influenced by these factors needs to address the relative effects of the variable aspects of the evaluation. The bases for assessing these in the comparison process are discussed in the following subsections and further discussed in Section 4.1 of EPRI Report No. 3002018262 [5]. Note that the similarity comparison technical basis discussed in Section 4.1.3 of EPRI Report No. 3002018262 [5] was excluded herein since the PSL Safety Class 1 RCPB components in major equipment and piping have explicit fatigue usage factor calculations.

3.2.1 Transient Section Technical Basis

A transient section is defined as a group of sub-components/locations that experience the same transients (i.e., thermal and related loadings). The concept of transient sections is typically used in design fatigue evaluations of system components for efficiency of application and is also effective for the screening process. Components that reside in the same transient section can be compared with each other to determine the most limiting component (or sentinel location). For locations within a given transient section evaluated with common stress analysis methods, the differences in stresses experienced by each component are generally the result of the material and geometry differences and can be quantified. A typical piping system or major equipment will be divided into several transient sections. Often, it is the section transients themselves that control which components have the highest usage factors in a given system. So, within a particular system, those transient sections with the most severe system transients will usually have components with the highest usage factors.

The transient sections are developed based on knowledge of the system function in relation to plant transients, system layouts and flow paths, and/or equipment configurations. This is typically determined from the fatigue analysis of record, since common transient local effects required for the analysis are defined for various groups of components.

3.2.2 Stress Basis Comparison Technical Basis

A major consideration in the comparison process for a comprehensive screening assessment is the fact that different stress analysis techniques may have been used for each component usage factor calculation. For example, assume there is a component that was analyzed using simplified analytical methods and yielded a usage factor of 0.8. Also, assume there is another component in the same transient section that had a usage factor of 0.8, but was qualified using plastic analysis methods. Although both locations have the same usage factor, the amount of technical rigor that was applied to the second component far exceeds that of the first component. Reanalyzing the first component with the same level of technical rigor as the second component would be expected to produce a lower usage factor. Therefore, the screening comparison must consider the various stress analysis methods and techniques (i.e., technical rigor) that were used in the usage factor evaluation to provide a consistent basis of comparison.

When performing such an assessment, the technical rigor characteristics considered in determining the limiting locations within a given transient section include:

1. Qualification Criteria
2. Stress Analysis Method
    a. Simplified or One-Dimensional Analysis
    b. Finite Element Analysis
        i. Thermal
        ii. Mechanical
    c. Elastic/Plastic Analysis

The EAF screening evaluation considered a consistent ranking system based on these characteristics in the comparisons and selection of leading locations.

4.0 References

1. "St. Lucie Nuclear Plant Units 1 and 2 Subsequent License Renewal Application (Public Version)," August 2021, (ADAMS Accession No. ML21215A318).
2. LTR-SDA-II-20-31-NP Revision 3, "St. Lucie Units 1 & 2 Subsequent License Renewal: Primary Equipment and Piping Environmentally Assisted Fatigue Evaluations."
3. LTR-SDA-II-20-30-NP Revision 1, "St. Lucie Units 1 & 2 Subsequent License Renewal: Primary Equipment and Piping Environmentally Assisted Fatigue Screening Evaluation Results."
4. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," (ADAMS Accession No. ML031480219).
5. *Environmentally Assisted Fatigue Screening Methods (Revision 1).* EPRI, Palo Alto, CA: 2020. 3002018262.
6. NUREG/CR-6909 Revision 1, "Effect of LWR Water Environments on the Fatigue Life of Reactor Materials," May 2018 (ADAMS Accession No. ML16319A004).
7. ASME Boiler and Pressure Vessel (BPV) Code, various editions, and addenda.
8. NRC Regulatory Guide 1.207 Revision 1, "Guidelines for Evaluating the Effects of Light-Water Reactor Water Environments in Fatigue Analyses of Metal Components," June 2018 (ADAMS Accession No. ML16315A130).

5.0 Electronically Attached File Listing

Table 5-1: Electronically Attached File Listing

Filename	Description
St. Lucie RAI Letter Set#1 (FINAL).pdf	Set #1 of RAIs including RAI 4.3.3-1

High-Energy Line Break Analyses TLAA

RAI 4.3.4-1

Regulatory Basis:

Pursuant to 10 CFR 54.21(c), the SLRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background:

SLRA Section 4.3.4 addresses the high-energy line break (HELB) analyses. SLRA Section 4.3.4 also explains that, as discussed in SLRA Section 4.3.1 and Table 4.3.1-2, the original Unit 2 design cycles (CLB cycles) bound the projected cycles for 80 years of operation. In comparison, if a cumulative usage factor (CUF) value is greater than 0.1, such locations are postulated as break locations in the HELB analysis. This CUF threshold for HELB postulation (0.1) is significantly lower than the CUF limit of 1.0 specified in fatigue design analyses.

Issue:

The applicant did not clearly address whether the 80-year operation may increase the CUF values at the Unit 2 Class 1 piping locations above the CUF threshold of 0.1 for HELB postulation such that additional break locations needs to be evaluated in the HELB analysis. For example, SLRA Section 4.3.1 indicates that the design cycles of the “loss of letdown flow” transient is increased from 50 cycles to 500 cycles for the subsequent period of extended operation (SPEO). Based on this cycle increase for the SPEO and the existing CUF threshold for HELB location postulation (0.1), the applicant may need to identify additional HELB locations. The staff also noted a possibility that the identification of additional HELB locations may be needed due to the increases in actual transient cycles during the SPEO.

Therefore, the staff found a need to confirm that, if new additional piping break locations are identified based on the CUF threshold of 0.1, the applicant will evaluate such new break locations in the HELB analysis. The staff also found a need to clarify the activities of the Fatigue Monitoring Program (SLRA Section B.2.2.1) related to the HELB TLAA.

Request:

1. Clarify whether additional break locations and their effects will be evaluated in the Class 1 piping HELB analysis if new additional piping break locations are identified based on the CUF threshold of 0.1 during the SPEO. If not, provide justification for why such additional HELB locations do not need to be evaluated in the HELB analysis.

2. The applicant proposed to use the Fatigue Monitoring Program for managing the aging effect associated with the HELB TLAA, as addressed in SLRA Section 4.3.4 and Enhancement 5 of the program (SLRA Section B.2.2.1). In relation to the program enhancement, clarify (1) whether the program will use the CUF threshold of 0.1 as an

acceptance criterion for HELB location postulations, consistent with SLRA Section 4.3.4 and (2) whether the program will take a relevant action to update the HELB analysis as needed based on potentially new additional HELB locations for 80 years of operation.

PSL Response:

The numbered responses below correspond to the numbered requests in the RAI.

1. Stress analyses of Class 1 piping are performed by assuming a number of design transient cycles that are expected to occur for a 40-year plant life, with margin. The original design transient cycles were estimated based on very conservative assumptions and were intended to bound a broad range of operating conditions and result in a conservative robust design. As a result, calculated CUF values would be expected to bound plant operation and would not be expected to change for the life of the plant.

   The original 40-year design transients, and the CUF values calculated based on those transients, remain bounding for 80 years of plant operation, except for the Loss of Letdown transient, which is applicable to the charging and letdown piping. When reviewing 10 years of plant operating data to project the Loss of Letdown transient cycles for 80 years, it was determined that more cycles occur that would be characterized by this transient than are specifically accounted for in the 40-year design cycles. The additional cycles are the result of starting and stopping a charging pump when letdown flow is isolated for an extended period, to periodically restore pressurizer water level that is gradually lost due to controlled bleed-off flow from the RCP seals. Plant operating data was used to project a total of 450 of these cycles.

   This charging pump cycle transient is enveloped by the Loss of Letdown design transient since the Loss of Letdown design transient includes the shutoff and re-initiation of charging flow while letdown flow is isolated, and the charging line is at ambient temperature. The projected 450 charging pump cycles were conservatively added to the original 50 design transients for Loss of Letdown in SLRA Tables 4.3.1-5 and 4.3.1-6. A reconciliation evaluation was performed to demonstrate that the CUFs for the Class 1 portions of the charging and letdown piping remain below the ASME Code limit of 1.0 when considering 500 Loss of Letdown cycles. In addition, the CUF still remains below 1.0 when considering more than 1000 total Loss of Letdown transient cycles.

   The stress analyses of record for the charging and letdown Class 1 piping impacted by the additional 450 Loss of Letdown transient cycles were reviewed to identify if any additional pipe break locations need to be considered for HELB based on the CUF for a location increasing from less than 0.1 to greater than 0.1. The review did not identify any Class 1 piping locations where the CUF increased from less than 0.1 to greater than 0.1 due to the additional 450 charging pump start and stop transients with letdown flow isolated. Therefore, no new additional piping break locations need to be considered for HELB based on the CUF threshold of 0.1 during the SPEO..

2. Attachment 11 to Subsequent License Renewal Application (SLRA) Supplement 1, (Reference ML22097A202), revised SLRA Appendix A2, Section 19.4, commitment No. 1 portion of Table 19-3, to delete Enhancement e) related to the acceptance criterion associated HELB CUF criteria. A similar deletion was made to Element 6, Acceptance

Criteria, in SLRA Section B.2.2.1. The basis for the deletion of the enhancement was to make the acceptance criteria element for the PSL Fatigue Monitoring program consistent with the acceptance criteria specified in Element 6 of NUREG-2191 Section X.M1. Element 6 of NUREG-2191 Section X.M1 states "The acceptance criterion is maintaining the value of all relevant fatigue parameters to values less than or equal to the limits established in the fatigue analyses, with consideration of reactor water environmental effects, where appropriate, as described in the program description and scope of program."

To remain consistent with Element 6 of the NUREG-2191 Section X.M1 program, the St. Lucie Fatigue Monitoring AMP program description is revised, and a new enhancement is added for Element 1, Scope of Program, to identify the St. Lucie Unit 2 Class 1 piping HELB fatigue CUF criteria of 0.1 and identify actions required if the criteria is exceeded.

SLRA Section B.2.2.1 is revised to include this information.

References:

None.

Associated SLRA Revisions:

SLRA Section B.2.2.1, page B-25, is revised as follows:

CUF_{en} is CUF adjusted to account for the effects of the reactor water environment on component fatigue life. For PSL to ensure that all potential limiting component locations are captured, all the reactor coolant pressure boundary components with existing ASME Code fatigue analyses, including those PSL site-specific NUREG/CR-6260 (Reference ML031480219) locations, have been evaluated for EAF. SLRA Section 4.3.3 provides details of the evaluation for environmentally assisted fatigue for the PSL SPEO. The effects of fatigue on the intended functions of the ASME Code, Section III ~~components~~ piping components ~~listed in Table 4.3.3-1~~ that have a calculated CUF_{en} value less than 1.0 will be managed by this AMP through the use of cycle counting and taking required actions prior to exceeding design limits that would invalidate their conclusions.

The PSL Unit 2 HELB analysis methodology used to define break locations is described in UFSAR Section 3.6.2 and indicates that the CUF criterion (CUF > 0.1) was used to determine Class 1 piping break locations. SLRA Section 4.3.4 provides details of the evaluation for HELB for the PSL Unit 2 SPEO. The AMP governing procedure will be enhanced to take action to revise the affected St. Lucie Unit 2 Class 1 piping fatigue analyses before 80-year plant design cycle limits are exceeded. The revised fatigue analysis will then be reviewed to determine if any new locations have a calculated CUF value greater than 0.1 (CUF > 0.1). If any new locations meeting the HELB criterion of CUF > 0.1 are identified, a pipe break at the new location(s) will be evaluated for impact on essential SSCs, including evaluation for associated dynamic affects (jet impingement, reactive forces and pipe whip, compartment pressure and environmental conditions), as required.

The PSL Fatigue Monitoring AMP provides for corrective actions when any actual transient cycle count comes within 80 percent of the design or projected cycle limit. Plant management is notified in accordance with the program procedural requirements, and the condition is entered into the CAP. Component reevaluation, enhanced inspection, repair or replacement is required to demonstrate that the fatigue design limit will not be exceeded during the SPEO.

NUREG-2191 Consistency

The PSL Fatigue Monitoring AMP, with enhancements, will be consistent without exception to the 10 elements of NUREG-2191, Section X.M1, “Fatigue Monitoring.”

Exceptions to NUREG-2191

None.

Enhancements

The PSL Fatigue Monitoring 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	**Update the AMP governing procedure to take action to revise the affected St. Lucie Unit 2 Class 1 piping fatigue analyses before 80-year plant design cycle limits are exceeded, and identify any new break locations (CUF > 0.1) requiring evaluation for impact on essential SSCs, including evaluation for associated dynamic affects (jet impingement, reactive forces and pipe whip, compartment pressure and environmental conditions), as required.**

Associated Enclosures:

None.

Unit 2 Structural Weld Overlay PWSCC Crack Growth Analyses TLAA

RAI 4.7.8-1

Regulatory Basis:

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

Background:

Under Section 2.4 PWSCC Crack Growth Mechanisms of the Framatome proprietary Document No. 86-9329645-000 (Reference 4.8.46), the PWSCC crack growth rate for Alloy 52M that the applicant used in the TLAA calculations is provided. The NRC has not endorsed this crack growth rate and is currently reviewing its adequacy.

Issue:

The crack growth rate is a key factor in the evaluation provided in this TLAA and could have an impact on the overall conclusion to address the TLAA for Port St. Lucie (PSL) Units 1 and 2. PSL Units 1 and 2 structural weld overlays (SWOLs) are examined in accordance with ASME Code Case N-770-5 (Reference 4.8.47). Code Case N-770-5 requires SWOL welds to be examined at a frequency defined by Table 1 and Note 10. Note 10 states, in part, "Those welds not included in the 25% sample shall be examined prior to the end of the mitigation evaluation period if the plant is to be operated beyond that time." The calculations of this TLAA would establish the scope of welds to be examined within the extended period of operation as well as the examination timeline.

Request:

1. If the NRC, at a future date, endorses a different crack growth rate than that used by the applicant, what actions would the applicant take to address the revised rate?

2. Where is this process documented in the applicant's procedures?

PSL Response:

1. PSL Units 1 and 2 SWOLs are examined as described in SLRA Sec. 4.7.8 in accordance with ASME Code Case N-770-5 (Reference 1). Upon notification through established industry or regulatory communication channels that the NRC has endorsed and/or provided

conditions concerning the PWSCC crack growth rate (CGR) of Alloy 52/152 materials, St. Lucie would enter the notification into its corrective action program. The endorsed criteria, including any conditions or limitations, would be compared to, and evaluated with the CGR presented in the SLRA (and approved by the SER). This endorsed CGR would be evaluated for applicability and extent of condition against current PSL plant operations and against proposed SPEO plant operations as presented in its SLRA. Changes to the TLAA and any resultant inspections, including any components to be added or analyzed, would be performed in a timely fashion and implemented into the applicable aging management program(s). For example, if such notification came within the next 2 years, the changes needed could be implemented in time for applicability to the SPEO for each unit.

2. The PSL corrective action program process is described in a NextEra Energy Nuclear Fleet Administrative Procedure. A copy of this procedure has been posted on the ePortal for review.

References:

1. ASME Code Case N-770-5, “ASME/BPVC CASE N-770-2, “Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1,” dated November 7, 2016.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Buried and Underground Piping and Tanks AMP

RAI B.2.3.27-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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 the staff to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

GALL-SLR Report Table XI.M41-1, “Preventive Actions for Buried and Underground Piping and Tanks,” recommends that the following are externally coated in accordance with the “preventive actions” program element of GALL-SLR Report AMP XI.M41: (a) buried metallic piping; and (b) underground steel piping.

During its audit, the staff reviewed the St. Lucie Nuclear Plant Asset Management Plan for the Underground Piping and Tanks Integrity Program and noted that a coating material of “none” is identified for several material/system combinations. For example, the staff noted that the following have “none” as the coating material: (a) safety-related stainless steel piping in the diesel fuel oil and auxiliary feedwater sub-systems; and (b) safety-related carbon steel piping in the diesel fuel oil sub-system.

During its review of the Updated Final Safety Analysis Report (UFSAR) for Unit 2 (ADAMS Accession No. ML20268A114), the staff noted the following: (a) the two inch piping run between the diesel oil storage tank and the day tanks is encased within a three inch guard pipe; and (b) the guard pipe is coated with a corrosion resistant coating.

By letter dated April 7, 2022 (ADAMS Accession No. ML22097A202), the applicant clarified the following: (a) there is approximately 890 feet of fuel oil piping housed within guard piping; and (b) the guard pipe prevents contact of the fuel oil piping with the soil environment.

Issue:

Based on its observations noted above, the staff seeks clarification on whether the following are coated in accordance with the "preventive actions" program element of GALL-SLR Report AMP XI.M41: (a) buried metallic piping; and (b) underground steel piping. With respect to the fuel oil piping housed within guard piping, the staff notes that the UFSAR specifies portions of guard piping are provided with a "corrosion resistant coating." However, the staff seeks clarification with respect to the following: (a) if this coating is provided for the 890 ft of guard piping referenced in the April 7, 2022, supplement; and (b) if the "corrosion resistant coating" is consistent with the coating types identified in the "preventive actions" program element of GALL-SLR Report AMP XI.M41 (e.g., coal tar enamel).

Request:

Provide clarification regarding if the following are coated in accordance with the "preventive actions" program element of GALL-SLR Report Table XI.M41-1: (a) buried metallic piping (including exterior surfaces of guard piping exposed to soil); and (b) underground steel piping. If all or portions of in-scope piping and piping components are not externally coated in accordance with the "preventive actions" program element of GALL-SLR Report AMP XI.M41, provide justification for why external coatings are not provided.

PSL Response:

The coatings of the buried, underground, and concrete-encased piping are summarized below. Note that the diesel oil guard piping is not within the scope of LR or SLR.

Buried Piping

The materials and coatings associated with buried piping and piping components within the scope of SLR (and diesel oil guard piping) are as follows:

- Cast iron is the primary material used within the buried fire protection system is coated with coal tar epoxy.
- Buried fire protection system carbon steel piping installed circa 2004 is externally coated using Ameron 351.
- Buried fire protection system ductile iron piping installed circa 2004 is externally coated in accordance with NFPA 24.
- Buried carbon steel piping associated with the intake cooling water (ICW) and emergency cooling canal systems is coated with coal tar epoxy, except for the Unit 1 ICW discharge which is coated with fusion bonded epoxy.
- The buried carbon steel guard piping for the diesel fuel oil system is externally coated with coal tar epoxy. This accounts for approximately 590 ft of the 890 ft of fuel oil guard piping.
- The buried stainless steel piping associated with the ICW system (i.e., buried vent and drain piping for Unit 1) is epoxy coated.

- A portion of the Unit 1 auxiliary feedwater (AFW) and condensate system (AFW pump suction) is buried stainless steel piping in sand beneath the turbine building. No coating was identified, however, due to the location beneath the turbine building, this buried piping is not susceptible to wetting.

Underground Piping

The materials and coatings associated with underground piping and piping components within the scope of SLR are as follows:

- The underground (pipe trenches) carbon steel AFW and condensate system piping is protectively coated with Carbo Zinc 11 primer with one finishing coat of High Build Chlorinated Rubber, Series 323. To supplement this coating, the respective piping is inspected on an interval of every 4th refueling outage.
- The underground carbon steel diesel generator fuel oil piping is within guard piping and coated with coal tar epoxy. This accounts for approximately 590 ft of the 890 ft of fuel oil piping housed within guard piping.
- The underground stainless steel diesel generator fuel oil piping associated with the Units 1 and 2 cross-tie is installed within stainless steel guard piping. No coating has been identified, however, the recommendations from NUREG-2191 Table XI.M41-1 state that no coating is recommended for underground stainless steel. This accounts for approximately 300 ft of the 890 ft of fuel oil piping housed within guard piping.

Concrete Encased Piping

The concrete encased piping within the scope of SLR (and diesel oil guard pipe) is as follows:

- A portion of the Unit 2 AFW and condensate system (AFW pump suction) is buried stainless steel piping encased in concrete.
- The buried stainless steel guard piping for the diesel fuel oil system is encased in concrete. This accounts for approximately 300 ft of the 890 ft of fuel oil guard piping.
- The buried stainless steel primary makeup water (PMW) system piping for Unit 2, between the emergency core cooling system (ECCS) pipe tunnel/trench and the refueling water tank, is encased in concrete.

In summary, the buried and underground carbon steel and cast iron piping and piping components are coated with coal tar epoxy or fusion bonded epoxy, meeting the guidance of NACE SP0169-2007 Table 1 and NUREG-2191 Table XI.M41-1. Any exceptions are noted above with justification provided. Buried and underground stainless steel piping and piping components are either not coated or are encased in concrete.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Outdoor and Large Atmospheric Metallic Storage Tanks AMP

RAI 19.2.2.17-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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. 10 CFR 54.21(d) requires each license renewal application to include a final safety analysis report (FSAR) supplement, containing a summary description of the programs and activities for managing the effects of aging. To complete its review and enable the staff to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

In its discussions about FSAR supplements, the Standard Review Plan for Subsequent License Renewal (NUREG-2192) notes that the description should be sufficiently comprehensive such that later changes to the program can be controlled by 10 CFR 50.59. NUREG-2192 also notes that the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report (NUREG-2191), Table XI-01 provides examples of the type of information to be included. GALL-SLR Report Table XI-01, “FSAR Supplement Summaries for GALL-SLR Report Chapter XI Aging Management Programs [AMP],” provides a description of the “Outdoor and Large Atmospheric Metallic Storage Tanks” program (AMP XI.M29), stating that loss of material is managed by conducting periodic internal and external visual examinations.

SLRA Section B.2.3.17, Outdoor and Large Atmospheric Metallic Storage Tanks program operating experience (OE) states that in “April 2011, the U1 RWT caulking inspection had not been performed on time. An extent of condition was performed and revealed that a weekly report on project preventive maintenance activities had not been performed.”

During the on-site audit, Action Report 02412714(AR) was submitted to the corrective action program which states, “While performing a walkdown with NRC, noted that tank base flange to concrete caulking requires repair/replacement.”

Issue:

The Updated Final Safety Analysis Report (UFSAR) supplement for the Outdoor and Large Atmospheric Metallic Storage Tanks program appears to lack a sufficient description of the activities (inspection frequency) that will be used for inspection of caulking or sealant based on the above OE.

Request:

Regarding SLRA Section 19.2.2.17 detailing UFSAR changes: a) provide additional information that explains how the current description of the program and aging management activities in the UFSAR supplement meets the intent of 10 CFR 54.21(d), and NUREG-2192 which states that the description should be sufficiently comprehensive such that later changes to the program can be controlled by 10 CFR 50.59 or b) modify the UFSAR supplement to include the inspection frequency that will be used for inspection of caulking or sealant on the in-scope tanks in the Outdoor and Large Atmospheric Metallic Storage Tanks program.

PSL Response:

Appendices A1, A2, and B of PSL SLRA Revision 1 included a commitment/enhancement to inspect the caulking or sealant of the tanks within the scope of the Outdoor and Large Atmospheric Metallic Storage Tanks AMP on an 18-month interval. The respective UFSAR summary sections in Appendices A1 and A2 are revised to clarify the 18-month frequency for inspecting tank caulking or sealant.

References:

None.

Associated SLRA Revisions:

SLRA Appendix A1 Section 19.2.2.17, pages A1-23 and A1-24, is revised as follows:

19.2.2.17 Outdoor and Large Atmospheric Metallic Storage Tanks

The PSL Outdoor and Large Atmospheric Metallic Storage Tanks AMP is an existing AMP, previously part of the PSPM Program and Structures Monitoring Program. This condition monitoring AMP manages aging effects associated with outdoor tanks sited on concrete and indoor large-volume tanks containing water designed with internal pressures approximating atmospheric pressure that are sited on concrete. The Unit 1 and Common Unit tanks included within the scope of this AMP are as follows:

- Unit 1 Refueling Water Tank (U1 RWT)
- Treated Water Storage Tank (TWST)
- Unit 1 Condensate Storage Tank (U1 CST)
- Diesel Oil Storage Tank 1A (DOST 1A)
- Diesel Oil Storage Tank 1B (DOST 1B)

This AMP includes preventive measures to mitigate corrosion by protecting the external surfaces of steel components per standard industry practice. Sealant or caulking is used for outdoor tanks at the tank bottom interface. This AMP manages loss of material and cracking by conducting one-time and periodic internal and external visual and surface examinations. **The periodic inspections of the respective tank caulking or sealant are performed on an 18-month frequency.** Inspections of caulking or sealant are supplemented with physical manipulation.

Surface exams are conducted to detect cracking for the aluminum U1 RWT. Thickness measurements of tank bottoms are conducted to detect degradation (e.g., loss of material on the inaccessible external surface). Inspections are conducted in accordance with ASME Code Section XI requirements as applicable or are conducted in accordance with plant-specific procedures that include inspection parameters such as lighting, distance, offset, and surface conditions.

SLRA Appendix A2 Section 19.2.2.17, pages A2-23 and A2-24, is revised as follows:

19.2.2.17 Outdoor and Large Atmospheric Metallic Storage Tanks

The PSL Outdoor and Large Atmospheric Metallic Storage Tanks AMP is an existing AMP, previously part of the PSPM Program and Structures Monitoring Program. This condition monitoring AMP manages aging effects associated with outdoor tanks sited on concrete and indoor large-volume tanks containing water designed with internal pressures approximating atmospheric pressure that are sited on concrete. The Unit 2 and Common Unit tanks included within the scope of this AMP are as follows:

- Unit 2 Refueling Water Tank (U2 RWT)
- Unit 2 Primary Water Storage Tank (U2 PWST)
- Unit 2 Condensate Storage Tank (U2 CST)

This AMP includes preventive measures to mitigate corrosion by protecting the external surfaces of steel components per standard industry practice. Sealant or caulking is used for outdoor tanks at the tank bottom interface. This AMP manages loss of material and cracking by conducting one-time and periodic internal and external visual and surface examinations. **The periodic inspections of the respective tank caulking or sealant are performed on an 18-month frequency.** Inspections of caulking or sealant are supplemented with physical manipulation. Surface exams are conducted to detect cracking for the stainless steel U2 RWT. Thickness measurements of tank bottoms are conducted to detect degradation (e.g., loss of material on the inaccessible external surface). Inspections are conducted in accordance with ASME Code Section XI requirements as applicable or are conducted in accordance with plant-specific procedures that include inspection parameters such as lighting, distance, offset, and surface conditions.

Associated Enclosures:

None.

Selective Leaching AMP – Justification for Number of Inspections

RAI B.2.3.21-1

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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 to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

SLRA Table 3.3.2-6, “Fire Protection / Service Water – Summary of Aging Management Evaluation,” states that loss of material due to selective leaching for gray cast iron piping exposed to soil will be managed by the Selective Leaching program.

SLRA Section B.2.3.21, “Selective Leaching,” states the following:

> For raw water, waste water, and soil environments, the AMP includes opportunistic and periodic visual inspections of selected components that are susceptible to selective leaching, coupled with mechanical examination techniques. Destructive examinations of components to determine the presence of and depth of dealloying through-wall thickness are also conducted.

The “plant specific operating experience” summary in SLRA Section B.2.3.21 describes the results of one cast iron fire protection system piping inspection.

NUREG-2222, “Disposition of Public Comments on the Draft Subsequent License Renewal Guidance Documents NUREG–2191 and NUREG–2192,” states the following regarding the staff’s basis for reducing the extent of inspections for selective leaching during the subsequent period of extended operation (i.e., 3 percent with a maximum of 10 components per GALL-SLR guidance) when compared to the extent of inspections for selective leaching during the initial period of extended operation (i.e., 20 percent with a maximum of 25 components per GALL Report, Revision 2 guidance):

1. Opportunistic inspections will be conducted throughout the period of extended operation whenever components are opened, buried, or submerged surfaces are exposed, whereas opportunistic inspections were not recommended in the previous version of AMP XI.M33;

2. Destructive examinations provide a more effective means to detect and quantify loss of material due to selective leaching;

3. The slow growing nature of selective leaching generally coupled *with the inspections conducted prior to the initial period of extended operation* [emphasis added by the staff] provides insights into the extent of loss of material due to selective leaching that can be used in the subsequent period of extended operation;

4. The staff's review of many license renewal applications has not revealed any instances where loss of intended function has occurred due to selective leaching;

5. The staff's review of industry OE [operating experience] has not detected any instances of loss of material due to selective leaching, which resulted in a loss of intended function for the component; and

6. Regional inspector input (provided based on IP 71003, "Post-Approval Site Inspection for License Renewal,") that selective leaching has been noted during visual and destructive inspections; however, no instances have been identified where there was the potential for loss of intended function.

The NRC issued Information Notice (IN) 2020-04, "Operating Experience Regarding Failure of Buried Fire Protection Main Yard Piping," to inform the industry of OE involving the loss of function of buried gray cast iron fire water main yard piping due to multiple factors, including graphitic corrosion (i.e., selective leaching), overpressurization, low-cycle fatigue, and surface loads. As noted in the IN, a contributing cause to the failures of buried gray cast iron piping at Surry Power Station (SPS) was the external reduction in wall thickness at several locations due to graphitic corrosion.

Issue:

The recommended extent of inspections in GALL-SLR AMP XI.M33 are based on the six conditions noted by the staff in NUREG-2222. The staff's comparison of these six conditions to the Selective Leaching program at St. Lucie Nuclear Plant (PSL) follows:

- Based on its review of SLRA Section B.2.3.21, the staff notes that opportunistic inspections and destructive examinations for selective leaching will be performed, consistent with the first and second conditions in NUREG-2222.

- Based on its review of plant-specific operating experience in SLRA Section B.2.3.21, one selective leaching inspection has been conducted for gray cast iron piping. Based on this observation (i.e., multiple inspections for buried gray cast iron piping may not have been performed prior to the initial period of extended operation), the third condition in NUREG-2222 may not be met at PSL for gray cast iron piping exposed to soil.

- The fourth, fifth, and sixth conditions in NUREG-2222 focus on the staff's review of industry OE not identifying any instances of loss of material due to selective leaching which had resulted in a loss of intended function for the component. Based on recent industry OE at SPS (as documented in IN-2020-04), the last three conditions in NUREG-2222 are no longer applicable for gray cast iron piping exposed to soil. Since these conditions are no longer applicable (i.e., there is now industry OE involving loss of material due to selective leaching which resulted in a loss of intended function for gray cast iron piping exposed to soil), the staff requires additional information to determine if the reduced extent of inspections in GALL-SLR AMP XI.M33 are appropriate for this material and environment combination.

Request:

Provide additional OE (or other technical justification) to demonstrate that the extent of inspections in GALL-SLR AMP XI.M33 (i.e., 3 percent with a maximum of 10 components) are appropriate for gray cast iron piping exposed to soil.

PSL Response:

The PSL Selective Leaching AMP is a new AMP for SLR. For initial License Renewal (LR), a selective leaching program was not required. In lieu of a selective leaching program, other aging management programs (e.g., the LR fire protection program) were used to detect loss of material due to various aging mechanisms including selective leaching through system walkdowns, flow testing, etc.

As a result of a GALL Gap Analysis performed during the PSL LR implementation, an opportunistic selective leaching inspection was performed on a buried fire protection line that had cracked, just below where it came out of the ground. As stated in the SLRA, this examination found that selective leaching was not a significant contributor to the pipe failure.

Additional word searches of the PSL 10-year action request (AR) database were performed and have been uploaded to the ePortal. The word searches were as follows:

- A search for "graphit" identified 12 ARs, of which, none were related to graphitic corrosion nor selective leaching.
- A search for "cast iron" and "leach" yielded only 1 AR, which was related to the opportunistic selective leaching inspection of the fire protection system piping near the north warehouse.
- A search for "cast iron" identified 28 ARs. Of the 28 ARs, 8 were unique ARs related to corrosion or pitting of cast iron and only 1 of these 8 ARs was related to the fire protection system. Additionally, 1 of the 28 ARs was related to the opportunistic selective leaching inspection of the fire protection system piping near the north warehouse.

Plant specifications state that the buried cast iron fire protection system piping is externally coated with a coal tar coating and internally lined with a cement lining. Additionally, backfill is per the original site design specifications. The Class I backfill has no more than 12% (for Unit 1) or 15% (for Unit 2) silt content (finer than No. 200 sieve), which meets and exceeds the recommended ASTM D 448-08 size recommendation from NUREG-2191. These preventive actions effectively reduce the occurrence and significance of loss of material due to selective leaching.

Comparison to Surry Environment and Conditions

As stated in the RAI Background, a buried cast iron fire protection pipe rupture event was identified at Surry Power Station (Reference 1). The respective INPO report identified two fracture locations (a northern section of pipe and a southern section of pipe). Both locations had significant graphitic corrosion (selective leaching) at the bottom of the pipe from where the cracking propagated.

- For the northern crack location, the hoop stress due to internal water pressure eventually caused a longitudinal crack in the piping. The graphitic corrosion had significantly reduced the strength of the piping since the pipe design pressure was 200 psig and the fire protection loop was routinely pressurized to approximately 135 psig.

- For the southern crack location, the piping was cracked circumferentially due to bending stress. The southern location was beneath a road where overload conditions may have contributed to the bending stress. The upward thrust generated at the bottom of the northern pipe rupture could have possibly caused this damage.

Several sections of removed Surry fire protection piping were later inspected using a surface abrasive blasting technique and further significant graphitic corrosion was identified, including areas with minimal remaining wall thickness.

The primary factors that caused the significant graphitic corrosion at Surry were as follows:

- The piping had been exposed to a wet soil environment for an extended period of time caused by a nearby fire protection valve packing leak.

- The piping had an inadequate/incorrect coating (thin asphaltic coating) applied.

The PSL buried fire protection system mitigates the risk of graphitic corrosion through the following:

- The PSL buried cast iron fire protection piping is coated with coal tar enamel, which provides an adequate barrier between the cast iron piping and the soil.

- The PSL soil generally has low or undetectable levels of sulfates, as identified in the soil samples within the RAI B.2.3.21-2 response. Per the Surry INPO Industry Reporting and Information System (IRIS) report, graphitic corrosion favors soil environments with higher levels of sulfates and moisture.

- The PSL buried cast iron fire protection piping is not exposed to continuously wetted soil. The soil samples identified in the response to RAI B.2.3.21-2 show that soil moisture at the PSL site was an average of 8.9 percent and a maximum of 11.9 percent. The piping is buried above the water table. Additionally, the fire water header pressure is continuously monitored with alarm setpoints, so that when leaks occur, they are corrected in a timely manner, reducing the risk for graphitic corrosion.

In conclusion, the environmental conditions (soil with low moisture and low sulfates) and preventive and leak detection actions (backfill, coal tar coating, and continuous pressure monitoring) greatly reduce the risk of graphitic corrosion (selective leaching) of the buried cast iron piping at PSL. Additionally, both the positive OE (opportunistic test showing insignificant selective leaching) and the lack of negative OE for the buried cast iron fire protection system piping demonstrate the effectiveness of the existing preventive actions (e.g., pipe coatings and backfill). Therefore, with respect to gray cast iron piping exposed to soil, adequate justification exists to use the number of visual and mechanical inspections recommended by the latest version of NUREG-2191 AMP XI.M33, i.e., 3 percent with a maximum of 8 components per unit per 10-year interval (reduced from 10 components due to being a 2-unit site).

References:

1. NRC Information Notice (IN) 2022-04, Operating Experience Regarding [Related to] Failure of Buried Fire Protection Main Yard Piping, U.S. Nuclear Regulatory Commission, (ADAMS Accession Number ML20223A333)

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

Selective Leaching AMP – Soil Sampling

RAI B.2.3.21-2

Regulatory Basis:

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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 to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

UFSAR Section 19.2.2.21, "Selective Leaching," states "[w]here the sample size is not based on the percentage of the population and the inspections will be conducted periodically (not one-time inspections), a reduction in the total number of inspections is acceptable as follows. Eight visual and mechanical inspections (reduced from 10 visual and mechanical inspections) and two destructive examinations will be conducted…"

GALL-SLR Report AMP XI.M33 states the following:

> For multi-unit sites where the sample size is not based on the percentage of the population and the inspections are conducted periodically (not one-time inspections), it is acceptable to reduce the total number of inspections at the site as follows. For two unit sites, eight visual and mechanical inspections and two destructive examinations are conducted at each unit…[i]n order to conduct the reduced number of inspections, the applicant states in the SLRA the basis for why the operating conditions at each unit are similar enough (e.g., flowrate, chemistry, temperature, excursions) to provide representative inspection results.

SLRA Section B.2.3.27, "Buried and Underground Piping and Tanks," states the following:

> "[d]uring excavations, many soil samples have been obtained and analyzed. In general, the pH at PSL is approximately 9.0 indicating an alkaline soil environment. Resistivity of samples typically range from 1700 to 5000 ohm-cm. Due to the consistency of soil samples, future soil sampling was determined to not be warranted."

GALL-SLR Report Table XI.M41-2, "Inspection of Buried and Underground Piping and Tanks," states soil is tested for soil resistivity, corrosion accelerating bacteria, pH, moisture, chlorides, sulfates, and redox potential.

Issue:

The staff requests additional information with respect to how the soil environment is consistent between both units. The SLRA does provide some discussion on soil testing; however, the staff notes that details such as the number of soil samples taken and proximity of these soil samples with respect to in-scope buried piping susceptible to selective leaching are not provided. The staff also notes that in addition to pH and soil resistivity, GALL-SLR Report Table XI.M41-2 recommends that soil is tested for corrosion accelerating bacteria, moisture, chlorides, sulfates, and redox potential.

Request:

Provide additional information demonstrating how the soil environment is consistent between both units. Alternatively, revise the SLRA as appropriate to reflect that the multi-unit site sample size reduction will not be used for components exposed to a soil environment.

PSL Response:

The soil environment in contact with the buried piping at both St. Lucie Units 1 and 2 is composed of backfill in accordance with plant specifications which are available on the ePortal. The primary requirements for backfill material per these specifications are summarized as follows (minor editorial differences between the two units' specifications):

- Material to be used as compacted backfill shall be a selected sand. It shall be free of muddy material, organic matter, rubbish, debris, or other unsuitable materials. The moisture content of the sand shall be within the limits required to obtain the specified compaction. Dredged material shall be stockpiled so as to facilitate drainage. No limerock shall be used for fill without approval of the soils engineer.

- Backfill material designated as Class I material shall have no more than 12% (for Unit 1) or 15% (for Unit 2) silt content (finer than No. 200 sieve), be free of clay balls, and no rock fragments larger than 6 inches shall be used for the fill except in areas where hand compaction is required wherein the maximum rock fragment size shall not exceed 3 inches. Sieve analyses are performed in accordance with ASTM D422 or D1140. Only Class I backfill is used at St. Lucie.

Soil sampling was performed for a number of locations across the St. Lucie Units 1 and 2 site from 2011 through 2014 as part of the NEI 09-14 buried piping program. Soil testing results are listed in the table below:

Sample Number	pH at 25°C	Soil Resistivity (ohms-cm dry)	Moisture (%)	Chlorides (mg/kg dry)	Sulfates (mg/kg dry)	Redox Potential (mV dry)	Microbiology
[U01] ACW 30	9.2	3100	8.0	Not Detected (<136)	Not Detected (<136)	Not Reported	Positive; Aerobic Iron-oxidizing Bacteria (AIOB), Other Bacteria (OB)
U01-CW-400	8.9	3420	9.7	Not Detected (<275)	Not Detected (<275)	77.4	Positive; OB
[U01] BCW 29	9.2	4050	4.3	Not Detected (<131)	Not Detected (<131)	Not Reported	Positive; AIOB, OB
U01 DIG #1 CW-90	8.8	1850	11.9	205	149	3.7	Positive; AIOB, OB
U01 DIG #2 [CW-90]	9.0	3380	11.4	Not Detected (<282)	Not Detected (<282)	130.6	Positive; AIOB
U01-WM-E01	9.2	5010	7.5	Not Detected (<269)	Not Detected (<269)	-5.0	Positive; AIOB, OB
U02-WM-A29	9.1	3240	9.8	Not Detected (<277)	Not Detected (<277)	205.4	Positive; AIOB, OB
Average	**9.1**	**3435**	**8.9**	**225***	**217***	**82**	**Positive; AIOB and/or OB**

* The limit of detectability was conservatively assumed for computing the average chloride and sulfate values.

The soil samples are obtained from soil adjacent to piping east of the plant, except for the CW-90 samples which are south of Unit 1, between the refueling water storage tank and the intake structure.

The table does not include an outlier soil sample associated with a Unit 2 intake cooling water (ICW) line, since that sample had been impacted by local saltwater foaming from the ICW discharge overflow/standpipe. The caulking between the standpipe and the concrete decking was weathered and porous, allowing saltwater foam to permeate into the soil. The soil sample was taken during the SL2-20 replacement of the respective standpipe; therefore, the associated caulking was also replaced. Since there is no SLR-scope buried cast iron piping within the vicinity of this discharge standpipe, additional sampling of cast iron piping is not necessary.

Per EPRI guidance, factors affecting the corrosiveness of soils include moisture, alkalinity, acidity, permeability of water and air (compactness or texture), and levels of oxygen, salts, and biological organisms. Many of these factors affect the electrical resistivity of soil, which is a good measure of corrosivity. Values less than 1,000 ohm-cm are highly corrosive and values greater than 20,000 ohm-cm are progressively less non-corrosive. (Reference 1)

The resistivity of the soil samples taken at St. Lucie ranged from 1850 to 5010 ohm-cm, which is corrosive (Reference 1). The range of resistivity values is not extreme. Therefore, based on samples presented above, the soil environment is consistent between the St. Lucie Unit 1 and 2 power blocks. Likewise, the sample size reduction for the buried piping selective leaching inspections is justified because soil sampling shows that the environments between the two units are comparable with respect to factors that impact corrosivity.

References:

1. EPRI Report 1025256, A Literature of Soil-Side Corrosion Rates for Buried Metallic Piping, Electric Power Research Institute, Palo Alto, California, September 2012.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

External Surfaces Monitoring of Mechanical Components AMP

RAI B.2.3.23

<u>Regulatory Basis:</u>

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One finding that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to 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>

In the program basis document (NEESL00008-REPT-067), "Subsequent License Renewal Aging Management Program Basis Document - External Surfaces Monitoring of Mechanical Components," Section 3.2 "Procedural Controls," states that the program is governed by procedure ADM-17.33, "License Renewal Systems/ Programs Monitoring."

Section 7.0, "Summary of Implementing Documents," of the program basis document identifies the procedures that implement the External Surfaces Monitoring of Mechanical Components (External Surfaces) AMP and includes an extensive list of changes for ADM-17.33 that correspond to the program enhancements for the External Surfaces AMP listed in the SLRA.

The staff notes that in the program basis document, the last action listed in the Section 7.0 table for ADM-17.33 is "Revise walkdown inspection forms to identify new requirements and components to be inspected." The staff notes that Attachments 1 through 21 of ADM-17.33 define the specific walkdown scope of the program's comprehensive condition monitoring and that the license renewal walkdowns shall be performed in accordance with these Attachments 1 through 21.

<u>Issue:</u>

Except for the last action in the Section 7.0 table noted above, all of the other actions describing changes to ADM-17.33 have corresponding enhancements discussed in the SLRA Section B.2.3.23, "External Surfaces Monitoring of Mechanical Components," with corresponding commitments in SLRA Table 19-3, Item No. 26, commitments a) through o). The staff notes that in the current version of ADM-17.33, Attachments 1 through 21 do not include all the components that were designated as being included in the External Surfaces AMP.

It is not clear to the staff why the action to revise the walkdown inspections forms found in Attachments 1 through 21 in ADM-17.33, as described in the program basis document, does not have a corresponding enhancement and commitment.

Request:

Provide a basis to clarify the apparent lack of an enhancement and commitment discussed above, or alternatively revise the SLRA to include an enhancement in B.2.3.23 and a commitment in Table 19-3, Item 26 to include revising the walkdown inspection forms in ADM-17.33 to identify new requirements and components to be inspected.

PSL Response:

A new commitment p) is added to Table 19-3, Item 26 of Appendices A1 and A2 on pages A1-95 and A2-96 respectively, to include revising the walkdown inspection forms to identify new requirements and components to be inspected. A new bullet is also added to the enhancement for element 4 in Section B.2.3.23 on page B-195.

References:

None.

Associated SLRA Revisions:

SLRA Appendix A1, Section 19.4, Table 19-3, Commitment No. 26 on pages A1-90 through A1-95 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
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	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

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
			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”), • 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	

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
			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	

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No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			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	

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No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			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	

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List of Unit 1 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			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	

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
			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 will not meet acceptance criteria prior to the next scheduled inspection, will have their inspection frequencies adjusted as determined by the corrective action program. p) **Revise walkdown inspection forms to identify new requirements and components to be inspected.**	

SLRA Appendix A2, Section 19.4, Table 19-3, Commitment No. 26 on pages A2-90 through A2-96 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
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	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

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List of Unit 2 SLR Commitments and Implementation Schedule

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			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”), • 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.	

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
			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	

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
			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	

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
			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	

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
			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,	

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
			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 will not meet acceptance criteria prior to the next scheduled inspection, will have their inspection frequencies adjusted as determined by the corrective action program. **p) Revise walkdown inspection forms to identify new requirements and components to be inspected.**	

SLRA Appendix B, Section B.2.3.23, Enhancements Table, Element Affected No. 4 (Detection of Aging Effects) on pages B-192 through B-196 is revised as follows:

Element Affected	Enhancement
1. Scope of Program	• Revise procedure to indicate the material and environment combinations where external examinations could be credited to manage the aging effects of the internal surfaces of components. • Revise the PSL External Surfaces Monitoring of Mechanical Components AMP procedure to incorporate the aging management activities currently performed for external corrosion of insulated piping at PSL.
3. Parameters Monitored or Inspected	• Revise procedures to ensure all components made of SS, 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. • Revise procedure to 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: o Surface cracking, crazing, scuffing, and dimensional change (e.g., “ballooning” and “necking”), o Loss of thickness, o Discoloration (evidence of a potential change in material properties that could be indicative of polymeric degradation), o Exposure of internal reinforcement for reinforced elastomers, o Hardening as evidenced by a loss of suppleness during manipulation where the component and material are appropriate to manipulation.
4. Detection of Aging Effects	• Revise procedure to 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,

Element Affected	Enhancement
	physical manipulation will be used to manage hardening or loss of strength and reduction in impact strength. • Revise procedure to 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. • Revise procedures to ensure that periodic visual inspections or surface examinations will be conducted on components made of SS, 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. • Revise procedure to 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-foot 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). • Revise procedure to 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, SS, copper alloy, aluminum) and environment (e.g., air outdoor,

<table>
<tr><th>Element Affected</th><th>Enhancement</th></tr>
<tr><td></td><td>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-foot 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-foot axial length sections and components for each material type is inspected, with a maximum of 25 inspections required in each population.

Revise procedure to 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.
Revise procedure to indicate that the following alternatives to removing insulation after the initial inspection will be acceptable:
<ol type="a">
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:
<ol type="i">
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

</td></tr>
</table>

Element Affected	Enhancement
	insulation should continue as conducted for the initial inspection. b. Removal of tightly adhering insulation that is impermeable to moisture is not required unless there is evidence of damage to the moisture barrier. If the moisture barrier is intact, the likelihood of corrosion under insulation 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. • **Revise walkdown inspection forms to identify new requirements and components to be inspected.**
5. Monitoring and Trending	• Revise procedure to 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.
6. Acceptance Criteria	• Revise procedure to include evaluation and acceptance guidance from EPRI TR-1009743 (Reference 1.6.43), "Aging Identification and Assessment Checklist," for visual/tactile inspections where appropriate.
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

Element Affected	Enhancement
	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 will not meet acceptance criteria prior to the next scheduled inspection, will have their inspection frequencies adjusted as determined by the corrective action program.

Associated Enclosures:

None.

Leak-Before-Break of Reactor Coolant System Piping TLAA

RAI 4.7.1-1

Regulatory Basis:

Pursuant to 10 CFR 54.21(c), the SLRA shall include an evaluation of time-limited-aging analyses (TLAAs). The applicant shall demonstrate that (i) the analyses remain valid for the [subsequent] period of extended operation; (ii) the analyses have been projected to the end of the [subsequent] period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the [subsequent] period of extended operation.

Background:

SLRA Section 4.7.1 “Leak-Before-Break of Reactor Coolant System Piping,” identifies Alloy 600/82/182 welds are susceptible to primary water stress corrosion cracking (PWSCC) and have been conservatively evaluated to consider the effects of PWSCC. Section 3.6.3 of NUREG-0800 states that PWSCC is considered an active degradation mechanism in Alloy 600/82/182 materials in PWR’s and needs to be addressed.

Issue:

It is not clear that the applicant’s evaluation is consistent with the corresponding SLRA section 4.7.1 and Section 3.6.3 of NUREG-0800.

Request:

- Please provide additional information to specifically describe what conservative evaluations were made to the Alloy 600/82/182 welds that are present at the PSL Unit 1 and 2 reactor coolant pump (RCP) suction and discharge nozzle that determined PWSCC is not a concern. Additionally, please identify how the applicant is demonstrating that PWSCC is not a potential source of pipe rupture as described in Standard Review Plan (SRP 3.6.3), Revision 1.

- Please provide additional information if PSL is considering an overlay of Alloy 690/52/152 to minimize the susceptibility to PWSCC based on the evaluations made to the Alloy 600/82/182 welds. If PSL is not considering an Alloy 690/52/152 overlay, please provide additional information to identify how the applicant is planning to monitor these welds for potential leakage from cracks or flaws.

Please revise the TLAA to include the requested information provided above.

PSL Response:

The responses in the bullets below are presented in the same order as the bullets in the request.

- The leak-before-break (LBB) evaluations of the St. Lucie Units 1 and 2 Alloy 82/182 locations at the reactor coolant pump (RCP) suction and discharge nozzles include a conservative factor of 1.69 on the leakage flaw size, which increases the leakage flaw size for the required margin of 10 on the leak rate. This factor accounts for the PWSCC morphology characteristics (e.g., surface roughness and number of turns), on the leakage rate of a given leakage crack size. This methodology is consistent with other approved LBB analyses (References ML14209A027, ML110410119, and ML21354A196). The evaluations for these locations with the conservative factor meet the required margins for leak-before-break per the Standard Review Plan (SRP) 3.6.3 Revision 1, as documented in WCAP-18167-P (Attachment 5 of Enclosure 5 to Reference ML21215A315) and WCAP-18617-NP (Attachment 10 to Reference ML21215A320). Additionally, Table 7-1 of EPRI Technical Report 1011808, MRP-140 (Reference 1), shows long periods of time for PWSCC growth for nickel-based alloy material in relation to LBB analyses. Also, the St. Lucie Units 1 and 2 Technical Specifications (TS) specify actions which require a reactor shutdown in the event of reactor coolant pressure boundary (RCPB) through-wall leakage. Considering the long periods of time for crack growth from a leakage crack size to a critical crack size and TS-required action for RCPB through-wall leakage, sufficient time is available for the flaw to be identified and for the reactor to be shut down.

- All St. Lucie Units 1 and 2 Alloy 600/82/182 components/welds in higher temperature locations have either been mitigated or replaced with PWSCC resistant materials. The only exceptions are the lower temperature St. Lucie Units 1 and 2 RCP suction and discharge nozzle Alloy 82/182 dissimilar metal welds. Due to the low susceptibility of PWSCC in these lower temperature applications, there are no plans to mitigate these Alloy 82/182 dissimilar metal welds.

  For SLR, the Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components AMP (SLRA Section B.2.3.5) will continue to manage the aging effect of PWSCC for St. Lucie Units 1 and 2 RCP suction and discharge nozzle dissimilar metal welds. This AMP is used in conjunction with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP (SLRA Section B.2.3.1), Boric Acid Corrosion AMP (SLR Section B.2.3.4), and Water Chemistry AMP (SLR Section B.2.3.2).

  The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP is a condition monitoring AMP that imposes in-service inspection requirements for ASME Class 1, 2 and 3 pressure retaining components and integral attachments. ASME Code Case N-770-5 (Reference 2) currently provides the requirements for visual and volumetric examination of the St. Lucie Units 1 and 2 RCP suction and discharge nozzle dissimilar metal welds. The Boric Acid Corrosion AMP is credited for the identification, evaluation, and corrective actions for potential borated water leaks in the St. Lucie Units 1 and 2 RCP suction and discharge nozzle dissimilar metal welds. Walkdowns for the detection of boric acid leakage from these locations are performed every outage during plant cooldown and heatup. The main objective of the Water Chemistry AMP (SLR Section B.2.3.2) with regard to the St. Lucie Units 1 and 2 RCP suction and discharge nozzles is to mitigate cracking of the dissimilar metal welds due to SCC and related mechanisms when exposed to a treated water environment. These AMPs are informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry OE, including research and development, such that the effectiveness of the AMPs is evaluated consistent with the discussion in NUREG-2191, Appendix B.

As an added measure of safety, the industry imposed an NEI 03-08 “needed” requirement, to improve their RCS leak detection capability in part due to the concern with PWSCC of Alloy 600 materials. St. Lucie Units 1 and 2 have adopted the standardized approach to measuring RCS leak rate in WCAP-16423 (Reference ML070310084) and has incorporated the action levels in WCAP-16465 (Reference ML070310082). The enhanced leak rate monitoring and detection procedure monitors specific values of unidentified leakage, seven day rolling average, and baseline means. Action levels are initiated as low as when the unidentified leak rate exceeds 0.1 gpm. The enhanced leak detection capability provides an increased level of safety that if a flaw were to grow through wall, although unlikely, it would be detected prior to it growing to a safety significant size.

SLRA Section 4.7.1 is revised to include this information.

References:

1. EPRI Technical Report 1011808, “Material Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds (MRP-140)”, November 2005

2. ASME BPV Code Case N–770–5. ASME BPV Code Case N–770–5, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1" (Approval Date: November 7, 2016)

Associated SLRA Revisions:

SLRA Section 4.7.1, page 4.7-2, is revised as follows:

For the critical locations, flaws are identified that will be stable because of the ample margins described in f, g, and h above.

The LBB analysis results for the RCP suction and discharge nozzle safe-end locations are acceptable for A351-CF8M CASS material from thermal aging effect and for Alloy 82/182 dissimilar metal weld material from PWSCC effect. All the LBB criteria are satisfied. The results for the reactor coolant loop remaining locations not evaluated in WCAP-18617-NP and WCAP-18617-P remain bounded by the analysis of record, CEN-367-A.

The leak-before-break (LBB) evaluations of the St. Lucie Units 1 and 2 Alloy 82/182 locations at the reactor coolant pump (RCP) suction and discharge nozzles include a conservative factor of 1.69 on the leakage flaw size, which increases the leakage flaw size for the required margin of 10 on the leak rate. This factor accounts for the PWSCC morphology characteristics (e.g., surface roughness and number of turns) on the leakage rate of a given leakage crack size. This methodology is consistent with other approved LBB analyses (References ML14209A027, ML110410119, and ML21354A196). The evaluations for these locations with the conservative factor meet the required margins for leak-before-break per the Standard Review Plan (SRP) 3.6.3 Revision 1, as documented in WCAP-18167-P and WCAP-18617-NP. Additionally, Table 7-1 of EPRI Technical Report 1011808, MRP-140 (Reference 4.8.50), shows long periods of time for PWSCC growth for nickel-based alloy material in relation to LBB analyses. Also, the St. Lucie Units 1 and 2 Technical Specifications (TS) specify actions which require a reactor shutdown in the event of reactor coolant pressure boundary (RCPB) through-wall leakage. Considering the long periods of time for crack growth from a leakage crack size to a critical crack size and TS-required action for RCPB through-wall leakage, sufficient time is available for the flaw to be identified and for the reactor to be shut down.

All St. Lucie Units 1 and 2 Alloy 600/82/182 components/welds in higher temperature locations have either been mitigated or replaced with PWSCC resistant materials. The only exceptions are the lower temperature St. Lucie Units 1 and 2 RCP suction and discharge nozzle Alloy 82/182 dissimilar metal welds. Due to the low susceptibility of PWSCC in these lower temperature applications, there are no plans to mitigate these Alloy 82/182 dissimilar metal welds.

For SLR, the Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components AMP (SLRA Section B.2.3.5) will continue to manage the aging effect of PWSCC for St. Lucie Units 1 and 2 RCP suction and discharge nozzle dissimilar metal welds. This AMP is used in conjunction with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP (SLRA Section B.2.3.1), Boric Acid Corrosion AMP (SLR Section B.2.3.4), and Water Chemistry AMP (SLR Section B.2.3.2).

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD AMP is a condition monitoring AMP that imposes in-service inspection requirements for ASME Class 1, 2 and 3 pressure retaining components and integral attachments. ASME Code Case N-770-5 currently provides the requirements for visual and volumetric examination of the St. Lucie Units 1 and 2 RCP suction and discharge nozzle dissimilar metal welds. The Boric Acid Corrosion AMP is credited for the identification, evaluation, and corrective actions for potential borated water leaks in the St. Lucie Units 1 and 2 RCP suction and discharge nozzle dissimilar metal welds. Walkdowns for the detection of boric acid leakage from these locations are performed every outage during plant cooldown and heatup. The main objective of the Water

Chemistry AMP (SLR Section B.2.3.2) with regard to the St. Lucie Units 1 and 2 RCP suction and discharge nozzles is to mitigate cracking of the dissimilar metal welds due to SCC and related mechanisms when exposed to a treated water environment. These AMPs are informed and enhanced when necessary, through the systematic and ongoing review of both plant-specific and industry OE, including research and development, such that the effectiveness of the AMPs is evaluated consistent with the discussion in NUREG-2191, Appendix B.

As an added measure of safety, the industry imposed an NEI 03-08 "needed" requirement, to improve their RCS leak detection capability in part due to the concern with PWSCC of Alloy 600 materials. St. Lucie Units 1 and 2 have adopted the standardized approach to measuring RCS leak rate in WCAP-16423 (Reference ML070310084) and has incorporated the action levels in WCAP-16465 (Reference ML070310082). The enhanced leak rate monitoring and detection procedure monitors specific values of unidentified leakage, seven day rolling average, and baseline means. Action levels are initiated as low as when the unidentified leak rate exceeds 0.1 gpm. The enhanced leak detection capability provides an increased level of safety that if a flaw were to grow through wall, although unlikely, it would be detected prior to it growing to a safety significant size.

SLRA Section 4.8, page 4.8-4, is revised as follows:

4.8.50 ASME EPRI Technical Report 1011808, "Material Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds (MRP-140)", November 2005

Associated Enclosures:

None.

Reactor Head Closure Stud Bolting Examination Results

RCI B.2.3.3-1

Regulatory Basis:

Part 54 of Title 10 of the Code of Federal Regulations (10 CFR), “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” is designed to elicit application information that will enable the U.S. Nuclear Regulatory Commission (NRC) staff to perform an adequate safety review and the Commission to make the necessary findings. Reliability of application information is important and advanced by requirements that license applications be submitted in writing under oath or affirmation and that information provided to the NRC by a license renewal applicant or required to be maintained by NRC regulations be complete and accurate in all material respects. Information that must be submitted in writing under oath or affirmation includes the technical information required under 10 CFR 54.21(a) related to assessment of the aging effects on structures, systems, and components subject to an aging management review. Thus, both the general submission requirements for license renewal applications and the specific technical application information requirements require that submission of information material to NRC’s safety findings (see 10 CFR 54.29 standards for issuance of a renewed license) be submitted by an applicant as part of the application.

Background:

By letter dated August 3, 2021 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML21215A314), as revised by letter dated October 12, 2021 (ADAMS Accession No. ML21285A107) and supplemented by letters dated April 7, 2022 (ADAMS Accession No. ML22097A202), May 12, 2022 (ADAMS Accession No. ML22139A083), Florida Power & Light Company (FPL or the applicant) submitted an application for the subsequent license renewal of Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie), to the U.S. Nuclear Regulatory Commission (NRC). FPL submitted the application pursuant to Title 10 of the Code of Federal Regulations Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” for subsequent license renewal.

Between October 4, 2021 and February 25, 2022, the NRC staff conducted audits of FPL’s records to confirm information submitted in the St. Lucie subsequent license renewal application.

Request:

During the audit, the staff reviewed several documents that contain information which will likely be used in conclusions documented in the Safety Evaluation Report (SER). To the best of the staff's knowledge, this information is not on the docket. Any information used to reach a conclusion in the SER must be included on the docket by the applicant. We request that you submit confirmation that the information gathered from the documents and listed below is correct or provide the associated corrected information.

Description:

Based on the review of the plant-specific operating experience in SLRA Section B.2.3.3 and the audit review of the volumetric examination results posted in the ePortal, the staff needs confirmation that there were no relevant indications or issues identified during the ASME Code Section XI IWB volumetric examinations for the 54 reactor closure head studs (Examination Category B-G-1, Item No. B6.20) and the 54 threads-in-flange (Examination Category B-G-1, Item No. B6.40) of each Saint Lucie unit from the last required examinations performed for the units.

Confirm that there were no relevant indications or issues identified during the last volumetric examinations performed, as required by ASME Code Section XI IWB, for the 54 reactor closure head studs (Examination Category B-G-1, Item No. B6.20) and the 54 threads-in-flange (Examination Category B-G-1, Item No. B6.40) of each Saint Lucie unit.

PSL Response:

By copy of this response, FPL confirms that there were no relevant indications or issues identified during the last volumetric examinations performed, as required by ASME Code Section XI IWB, for the three (3) sets of 54 reactor head closure studs and the 54 threads-in-flange (Examination Category B-G-1, Item No. B6.40) of each St. Lucie unit.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.