ST. LUCIE NUCLEAR PLANT, UNITS 1 AND 2
SUBSEQUENT LICENSE RENEWAL APPLICATION (SLRA)
REQUESTS FOR ADDITIONAL INFORMATION
(SET #2)

SAFETY REVIEW

RAI B.2.3.10-2

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

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

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

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

RAI B.2.3.10-3

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

Background
Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ADAMS Accession No. ML22097A202), revised Section 19.2.2.10 of Appendices A1 and A2, Section B.2.3.10 of SLRA Appendix B to clarify that steam generator (SG) tubes not meeting the performance criteria are plugged, not repaired. St. Lucie Units 1 and 2 are not approved for alternate repair criteria or alternate repair methods.

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

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

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

RAI B.2.3.10-4

Regulatory Basis

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

Background

The straight lengths of the tubes in the Unit 1 steam generators (SGs) are supported by lattice grid tube supports and the u-bend region of the tubes is supported by flat fan bars. The straight lengths of the tubes in the Unit 2 SGs are supported by broached support plates, and the u-bend region of the tubes are supported by anti-vibration bars. The Unit 2 SGs also have v-shaped support pads and v-shaped support bars.

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

Issue

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

Request

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

RAI B.2.3.16-1

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

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

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

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

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

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

RAI B.2.3.15-1

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

Background
The implementation schedule for the Fire Protection program in Table XI-01 of NUREG-2191, Volume 2," Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report" (ML17187A204), states, "Program and SLR enhancements, when applicable, are implemented 6 months prior to the subsequent period of extended operation." However, Tables 19-3 in SLRA Appendices A1 and A2 state, in part, that the implementation schedule for the Fire Protection program is "No later than 6 months prior to the SPEO, or no later than the last refueling outage prior to the SPEO…"

Issue

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

Request

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

RAI B.2.3.15-2

Regulatory Basis

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

Background

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

Issue

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

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

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

RAI B.2.3.15-3

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

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

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

Issue
SLRA Supplement 1 appears to be making a commitment to add limitations related to installing screws in holes on doors and frames, however, the supplement did not include an associated

enhancement to the Fire Protection program. In addition, SLRA Supplement 1 did not describe the limitations and discuss their basis.

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

RAI B.2.3.15-4

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

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

Issue
SLRA Table 3.5.2-1 does not cite fire barrier as an intended function for penetrations (mechanical), thermal insulation (type III semi-hot penetrations). In addition, SLRA Section 3.5.2.1.1 does not identify the Fire Protection program for managing aging effects for containment building structures. Therefore, it is not clear whether the penetrations (mechanical),

thermal insulation (type III semi-hot penetrations) have a fire barrier intended function and, consequently, do not need to be included in the Fire Protection Program.

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

RAI B.2.3.15-5

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

Background
Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ML22097A202), revised SLRA Table 3.5.2-6 by adding the Fire Protection program for managing applicable aging effects for various concrete components with a fire barrier intended function in the emergency diesel generator buildings.

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

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

RAI B.2.3.15-6

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

Background
Subsequent License Renewal Application (SLRA) Supplement 1, dated April 7, 2022 (ML22097A202), revised SLRA Table 3.5.2-13 to change the intended function for the concrete curbs (Unit 2, 2B switchgear room) (accessible) from "fire protection" to "fire prevention." SLRA Table 2.1-1, dated August 3, 2021 (ML21215A314 (Package)), defines fire prevention as confining or retarding a fire from spreading.

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

- 3.5-1, 054: The Fire Protection AMP (Aging Management program) also manages cracking due to chemical reaction for structural fire barriers.
- 3.5-1, 066: The Fire Protection AMP also manages cracking and loss of material (spalling, scaling) due to corrosion of reinforcement for structural fire barriers.
- 3.5-1, 067: The Fire Protection AMP also manages cracking and loss of material (spalling, scaling) due to chemical reaction for structural fire barriers.

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

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

RAI 3.5.2.2.1.6-1

Regulatory Basis

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

Background

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

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

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

1. Electrical penetration dissimilar metal welds
2. SS fuel transfer tube, expansion bellows, and flange
3. SS mechanical penetration bellows

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

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

2. Update the SLRA based on the response above.

RAI B.2.3.33-1

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

Background
Subsequent License Renewal Application (SLRA) Section B.2.3.33, “Structures Monitoring,” stated that that groundwater/soil at St. Lucie Nuclear Plant (PSL) is judged to be aggressive (chlorides > 500 ppm).

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

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

Issue
SLRA Section B.2.3.33, as amended by SLRA Supplement 1 dated April 7, 2022 (ADAMS Accession No. ML 22097A202), provides a site-specific enhancement to the “detection of aging effects” program element of the Structures Monitoring program for revising applicable procedures to conduct a baseline visual inspection, pH analysis, a chloride concentration test, and evaluation to address the potential degradation of concrete due to exposure of aggressive chemical attack in groundwater/soil or leaching and carbonation in the water-flowing environment, and to perform the focused periodic inspections and evaluation updates on an interval not to exceed 5 years throughout the SPEO to ensure that aging effects of inaccessible concrete are adequately managed.

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

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

RAI B.2.3.33-2

Regulatory Basis
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to

managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background
Subsequent License Renewal Application (SLRA) Section B.2.3.33, "Structures Monitoring," states that the PSL Structures Monitoring AMP with enhancements will be consistent with exception to the 10 elements of NUREG-2191, Section XI.S6, "Structures Monitoring."

For the "detection of aging effects" program element, the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report Aging Management Program (AMP) states that the program needs to (a) evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas, and (b) examine representative samples of the exposed portions of the below grade concrete, when excavated for any reason.

Issue
During the review of procedure ADM-17.32 "Structures Monitoring Program", it was noted that, in Section 3.1, the current plant procedure includes a provision to ensure that inaccessible, below grade concrete will be visually inspected in accordance with NUREG-1801 (GALL), when excavated for any reason (opportunistic inspections), which effectively addressed item (b) from the GALL-SLR Report for underground structures. However, it is not clear what provision exists in the current procedure to ensure that inaccessible areas are evaluated for acceptability when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (i.e., to effectively address item (a) from the GALL-SLR Report). Therefore, it is not clear how the structures monitoring program will be consistent with the GALL-SLR Report for adequately managing the aging effects of inaccessible structural elements.

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

RAI B.2.3.34-2

Regulatory Basis
Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the

current licensing basis for the period of extended operation. One of the findings that the U.S. Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background
Subsequent License Renewal Application (SLRA) Section B.2.3.34, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," states that the St. Lucie Nuclear Plant (PSL) Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program (AMP), with enhancements, will be consistent without exception to the ten program elements of Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report AMP XI.S7, "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

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

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

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

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

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

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

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

RAI 3.5.2.2.2.1-1

Regulatory Basis

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

Background

Subsequent License Renewal Application (SLRA) Section B.2.3.34, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” stated that the Inspection of Water-Control Structures Associated with Nuclear Power Plants Aging Management Program (AMP), with enhancements will be consistent without exception to the ten program elements of Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report AMP XI.S7, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.”

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

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

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

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

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

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

RAI 3.5.2.2.2.1-2

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

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

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

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

On November 30 - December 1, 2021, the staff performed an on-site audit at PSL to gain a general overview of current conditions of the structures compared to the provided operational experience (OE), and an understanding on the pattern cracking or crazed concrete cracking identified in the Turbine Building and the Reactor Auxiliary Building. During the walkdown of the Turbine Building and the Reactor Auxiliary Building, the staff noticed the pattern cracking or crazed concrete cracks on the top of the concrete roof slab. In the review of operating experience AR 01693560 and AR 01725652, the staff also noted that the applicant initially identified potential ASR issue in the roof protecting the Unit 1 Reactor Auxiliary Building, but the applicant’s subsequent engineering evaluation determined that ASR was not present because the critical ASR characteristics were missing. The ARs also stated that ASR can be visually detected based on the “crazing” pattern on the concrete surfaces, but confirmation requires petrographic analysis.

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

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

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

Request

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

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

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

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

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

RAI 3.5.2.2.2.4-1

Regulatory Basis

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

Background

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

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

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

SLRA Table 3.5.2-1 and Table 3.5.2-7, as amended by Supplement 1, list SS reactor cavity and fuel pool components associated with Table 3.5-1 item 3.5.1-100, with generic note E and plant-specific note 12 and 6, respectively. PSL credits the Water Chemistry AMP and monitoring cavity or pool water levels to manage the aging effects for the pressure boundary components of reactor cavity seal ring and fuel pool gates exposed to an uncontrolled indoor air environment. Also, for the pressure boundary components of refueling cavity liner plate and pool liner plates exposed to an uncontrolled indoor air environment, PSL credits the Water Chemistry AMP and monitoring cavity or pool water levels and leakage from the leak chase channels to manage the aging effects.

<u>Issue</u>

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

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

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

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

Request

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

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

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

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

RAI 4.6-1

Regulatory Basis

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

Background

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

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

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

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

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

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

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

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

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

<u>Issue</u>

1) Based on the audit and information provided in SLRA Sections 4.6 and 3.5.2.2.1.5, as amended by Supplement 1 dated April 7, 2022, the staff does not have sufficient

information to verify that fatigue waiver analyses exist for containment penetration nozzles, personnel airlocks, equipment hatches, and other credited components in SLRA Table 3.5.2-1 as amended, or justified as being included in or bounded by the fatigue waiver evaluations of the steel containment vessel shell, given that material, geometry and stress conditions of these components may be different from that of the containment vessel shell.

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

Request

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

   OR,

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

2) In the summary of results of the fatigue waiver evaluation for the PSL Unit 1 and Unit 2 containments vessel shell provided in the current SLRA Section 4.6 under "TLAA Evaluation" for "Metal Containment Fatigue," state and justify the material based on which the evaluation was performed.

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

RAI 3.5.2.2.2.6-1

Regulatory Basis

Section 54.21(a)(3) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the U.S.

Nuclear Regulatory Commission (NRC) staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background

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

Issue

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

Request

1. Clarify how this evaluation makes the safety conclusion that 20% (or 25% for some components) is sufficiently conservative when projecting fluence estimates to the end of the renewed operating licenses.

2. Identify necessary updates of pertinent areas of the SLRA to reflect this clarification or justify your position for not updating.

RAI 3.5.2.2.2.6-2

Regulatory Basis

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

Background

Subsequent License Renewal Application (SLRA) Section 3.5.2.2.2.7, as amended by Supplement 2 dated April 13, 2022 (ADAMS Accession No. ML22103A014) describes the applicant's further evaluation for loss of fracture toughness due to irradiation embrittlement of the reactor pressure vessel (RPV) structural steel support assemblies of the St. Lucie Nuclear Plant (PSL) units. Specifically, SLRA Page Numbers: 3.5-44, 3.5-45, 3.5-46, and the discussion contained in L-2022-044 Attachment 2 Page 1 of 10 addresses impacts to the RPV sliding plates.

Issue

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

Request

1. Identify the radiation embrittlement basis provided for this manganese bronze alloy component.

2. Clarify whether the manganese bronze component and its Lubrite lubricant has been evaluated for inclusion of gamma ray heating.

3. Identify necessary updates of pertinent areas of the SLRA reflecting this RAI input or justify PSL position for not updating.

RAI 3.5.2.2.2.7-1

Regulatory Basis

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

Background

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

Issue

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

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

RAI 3.5.2.2.2.7-2

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

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

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

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