NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

1. Bolting Integrity, GALL Aging Management Program (AMP) XI.M18

RAI B.2.3.9-1a

Background:

Title 10 of the *Code of Federal Register* (CFR) Section 54.21(a)(1) requires that for those systems, structures, and components within the scope of license renewal that the applicant identify and list those structures and components subject to an aging management review (AMR). Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function will be maintained consistent with the current licensing basis for the period of extended operation (PEO). As described in NUREG-2192, Rev. 0, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), dated July 2017, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," dated July 2017, and when evaluation of the matter in the GALL-SLR Report applies to the plant.

The GALL-SLR Report lists AMR items in a tabular form for structures and components in major plant systems (e.g., reactor coolant systems, engineered safety features, auxiliary systems) found in light-water reactor nuclear power plants. Turkey Point's SLRA contains Table 2s which provide the detailed results of the AMRs for those components identified as subject to an AMR for each of the systems or components within a major plant system grouping in the GALL-SLR Report. SLRA Section B.2.3.9, "Bolting Integrity," states that the Turkey Point Bolting Integrity Program is an existing AMP that will be consistent with enhancements with the GALL-SLR Report AMP XI.M18, "Bolting Integrity."

By letter dated October 4, 2018, the staff requested that the applicant clarify whether high strength (HS) closure bolting material with a yield strength greater than or equal to 150 ksi and a diameter greater than 2 inches would be used at Turkey Point and that if so the applicant was requested to clarify how the aging effects of cracking due to stress corrosion cracking (SCC) will be managed consistent with recommendations in GALL-SLR Report AMP XI.M18. In its response letter dated November 2, 2018, the applicant stated that "[s]ite specifications currently list bolting material ASTM No. SA/A540, Grade B23 Cl.1 as acceptable for use at the site, and therefore, HS closure bolting is assumed to be in use at Turkey Point Units 3 and 4." In its response the applicant also revised SLRA Section B.2.3.9, "Bolting Integrity," to state, in part, the following:

> If closure bolting greater than 2 inches in diameter (regardless of code classification) with actual yield strength greater than or equal to 150 ksi (1,034 MPa) is found and for closure bolting for which yield strength is unknown, volumetric examination in accordance to that of ASME [American Society of Mechanical Engineers] Code Section XI, Table IWB-2500-1, Examination Category

B-G-1, is performed. Specified bolting material properties may be used to determine if the bolting exceeds the threshold to be classified as high strength.

Issue:

The GALL-SLR Report states that for all closure bolting greater than 2 inches in diameter (regardless of code classification) with actual yield strength greater than or equal to 150 ksi (1,034 MPa) and closure bolting for which yield strength is unknown, volumetric examination in accordance with that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, is performed (e.g., acceptance standards, extent and frequency of examination). The applicant stated that this closure bolting material is assumed to be currently installed at Turkey Point; however, the applicant did not state or revise the SLRA to indicate where this closure bolting is located. Further, the applicant did not assign Table 2 AMRs to address SCC for this closure bolting material; this also indicates that the applicant had not identified the specific location of the bolting greater than 2 inches that would need to receive volumetric examinations because their actual yield strength is greater than or equal to 150 ksi (1,034 MPa) or because their yield strength is unknown. The staff noted that the ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, states, in part, that volumetric inspections of flange bolting in piping systems may be limited to one bolted connection among a group of bolted connections that are similar in design, size, function, and service. The ASME Code also states that volumetric examination of bolting for heat exchangers and pumps may be conducted on one heat exchanger or one pump among a group of heat exchangers or pumps that are similar in design, type, and function.

It is not clear how the program will determine the scope of volumetric inspection of all closure bolting greater than 2 inches in diameter with actual yield strength greater than or equal to 150 ksi and closure bolting for which yield strength is unknown consistent with the above criteria, as well as other acceptance standards and extent and frequency of examination in ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, considering the applicant has not identified the location of such bolts.

Request:

1. State how the GALL-SLR Report recommended volumetric inspection of all closure bolting greater than 2 inches in diameter with actual yield strength greater than or equal to 150 ksi and closure bolting for which yield strength is unknown will be performed consistent with the acceptance standards and extent and frequency of examination in ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 during the subsequent period of extended operation (SPEO).

2. Clarify whether the AMR tables (including Table 2s) address the component, material, environment, and aging effects associated with HS closure bolting.

FPL Response

Previous Turkey Point Units 3 and 4 correspondence with the NRC (References 1 and 2) stated that the potential for SCC of fasteners is minimized by using ASTM A193, Gr. B7 bolting material in Auxiliary Systems and Reactor Coolant Systems. As discussed in the response to RAI B.2.3.9-1 in letter L-2018-193 (Attachment 22 of Reference 3), the site maintenance bolting specification allows for the use of ASTM No. SA/A540, Grade B23 Cl.1 high strength bolting. The conservative conclusion made as part of that RAI response was that high strength steel closure bolting greater than 2 inches in diameter could be used for pressure retaining functions in mechanical systems. However, review of other bolting, piping, and tubing specifications along with a second review of component specific references has determined that ASTM No. SA/A540, Grade B23 Cl.1 bolting and other high strength bolting greater than 2 inches in diameter is not acceptable to use as initial or replacement closure bolting in piping, tubing, and other pressure retaining applications at Turkey Point Units 3 and 4. Therefore, it is concluded that reactor vessel closure studs (as discussed in SLRA Table 3.1.2-3 and Section B.2.3.3) represent the only high strength steel bolting greater than 2 inches in diameter allowed by the site maintenance bolting specification.

Additionally, SLRA Section B.2.3.9, "Bolting Integrity" and Table 17-3, item 13 states, per the recommendation of GALL-SLR, that any replacement or new pressure-retaining bolting will have an actual yield strength less than 150 ksi, which ensures ASTM No. SA/A540, Grade B23 Cl.1, or any other high strength closure bolting will not be installed at PTN in the future. The SLRA is amended to remove discussion of aging management for high strength closure bolting for the Bolting Integrity Program that was added in response to RAI B.2.3.9-1.

Also, the enhancement related to prohibiting Molybdenum Disulfide is adjusted for consistency with the response to RAI B.2.3.32-2 (Attachment 9 of this letter) to also prohibit any sulfur containing lubricants.

References:

1. FPL Letter L-2001-50 to NRC dated March 22, 2001, Response to Requests for Additional Information (RAI) for the Review of the Turkey Point Units 3 and 4 License Renewal Application

2. FPL Letter L-2001-76 to NRC dated April 19, 2001, Response to Requests for Additional Information (RAI) for the Review of the Turkey Point Units 3 and 4 License Renewal Application

3. FPL Letter L-2018-193 to NRC dated November 2, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application, Safety Review Requests for Additional Information (RAI) Set 6 Responses

Associated SLRA Revisions

Changes made by Attachment 22 to Reference 1 in response to RAI B.2.3.9-1 are amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions.

Revise SLRA Section B.2.3.9 on page B-103 as follows:

Element Affected	**Enhancement**
1. Scope of Program 4. Detection of Aging Effects	Create a new governing procedure and update existing procedures for this AMP to do the following in accordance with this NUREG-2191 element XI.M18: • Include submerged pressure-retaining bolting in inspections. • Include closure bolting for piping systems that contain air or gas for which leakage is difficult to detect. ~~• Include monitoring of high strength bolting for surface and subsurface discontinuities indicative of cracking. This will be accomplished by performing volumetric examination, in accordance with ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 (e.g., acceptance standards, extent and frequency of examination). Specified bolting material properties (e.g., design and procurement specifications, fabrication and vendor drawings, material test reports) may be used to determine if the bolting exceeds the threshold to be classified as high strength.~~
2. Preventive Actions	Create a new governing procedure and update existing procedures for this AMP to do the following in accordance with this NUREG-2191 element XI.M18: • Ensure any replacement or new pressure-retaining bolting has an actual yield strength less than 150 ksi. • Ensure that lubricants containing molybdenum disulfide **<u>or other lubricants containing sulfur</u>**

Element Affected	Enhancement
	will not be used in conjunction with pressure-retaining bolting.
6. Acceptance Criteria	Include appropriate acceptance criteria for submerged pressure-retaining bolting and closure bolting for piping systems that contain gas or air for which leakage is difficult to detect.

Revise SLRA Section B.2.3.9 paragraph 2 on page B-103 as follows:

Per the GALL-SLR, SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769). Additionally, operating experience and laboratory examinations show that the use of molybdenum disulfide as a lubricant is a potential contributor to SCC. Based on investigation in response to an NRC request for information supporting license renewal for the current PEO (reported in March 2001), **there is currently no high strength bolting within the scope of this program** ~~molybdenum disulfide lubricant is not in use.~~ ~~PTN may have high strength bolting installed. If high strength bolting is found, it shall be monitored for surface and subsurface discontinuities indicative of cracking.~~ The existing activities of this AMP will be enhanced to ensure that no new high strength bolting within the scope of this program will be installed and molybdenum disulfide **or other lubricants containing sulfur** will not be used as a lubricant.

Remove the following paragraph into SLRA Section 17.2.2.9 after the 1st paragraph on page A-18 as follows:

~~If closure bolting greater than 2 inches in diameter (regardless of code classification) with actual yield strength greater than or equal to 150 ksi (1,034 MPa) is found and for closure bolting for which yield strength is unknown, volumetric examination in accordance to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, is performed. Specified bolting material properties may be used to determine if the bolting exceeds the threshold to be classified as high strength.~~

Revise SLRA Table 3.2-1, item 012 as follows:

Table 3.2-1: Summary of Aging Management Evaluations for the Engineered Safety Features

Item Number	Component	Aging Effect / Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2-1, 012	High-strength steel closure bolting exposed to air, soil, underground	Cracking due to SCC; cyclic loading	AMP XI.M18, "Bolting Integrity"	No	**Not applicable.** **There is no high-strength steel closure bolting in the Engineered Safety Features systems.** ~~Consistent with NUREG-2191. High-strength closure bolting may be used in the Engineered Safety Features. If used the Bolting Integrity AMP is used to manage cracking.~~

Revise SLRA Table 3.3-1, item 010 as follows:

Table 3.3-1: Summary of Aging Management Evaluations for the Auxiliary Systems

Item Number	Component	Aging Effect / Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3-1, 010	High-strength steel closure bolting exposed to air, soil, underground	Cracking due to SCC; cyclic loading	AMP XI.M18, "Bolting Integrity"	No	**Not applicable.** **There is no high-strength bolting associated with the Auxiliary Systems.** ~~Consistent with NUREG-2191. High-strength closure bolting may be used in the Auxiliary Systems. If found the Bolting Integrity AMP is used to manage cracking.~~

Revise SLRA Table 3.4-1, item 007 as follows:

Table 3.4-1: Steam and Power Conversion Systems

Item Number	Component	Aging Effect / Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.4-1, 007	High-strength steel closure bolting exposed to air, soil, underground	Cracking due to SCC; cyclic loading	AMP XI.M18, "Bolting Integrity"	No	**Not applicable.** **There is no high-strength steel bolting in the Steam and Power Conversion Systems.** ~~Consistent with NUREG-2191. High-strength closure bolting may be used in the Steam and Power Conversion Systems. If found the Bolting Integrity AMP is used to manage cracking.~~

Revise SLRA Table 17-3, Item 13 as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
13	Bolting Integrity (17.2.2.9)	XI.M18	Continue the existing PTN Bolting Integrity AMP, including enhancement to: a) Inspect submerged pressure-retaining bolting when submerged portions of	No later than 6 months prior to the SPEO, i.e.: PTN3: 1/19/2032

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			components (e.g., pump casings) are overhauled or replaced during maintenance activities; b) Evaluate closure bolting for piping systems that contain air or gas, for which leakage is difficult to detect, on a case-by-case basis through – • Visual inspection during maintenance activities; • Visual inspection for discoloration of nearby external surfaces; • Monitoring and Trending of pressure decay within an isolated boundary; • Soap bubble testing; or • Thermography when fluid temperature is higher than ambient. c) Ensure any replacement or new pressure-retaining bolting has an actual yield strength less than 150 ksi; d) Ensure that lubricants containing molybdenum disulfide **<u>or other lubricants containing sulfur</u>** will not be used in conjunction with pressure retaining bolting; e) Include appropriate acceptance criteria for submerged pressure-retaining bolting and closure bolting for piping systems	PTN4: 10/10/2032

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			that contain gas or air for which leakage is difficult to detect. ~~f) Include monitoring high strength bolting for surface and subsurface discontinuities indicative of cracking. This will be accomplished by performing volumetric examination, in accordance with ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 (e.g., acceptance standards, extent and frequency of examination). Specified bolting material properties (e.g., design and procurement specifications, fabrication and vendor drawings, material test reports) may be used to determine if the bolting exceeds the threshold to be classified as high strength.~~	

Associated Enclosures:

None.

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2. Scoping and Screening – Mechanical Systems

RAI 2.3.1.1-1

Regulatory Basis:

Section 54.21(a)(1) of 10 CFR requires an applicant to identify and list structures and components subject to an AMR.

Issue:

On boundary drawing 5613-M-3041, Sheet 3, "Reactor Coolant System, Reactor Coolant Pumps," the piping to/from the component cooling water for reactor coolant pump (RCP) A (location G4 on the drawing) is shown as not within the scope of license renewal (i.e., not highlighted). This is inconsistent with the piping for the other two RCPs in Unit 3 and all three RCPs in Unit 4.

In addition, the subject piping is connected to the thermal barrier heat exchanger for RCP A. The thermal barrier heat exchanger appears to be included in the RCS components subject to aging management (coil type heat exchanger) as shown in Table 2.3.1-1 of the SLRA, but it is not specifically described with the reactor coolant pump seal discussion in SLRA Section 2.3.1.1.3 or elsewhere in SLRA Section 2.3.1.1.

Request:

1) Given that the RCPs perform a safety related function, verify whether the piping to/from the component cooling water for RCP A is within the scope of license renewal (and should have been highlighted in green on the drawing) in accordance with 10 CFR 54.4(a) and whether it is subject to an AMR in accordance with 10 CFR 54.21(a)(1). If it is not within the scope of license renewal and is not subject to an AMR, the staff requests that the applicant provide justification for the exclusion.
2) Confirm that the thermal barrier heat exchanger is within the scope of license renewal in accordance with 10 CFR 54.4(a) and whether it is subject to an AMR in accordance with 10 CFR 54.21(a)(1). If it is not within the scope of license renewal and is not subject to an AMR, the staff requests that the applicant provide justification for the exclusion.

FPL Response:

1) The component cooling water supply to the RCP A thermal barrier heat exchanger is within the scope of subsequent license renewal (SLR) in accordance with 10 CFR 54.4(a) and subject to an AMR in accordance with 10 CFR 54.21(a)(1). Shading of this piping on boundary drawing 5613-M-3041, Sheet 3, "Reactor Coolant System, Reactor Coolant Pumps," was inadvertently omitted. The subject boundary drawing has been revised to include this piping and is available on the ePortal.

2) The RCP thermal barrier heat exchangers are within the scope of license renewal in accordance with 10 CFR 54.4(a) and subject to an AMR in accordance with 10 CFR 54.21(a)(1). The thermal barrier heat exchangers are included in SLRA Table 2.3.1-1 as the component type "Heat exchanger (tubes and coils)". The aging management review of the component type "Heat exchanger (coil)" is included in SLRA Table 3.1.2-1.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

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RAI 2.3.1.1-2

Regulatory Basis:

Section 54.21(a)(1) of 10 CFR requires an applicant to identify and list structures and components subject to an AMR.

Issue:

On boundary drawing 5613-M-3041, Sheet 2, "Reactor Coolant System," three instrument lines off of the piping downstream of the pressurizer safety relief valves are shown as not within the scope of license renewal (i.e., not highlighted). These lines connect to ZS-3-6303A, ZS-3-6303B and ZS-3-6303C and are located at B3 and B4 on the drawing. These same lines are highlighted (i.e., are within the scope of license renewal) on the equivalent drawing for Unit 4 (5614-M-3041, Sheet 2, "Reactor Coolant System"). It is unclear why the in-scope determination is different for each unit.

Request:

Given that Section 2.3.1 of the SLRA states that the RCS for Units 3 and 4 are essentially identical, verify whether these lines are within the scope of license renewal in accordance with 10 CFR 54.4(a) and whether they are subject to an AMR in accordance with 10 CFR 54.21(a)(1). If they are not within the scope of license renewal and are not subject to an AMR, the staff requests that the applicant provide justification for the exclusion.

FPL Response:

The three instrument lines off of the piping downstream of the pressurizer safety relief valves connected to ZS-3-6303A, ZS-3-6303B and ZS-3-6303C are within the scope of subsequent license renewal (SLR) in accordance with 10 CFR 54.4(a) and are subject to an AMR in accordance with 10 CFR 54.21(a)(1). Shading of these instrumentation lines on boundary drawing 5613-M-3041, Sheet 2, "Reactor Coolant System" was inadvertently omitted. The subject boundary drawing has been revised to include these instrumentation lines and is available on the ePortal.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

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RAI 2.3.2.5-1

Regulatory Basis:

Section 54.21(a)(1) of 10 CFR requires an applicant to identify and list structures and components subject to an AMR.

Issue:

On boundary drawing 5614-M-3050, Sheet 1, "Residual Heat Removal System," residual heat removal (RHR)/low head safety injection (LHSI) Pumps 4A and 4B and their associated piping are shown in Detail 1 and Detail 2 (locations G2 to G4 and H2 to H4) as highlighted in blue. This is inconsistent with the equivalent drawing for Unit 3 (5613-M-3050, Sheet 1, "Residual Heat Removal System") which shows these components highlighted in green. Section 2.1.1 of the SLRA states that "Nonsafety-related mechanical components that are included within the scope of license renewal because component failure could prevent the accomplishment of a safety-related function due to potential physical interaction with safety-related SSCs are shown highlighted in blue."

Request:

Given that these pumps (and associated piping) provide a safety related function as specified in Section 2.3.2.5 of the SLRA, confirm that these components are indeed safety related and should have been highlighted in green (noting they perform a safety related function) and are within the scope of license renewal in accordance with 10 CFR 54.4(a) and whether they are subject to an AMR in accordance with 10 CFR 54.21(a)(1).

FPL Response:

The PTN Unit 4 RHR pump piping within the "B" class boundary shown on Detail 1 and 2 of boundary drawing 5614-M-3050, Sheet 1, "Residual Heat Removal System," is safety related and within the scope of subsequent license renewal (SLR) in accordance with 10 CFR 54.4(a) and subject to an AMR in accordance with 10 CFR 54.21(a)(1). The subject piping was inadvertently highlighted in blue. The subject boundary drawing has been revised to highlight the subject piping green consistent with its safety related classification in accordance with 10 CFR 54.21(a)(1). The subject boundary drawing is now consistent with the equivalent Unit 3 drawing and is available on the ePortal.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

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3. Concrete, GALL AMR XI.S6

RAI 3.5.2.2.2.6-10

Background:

SRP-SLR Section 3.5.2.2.2.6 states that data related to the effects and significance of neutron radiation on concrete mechanical and physical properties is limited, especially for conditions (dose, temperature, etc.) representative of light-water reactor (LWR) plants. The SRP-SLR also states that based on literature review of existing research, a fluence limit of 1×10^{19} neutrons/cm^2 radiation is considered a conservative radiation exposure level beyond which concrete material properties may begin to degrade markedly.

Turkey Point Units 3 and 4 (Turkey Point) SLRA Section 3.5.2.2.2.6, as supplemented by letter dated October 5, 2018, states that the reduction in concrete strength due to neutron fluence would be 10 percent up to a depth of 2.6 inches into the primary shield wall (PSW) based on Turkey Point's calculated neutron fluence of 3.57×10^{19} n/cm^2 at the inner face PSW concrete. The Turkey Point SLRA also relies on research work performed by Murayama (2017) citing Figure 54 "Comparison of observed strength ratio (F_c/F_{co}) and total neutron fluence in preceding research and the present study." The SLRA also states that "[d]ue to the [radiation induced volumetric expansion] RIVE effect, the excessive compressive stress was calculated and the inner side of the concrete (up to 3.14 inches) is considered as yielded (cracked)."

With regard to prior studies made on the effects of radiation in concrete strength, the SLRA states that data presented in studies made by Hilsdorf (1978) and Field, et al. (2015) contained results of specimens tested at varying neutron energy levels (fluence) and temperatures and concluded that "compressive strength appears to begin to decrease at a fluence of approximately 1×10^{19} neutrons/cm^2."

The staff noted that Maruyama's (2017) paper stated that the main reason for degradation due to neutron irradiation is the metamictization of rock-forming minerals in aggregates that leads into aggregate expansion and then cracking of the surrounding concrete. A comparable observation was made by Hilsdorf (1978) which stated that "neutron radiation with a fluence of more than 1×10^{19} n/cm^2 causes a marked volume increase of the concrete [that] can be tracked back to microstructural changes in the crystalline aggregates of the concrete and is with all likelihood responsible for the concrete deterioration." Field et al. (2015) also concluded that indications show that the mechanism of RIVE (i.e., aggregate expansion) "is a first-order mechanism for loss of mechanical properties [of quartz aggregates] under neutron irradiation" and could affect limestone aggregates containing minor amounts of quartz or feldspar embedded in the calcite matrix as well.

In addition, the staff noted that the studies referenced in the SLRA regarding compressive strength loss indicate that operational temperature is a factor that must be considered in the degradation of concrete due to neutron irradiation. The staff noted that with regard to neutron irradiation of concrete, Maruyama's (2017) paper further stated that "there is no data of aggregate expansion for neutron flux in commercial reactors;" and that "the variety and rates of expansion behavior of rock-forming minerals, and their respective roles in thermal healing roles, are key factors to incorporate into soundness assessment;" and "[m]ore extensive data should be obtained for long-term operation of nuclear power plants."

NUREG/CR-7171 is an informational research document that was published by the NRC in 2013. It provides a summary of the effects of neutron and gamma radiation on the mechanical and physical properties of concrete through 2012.

Issue:

The staff notes that there are several bodies of research associated with irradiation of concrete, as noted above. In addition, in research performed for the NRC in NUREG/CR-7171 (2013), an assessment of the results of several past studies related to the degradation of concrete due to irradiation was performed. The staff notes that the reduction of strength in concrete due to radiation is complex and depends on many variables such as type of cement, aggregates, water/cement (w/c) ratio, and temperature to which the concrete is exposed. In order for the staff to assess the reasonableness of the applicant's evaluation approach and assumptions, the applicant should provide a justification for the applicability of the cited study assessing the extent of degradation and reduction in strength for the PSW concrete.

The staff noted that the SLRA did not provide a plant-specific comparison of its concrete constituents (w/c ratio, aggregate type) and its environment (operating temperature) with those of applicable specimens used in the applicant's referenced studies, nor did it appear to present a basis to bound the Turkey Point concrete. Without plant-specific concrete considerations, it is not clear how the applicant reached the conclusion that the Maruyama studies are applicable to Turkey Point. The staff noted that the Figure 54 in Maruyama's (2017) paper (referenced in the SLRA) shows data from a variety of concretes with different cement, aggregates, w/c ratios, and test temperatures that were bounded by a "lower boundary curve." In order to assess the acceptability of the applicant's use of the Maruyama (2017) study in development of the SLRA, the staff needs an explanation and justification for how the following constituents/variables used in assessing Turkey Point's concrete relate to those used in Maruyama's study. In its review of Maruyama's study, the staff noted the following:

- **type of cement and w/c ratio**: Maruyama's (2017) study used a high early-strength ordinary Portland cement with a w/c ratio of 0.5. Turkey Point's uses an ASTM C-150-64 Florida Type II cement with a w/c ratio of 0.59.
- **aggregates**: Maruyama's (2017) study used a combination of fine aggregates (land sand and sandstone) and coarse aggregates (altered tuff crushed and sandstone

gravel) and confirmed findings of previous studies by stating that "the degree to which an aggregate expands depends of its mineral composition, and accordingly, that concretes containing different aggregates incur different levels of damage even following exposure to identical neutron fluence." Turkey Point's concrete fine and coarse aggregates (Miami Oolite (limestone) with some quartz sand) conformed to ASTM C-33-64.

- **temperature**: The staff noted that the test temperature of the Maruyama study specimens was lower (10 to 46 degrees Celsius) than the operating temperature for the concrete at Turkey Point's PSW (approximately 49 degrees Celsius). The staff also noted that in relation to Figure 54 of Maruyama's (2017) paper there is a statement that "it is necessary to include the caveat that no corrections have been made for temperature in this figure." The staff notes that the temperature of the environment can affect the amount of degradation of concrete exposed to neutron radiation due to expansion of the aggregate, in particular for siliceous aggregates (e.g., quartzite).

Considering the variability in the data of Figure 54 in Maruyama (2017) and considering how the varying factors (cement type, aggregate, w/c ratio, and environment temperature) of the concrete at Turkey Point PSW compare to the data in Figure 54, the applicant should clarify or provide a justification of its basis for selecting a value for F_c/F_{c0} of 0.9 (i.e., a 10 percent reduction in concrete compressive strength as a measure of concrete degradation). The applicant should discuss why it chose a value that is less conservative than the "lower boundary curve" value of approximately 0.8 (i.e., a 20 percent reduction in concrete compressive strength) for a neutron fluence of 3.57×10^{19} n/cm^2 as seen in Figure 54 of Maruyama's (2017) paper referenced in the SLRA. In addition, the applicant should provide a discussion on how it considered the results of other studies referenced in the SLRA such as those by Hilsdorf (1978) and Field et al (2015) that show a greater loss of strength due to neutron radiation as shown in the lower bound curve value of approximately .75 (25 percent loss of strength) shown in Figure 2 of Hilsdorf (1978) and 0.5 (50 percent loss of strength) shown in Figure 3 of the SLRA (from Field et al. (2015)) for a neutron fluence of 3.57×10^{19} n/cm^2.

Request:

1. Justify the plant specific evaluation approach and specific assumptions associated with cement, aggregate, w/c ratio, and operating temperature of the concrete at Turkey Point PSW compared to those cited in applicable tests of referenced studies (Figure 54 of Maruyama's 2017 paper, Figure 2 of Hilsdorf (1978) and Figure 3 of Field, et al. (2015)) for determining the reduction in strength and of other mechanical properties of concrete due to neutron fluence at the PSW. If they are not comparable, or if the SLRA credits a bounding case, provide justification that such consideration is unnecessary.

2. Provide a basis for selecting a 10% reduction in strength and mechanical properties of concrete due to neutron fluence at Turkey Point PSW. Clarify whether the selected value is solely based on Figure 54 of Maruyama's (2017) paper. If so, clarify and justify why the more conservative values in Figure 2 of Hilsdorf (1978) and Figure 3 of Field, et al. (2015) are not applicable or are less representative of the concrete at Turkey Point's PSW of the multiple radiation aging effects articulated above.

FPL Response:

The following numbered items respond to the comparable numbered requests above:

1. A plant-specific evaluation of the PTN Primary Shield Wall (PSW) was performed (including comparison to referenced studies) to estimate the reduction in concrete strength and other mechanical properties due to neutron fluence. The same tendency of reduced strength ratios due to neutron fluences are shown in studies by Hilsdorf (1978), Field, et al. (2015) and Maruyama (2017), as well as the related NUREG/CR-7171 (Reference 1).

   The evaluation, summarized in SLRA 3.5.2.2.2.6, Rev. 1 (attachment to Reference 2), relies on material and section properties, Current Licensing Basis (CLB) loading and estimated neutron fluence for the PTN PSW. This PTN-specific information is used with industry standard equations (Reference 3) to evaluate the radiation effects (i.e., swelling strain, compressive & tensile strengths, and elastic modulus) as well as Radiation Induced Volumetric Expansion (RIVE) on the reinforced concrete of the PTN PSW. The specific considerations associated with type of cement and w/c ratio, aggregates, and temperature for this evaluation in relation to the (more recent) Maruyama paper (2017, Reference 4) are discussed as follows:

   <u>**Type of cement**</u>

   As provided in the PTN SLRA, PTN uses Type II cement, which is for general purpose with moderate sulfate resistance. The cement used in the Maruyama paper (2017) is Type III (high-early-strength cement). Per ASTM C-150 (Reference 5), both Type II and III are identified as Ordinary Portland Cement (OPC) having similar chemical and physical requirements. The required compressive strengths for Type II (at 28 days) and for Type III (at 3 days) are 4,000 psi and 3,500 psi, respectively, and are considered the typical range for compressive strength. Although Type II cement is required to gain a targeted compressive strength at 28 days, and Type III cement is required to gain the targeted compressive strength at 3 days as shown in ASTM C-150, Tables 3 and 4 (Reference 5), the concrete composition is similar. Thus, the concrete used by Maruyama is comparable to the concrete used for PTN.

Water cement (w/c) ratio

Water-to-cement (w/c) ratio is related to concrete compressive strength. Per ACI 211.1, Table 6.3.4(a) (Reference 6), typical concrete compressive strength corresponding to the w/c ratio of 0.59 is 3,000 psi, while it is 4,000 psi for a w/c ratio of 0.48. Per the attachment of Reference 2, the estimated w/c ratio of the PTN PSW is between 0.54 and 0.56. The corresponding concrete strength is estimated somewhere between 3,000 psi and 4,000 psi. In the Maruyama paper (2017), a w/c ratio of 0.50 is used. The corresponding compressive strength is also estimated between 3,000 and 4,000 psi, which is bounded by the compressive strength range of 3,000 to 7,500 psi (i.e., achieved at 28 days and 90 days, respectively) for the PTN primary shield wall (PSW) concrete.

Maruyama (2017) selected the high-early-strength cement with the w/c ratio of 0.50 to stabilize hydration as much as possible over a preparation period of one year for the test specimens, with the aim of avoiding hydration-induced strength development appearing in the irradiation tests (see Section 2.2.4 of the paper for details). The w/c ratio used in the Maruyama paper (2017) represents typical concrete compressive strength and are comparable to the w/c ratio used for PTN.

Aggregates

Maruyama (2017) tested different types of aggregates (including limestone, as shown in Table 11 of the paper) and concluded in Section 2.5 of the paper that *"For aggregates, it was confirmed that quartz, with its high covalent bond content, has poor neutron resistance and expanded, while limestone, which contains ionically bonded calcite, did not expand for fast neutron fluences of up to 8.09 x 10^{19} n/cm^2."* In the same section, the paper also stated that *"It was confirmed that the cement paste did not reduce in strength when exposed to fast-neutron (> 0.1 MeV) fluences of up to 8.09 x 10^{19} n/cm^2."*

Per UFSAR, Section 5.1.6.2, the aggregates used for the PTN PSW are ASTM C-33-64 (fine and coarse aggregate, Miami Oolite). Miami Oolite is now referred to as Miami Limestone. In the Maruyama paper (2017), neutron radiation tests were performed on specimens (Con-A and Con-B) which included high contents of tuffaceous sediments and the origin of quartz (silica) as provided in Tables 10 and 11 of the paper. Based on the aggregate expansion results shown in Figure 42 for different types of aggregates, the limestone aggregates (GF) are less sensitive than the others and bounded by quartz aggregates (GA). Therefore, the PTN aggregates (limestone) are less sensitive and bounded by the Maruyama test results performed with specimens including quartz.

Temperature

The temperature range of 10 to 46 degrees Celsius (°C) in the issue above is for the Heating Test (HT) to reproduce the heating and drying experienced by specimens exposed to gamma radiation as provided in Figure 59(a) of the Maruyama paper (2017). For the neutron radiation experiments, the temperature of concrete specimens was measured and provided in Table 23 of the paper where the measured average temperature ranged from 58.9 to 72°C. This temperature range bounds expected operating temperatures of 49 to 65.6°C (i.e., 120 to 150°F) in the reactor cavity and at the RPV supports, respectively. Thus, the measured temperature in the Maruyama paper (2017) bounds the one for PTN.

Based on the above description of concrete properties (i.e., cement type, aggregate, w/c ratio) and operating temperature, the concrete of the PTN PSW is comparable to corresponding information in applicable tests of the referenced studies. Therefore, there is reasonable assurance that the referenced studies are suitable for determining the reduction in strength and of other mechanical properties of concrete due to neutron fluence at the PSW.

2. Based on the neutron fluence limit of 1.0 x 10^{19} n/cm^2 and the calculated PTN neutron fluence of 3.57 x 10^{19} n/cm^2 incident on the primary shield wall at the end of the SPEO, the irradiation effect (i.e., about 10% reduction in concrete strength up to a depth of 2.6 inches) was not selected from Figure 54 of Reference 4, but calculated by using Equations 5-1 to 5-5 in EPRI report number 3002011710. PTN neutron fluence attenuation is calculated for different depths into the concrete, and the reduced strength in concrete is calculated for the corresponding fluence attenuation. The maximum strength reduction in the concrete is calculated at the inner surface of the concrete and it is about 10% as indicated on page 10 of 19 in the attachment to Reference 2. However, due to the RIVE effect (excessive swelling strain), the inner side of the concrete is yielded (cracked). The strength of concrete up to a depth of 3.14 inches is reduced by 100%, which is considered in the PTN PSW evaluation. This is conservative and bounds the concrete strength reduction ratios (due to neutron fluence) presented in Maruyama (2017), Hilsdorf (1978) and Field, et al. (2015).

References:

1. NUREG/CR-7171, “A Review of the Effects of Radiation on Microstructure and Properties of Concretes Used in Nuclear Power Plants”, Nuclear Regulatory Research, Washington D.C., November 2013.

2. FPL Letter L-2018-187 to NRC dated October 5, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Revision to SLRA Section 3.5.2.2.2.6,

Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation (ADAMS Accession No. ML18283A308)

3. EPRI Report No. 3002011710, "Irradiation Damage of the Concrete Biological Shield – Basis for Evaluation of the Concrete Biological Shield Wall for Aging Management", Electric Power Research Institute, Charlotte, NC, May 2018.

4. Maruyama, I., Kontani, O., Takizawa, M., Sawada, S., Ishikawa, S., Yasukouchi, J., Sato, O., Etoh, J., and Igari, T., "Development of Soundness Assessment Procedure for Concrete Members Affected by Neutron and Gamma-Ray Irradiation", Journal of Advanced Concrete Technology, Vol. 15, pp 440-523, 2017 (https://www.jstage.jst.go.jp/article/jact/15/9/15_440/_article)

5. ASTM C-150-07, "Standard Specification for Portland Cement"

6. ACI 211.1-91, "Standard Practice for Selecting Proportions for Normal, Heavyweight, and Mass Concrete", Reapproved 2002.

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

RAI 3.5.2.2.2.6-12

Background:

The SRP-SLR Section 3.5.2.2.2.6 states the following:

> Higher fluence or dose levels may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and/or loss of mechanical properties of concrete from those fluence levels, at or above the operating temperature experienced by the concrete, and the effects are applied to the design calculations. Supporting calculations/analyses, test data, and other technical basis are provided to estimate and evaluate fluence levels and the plant-specific program.

The PTN SLRA Section 3.5.2.2.2.6 states:

> Radiation effects such as neutron fluence and Radiation-Induced-Volumetric-Expansion (RIVE) effects were determined. The existing primary shield wall was evaluated for the CLB loading with the radiation effects by using the same design/analysis approach as the recently updated CLB calculation. Due to the RIVE effect, the excessive compressive stress was calculated and the inner side of the concrete (up to 3.14 inches) is considered as yielded (cracked). The design stresses were then re-calculated for the reduced concrete section under the CLB loading in which the reduced strengths and modulus of the irradiated concrete were also considered. Comparing with the un-irradiated concrete (where the maximum interaction ratio (IR) is calculated as 0.74), the maximum IR for the irradiated concrete (including the cracking discussed above) was calculated as 0.82, which is increased but is still less than 1.0. Therefore, the existing primary shield wall including the radiation effects is qualified for the CLB loading based on the evaluation results.
>
> [...]
>
> Upon NRC approval, the loads on the reactor vessel supports and Primary Shield Wall [PSW] concrete will be significantly reduced. For the [PSW], the implementation of auxiliary line LBB will result in the IR being reduced to 0.41 (tension). The governing load case would be Normal (IR = 0.41 for tension) and Emergency (IR = 0.32 for compression). Considering the IR increasing ratios (i.e., 10.8% for tension and 10.2% for the maximum compression), the maximum IRs are approximated as 0.45 (=0.41 x 1.108) for tension and 0.35 (= 0.32 x 1.102) for the maximum compression.

Issue:

The SLRA does not provide a clear description of the CLB design basis with load combinations, governing load case(s), and their respective maximum IRs and their locations for all stress conditions (tension, compression, and shear stresses) of the Turkey Point PSW concrete structure, or a justified bounding case. The staff needs this information to assess margins in available capacities considering the effects of concrete degradation due to irradiation (i.e., cumulative effects of neutron fluence, gamma dose, and RIVE effects) for the PSW concrete structure during the SPEO.

Request:

Taking into consideration the loss of strength and change in mechanical properties of irradiated concrete due to cumulative effects of neutron fluence, gamma dose, and RIVE effects, describe all affected design basis load combinations, identify the governing load case(s), provide the respective maximum horizontal, vertical loads, and bending moments on the PSW surface and at the point of termination of concrete loss of strength. For the governing load case(s) provide the resulting maximum IRs and their location under all stress conditions (tension, compression, shear) for the Turkey Point PSW concrete structure. Alternatively, provide a justified bounding case.

FPL Response:

Per FSAR, Section 5.1.8.2(c), the PSW has been designed to withstand the proper load combinations of dead, live, thermal, seismic, and accidental loads. For the CLB design loads that include the new reactor vessel head weights and new LOCA, its functionality has been examined by using the same analysis approach and considerations used in the original PSW design. The considered load combinations are as follows:

- Normal operating: D + L + T
- Emergency (with seismic): D + L + T + E
- Faulted (with original LOCA): D + L + T + original LOCA
- Faulted (with new LOCA): D + L + T + new LOCA

where D = dead load, L = live load, T = thermal load due to radiation, E = seismic load, and LOCA = Loss-Of-Coolant-Accident (refer to the design basis calculation for loading of the PSW for more detail).

The structural responses of the existing PSW have been examined in its radial, tangential, and longitudinal directions under the above load combinations. Based on the examination, it was observed that the stress and displacement in the radial direction are insignificant as shown in the design basis calculation for loading of the PSW (for instance, the radial displacement for the thermal gradient is 0.027 inches). The maximum IRs were calculated for the normal, emergency (seismic) and faulted load cases (with the original and new LOCA) and summarized in the calculation. Based on the calculated IRs, the faulted load case (with new LOCA) is governing where the

maximum IR for the longitudinal reinforcement is 2 times larger than the other load cases (normal, seismic and original LOCA). The design basis calculation for loading of the PSW contains a summary of the IRs for different load cases.

The radiation effect on the PSW is representatively examined for this governing load case in the evaluation report of the effect of radiation on the PSW and supports. This report is available on the ePortal. In addition, the existing PSW was evaluated for the radiation effects (i.e., neutron, gamma, and RIVE) by EPRI for the governing load case which is faulted loading with new LOCA. Due to the RIVE effect, the excessive compressive stress is calculated, and the inner side of the concrete (up to 3.14 inches) is considered as yielded (cracked). The corresponding axial force and bending moment for the reduced concrete section under the governing external loads (i.e., dead, live, and new LOCA loads) are calculated as 215 kip and 558.4 kip-ft, respectively. The total stresses including thermal stresses are then re-distributed to the reduced concrete section under the governing CLB loading considering the radiation effects (reduced strengths and modulus of the irradiated concrete). The maximum stresses in the reinforcement and in concrete are calculated and compared with the un-irradiated concrete in the evaluation report.

For horizontal seismic loads, its shear capacity was examined at the base by using the concrete shear capacity alone. The shear capacity is 8,333.6 kip, which is much greater than the 3,119 kip seismic base shear demand. Thus, the existing PSW has sufficient capacity for the horizontal seismic loads.

The maximum IR for the un-irradiated concrete is 0.74 for tension, while the maximum IR for the irradiated concrete is 0.82. The maximum IR has increased by 10.8% (= [0.82-0.74] / 0.74) but is still less than 1.0. Therefore, the existing PSW including the radiation effects is qualified for the CLB loading based on the evaluation results. The maximum IR for tension is calculated at the outer side longitudinal reinforcement, while the maximum IR for compression is calculated at the inner side of the concrete, which is between 10 and 15 inches from the inner surface.

It should be noted that the Leak-Before-Break (LBB) analysis of reactor coolant system auxiliary lines has been submitted as part of the SLRA (Section 4.7.4 and Enclosure 4, Attachment 12). Upon NRC approval of the LBB analysis, the loads on the reactor vessel supports and PSW concrete may be significantly reduced and more design margin is expected.

References:

1. FPL Letter L-2018-187 to NRC dated October 5, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Revision to SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation (ADAMS Accession No. ML18283A308).

Associated SLRA Revisions:

SLRA Section 3.5.2.2.2.6, Rev. 1, is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions.

Revise SLRA Section 3.5.2.2.2.6, Rev. 1, Page 14 of 19 (2nd paragraph) of Reference 1 as follows:

Due to the RIVE effect, the excessive compressive stress was calculated and the inner side of the concrete (up to 3.14 inches) is considered as yielded (cracked). The design stresses were then re-calculated for the reduced concrete section under the **governing** CLB loading **case (i.e., faulted with new LOCA)** in which the reduced strengths and modulus of the irradiated concrete were also considered.

Associated Enclosures:

None

NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

RAI 3.5.2.2.2.6-13

Background:

The SRP-SLR Section 3.5.2.2.2.6 states the following (in part):

> Higher fluence or dose levels may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and/or loss of mechanical properties of concrete from those fluence levels, at or above the operating temperature experienced by the concrete, and the effects are applied to the design calculations. Supporting calculations/analyses, test data, and other technical basis are provided to estimate and evaluate fluence levels and the plant-specific program.

The SLRA states in part the following:

> The [RPV] support structure for each PTN [Turkey Point] Unit consists of six (6) individual supports, one of which is placed under each of the three hot leg and three cold leg Reactor Coolant System pipe nozzles at elevation (EL) 25'-7 ½". A majority of each [RPV] support is embedded in the primary shield wall. [...] The [RPV] support structure includes vertical columns, cantilever beams, horizontal (cross) beams and roller assembly. The columns and portion of the cantilever beams are located inside the primary shield wall, with the centerline of the cantilever beams at a height approximately equal to the top of the active fuel, and the inboard edge of the innermost column ~ 5 inches from the inside surface of the primary shield wall.

The SLRA provides an evaluation of the RPV steel supports for the aging effect of reduction in fracture toughness due to irradiation embrittlement.

Issue:

The staff noted that the RPV steel support assemblies are partially embedded into the concrete of the PSW. As stated in the SLRA, this concrete is expected to have a loss of strength and change in mechanical properties due to the aging effects of radiation. The SLRA provides an evaluation of the RPV structural steel support assemblies for the aging effect of reduction in fracture toughness due to irradiation embrittlement. The staff noted, however, that the SLRA does not include a consideration of how the degradation of the PSW concrete due to irradiation would affect the CLB structural performance/integrity and intended function of the RPV supports – particularly their embedded portion into the concrete (e.g., degree of fixity of steel beams) – and the state of the local concrete (e.g., local crushing of concrete).

The staff notes that a loss of strength and change in mechanical properties of concrete in which the RPV steel support structure is embedded would result in partial fixity of the steel beam supports into the PSW, thus potentially changing behavior of the composite concrete steel RPV support system which could affect the intended function of RPV support, including limits of its displacement.

The staff needs additional information to assess, with regard to the CLB design loads and intended function, the margin in structural capacity available under critical stress conditions for the RPV support structure and the ability of the steel support structure to prevent excessive movement (per CLB design) of the RPV during the SPEO. The staff needs this information regarding assessments of the degree of fixity and load transfer of the RPV steel supports into the degraded PSW concrete in order to evaluate the impact such degradation could potentially have on the CLB intended functions of the reactor vessel supports during SPEO. Specifically, the staff needs information regarding (1) the governing CLB (or credited LBB) design basis load combination(s), consideration of possible redistribution of maximum stresses (e.g., tension, compression, and shear) or change in maximum IRs (e.g., tension, compression, and shear) and their location, consideration of potential pull-out or slippage of the concrete/steel support system, and any potential settlement of the RPV supports due to the expected degradation of the surrounding concrete caused by the combined effect of neutron fluence, gamma dose, and RIVE; or (2) a justified bounding case.

Request:

1. Discuss whether and how the loss of strength and change in mechanical properties of concrete due to irradiation has a local effect on the degree of fixity and load transfer of the RPV steel supports into the degraded PSW concrete, or provide justification for not needing to consider these local effects.

2. Taking into consideration the values provided by the SLRA for loss of strength and change in mechanical properties of concrete due to irradiation discussed in the SLRA, and variation in the degree of fixity of the steel beams, if any, provide an analysis that includes the governing design basis load combinations (identified in RAI 3.5.2.2.2.6-12) with their respective maximum horizontal, vertical loads, and bending moments under all stress conditions (e.g., tension, compression, shear) including IRs, for the supports, and any potential settlement for the RPV steel support structure; or provide a justified bounding case.

FPL Response:

The following numbered responses correspond to the numbered requests above:

1. Due to the radiation effects (i.e., loss of strength, change in mechanical properties, and swelling strain), the inner side of the concrete (up to 3.14 inches) in the PTN Primary Shield Wall (PSW) is calculated to be yielded (cracked). Considering the overall wall thickness of 7 ft, the crack is limited and localized to the inner surface of

the PSW. The Reactor Pressure Vessel (RPV) steel supports are integrated into the concrete over the full cross section of the wall. The majority portions of the horizontal cantilever beams and the vertical columns which are the main structural members transferring RPV support loading are embedded into the concrete.

Per the plant drawings, approximately 4.5 ft out of 6 ft of the horizontal cantilever beams are embedded into the concrete. Based on the span-depth ratio, compact section, and 1" thick stiffener plates, the horizontal cantilever beam 14WF342 is considered as a deep beam, which is governed by shear as shown in the Westinghouse RPV support calculation. The effective length of the cantilevered portion of the beam may be increased by 3.14 inches due to the crack. However, the span-depth ratio is less than two. Thus, the horizontal cantilever beam is still considered a deep beam. The governing structural response (shear demand of the beam) will not change due to this localized cracking depth. Per the PSW liner plate drawing, the inner surface of the PSW is covered by ¼" thick liner plates with angles and channels that are welded to the liner plates from the top to the bottom of the PSW. Considering the resistance from the liner plates and the remaining concrete in compression, the effect to the fixity will be reduced. Therefore, it is considered that the horizontal cantilever beam, its fixity, and load transfer to the concrete are not significantly affected by this local effect in the concrete.

The vertical columns of the RPV steel supports are fully embedded into the concrete with forty-eight (48) 7/8" dia. by 3 ½" long headed studs for each column. Among these studs, the outer row of studs is located at about 5 inches away from the inner surface of the concrete wall. Therefore, the interface between the RPV steel support and concrete will not be affected by the cracking of the concrete (up to of 3.14 inches) due to the radiation effects. Even if the outer row of studs is affected, it will be limited and localized in a small area. A significant number of studs is still remaining effective.

2. The Westinghouse RPV support calculation provides the related analysis and evaluation details on the existing RPV steel supports for the CLB design loads (i.e., normal operating, seismic, and old and new LOCA loads). Among the considered CLB loading cases, the faulted load case with new LOCA is determined as the governing load case. The governing structural behavior of the cantilever beam is shear (not bending). The maximum IR is calculated for the shear in the faulted load case (with new LOCA), and it is four times larger than the other IRs for the upset (seismic) load case. The increased span length due to the cracking depth will not provide any appreciable impact on the structural demand.

   With respect to the loss of strength and degree of fixity in the concrete up to the cracking depth, the corresponding displacement at the end of cantilever beam is calculated by using the minimum vertical stiffness of the RPV support provided in the Westinghouse RPV support stiffness calculation. From Figure 5-9 of the calculation, the average length of the cantilever beam is calculated, and the span length of the

cantilever beam is increased by about 21% considering the cracking depth of 3.14 inches. Vertical stiffness of the cantilever beam is inversely proportional to (span length)^3, so the displacement corresponding to the maximum RPV support reaction for the faulted load case (with new LOCA) can be calculated by using the vertical stiffness considering the radiation effect in the concrete. The corresponding maximum displacement is calculated to be less than 0.1 inches. Therefore, the local effect (including associated settlement) is considered as miscellaneous and not needed to be considered with respect to the degree of fixity, related displacement, and load transfer. The Westinghouse calculations and the PSW evaluation report are available on the ePortal.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter No. ML18341A004 and ML18341A005 Dated January 15, 2019

4. ASME Section XI, Subsection IWF, GALL AMP XI.S3

RAI B.2.3.32-1

Background:

SLRA Section B.2.3.32 states that the program takes exception to the GALL-SLR Report AMP XI.S3, which recommends using bolting material which has an actual measured yield strength limited to less than 150 ksi, as a preventive measure against cracking due to SCC for structural applications. The SLRA states that HS ASTM A325 and ASTM A490 bolting is used for bolting repair and maintenance activities (the staff clarified with the applicant during its in-office audit that there is no plan to discontinue use). The SLRA includes an enhancement to the current AMP to include volumetric examinations to detect cracking due to SCC during the SPEO.

Issue:

SLRA Section B.2.3.32 states that the current program limits the use of lubricants containing sulfur as a preventive measure to reduce the chance that cracking due to SCC can occur in high-yield bolting and that for the SPEO the AMP will be enhanced to explicitly state that lubricants cannot contain molybdenum disulfide (MoS_2). However, the staff noted during its in-office audit that the applicable procedures currently do allow the use of these lubricants and do not appear to prohibit or limit their use. SRP-SLR Section A.1.2.3.4 states that "detection of aging effects should occur before there is a loss of the SC-intended function(s)." The staff noted that the current visual inspections cannot detect cracking due to SCC. Since preventive measures are not in place before the SPEO (i.e., lubricants containing sulfur may have been used), it is possible that the aging effect may be present (or become present due to continued use of HS bolts as replacements, and lubricants containing MoS_2) and may remain undetected until volumetric examinations are performed during the SPEO. Therefore, for the period of time between the start of the SPEO and when the volumetric examinations are performed is not clear how the aging effect of cracking due to SCC will be detected prior to a loss of intended function.

As indicated in the exception above, the applicant is expected to continue its use of HS bolts susceptible to SCC and thus has the potential to increase the population of installed HS bolts (i.e., install additional HS bolts as replacement bolting) susceptible to SCC on site. It is not clear from the SLRA how the HS bolting sample subject to volumetric examination will be established, and how the program will assess the sample size and scope to ensure that it continues to represent the entire population of HS bolts, especially considering those exposed to MoS_2.

Request:

1. Since volumetric examinations are planned for some time into the SPEO, provide information on whether and how the aging effect of cracking due to SCC will be detected for the population of HS bolts such that this aging effect can be managed from the start of the SPEO.
2. Discuss how the program will assess the adequacy of the HS bolting sample inspected for cracking due to SCC when additional HS bolts are installed.

FPL Response:

1. To manage the aging effect of cracking due to SCC in bolting within the boundaries of IWF-1300 with a nominal diameter of greater than one inch and a maximum yield strength of 150 ksi or greater, inspections will begin during the inspection interval prior to the start of the SPEO. The PTN SLRA Section B.2.3.32 is revised to clarify the inspection schedule.
2. The PTN high-strength bolting sample will consist of 20% of high strength bolting within the boundaries of IWF-1300, up to a maximum of 25 bolts per unit. The maximum sample population is consistent with other SLR inspection programs at PTN. Consistent with the response to RAI B.2.3.32-2, MoS_2 and other lubricants which contain sulfur will be explicitly prohibited at PTN prior to the start of the SPEO. This ensures that any high strength bolting installed as replacement bolting at PTN would not be exposed to this potential contributor to SCC and would not be representative of the condition of the original population. As such, the original population would represent the most susceptible locations and adequacy of the sample size will be assured. The PTN SLRA is revised to clarify the sample population definition as well as inspection techniques.

References:

None

Associated SLRA Revisions:

Refer to the FPL response to NRC RAI No. B.2.3.32-2 (Attachment 9).

Associated Enclosures:

None

NRC RAI Letter No. ML18341A004 and ML18341A005 Dated January 15, 2019

RAI B.2.3.32-2

Background:

SLRA Section B.2.3.32 states that the ASME Section XI, Subsection IWF program is consistent, with an exception and enhancements, with GALL-SLR Report AMP XI.S3.

Issue:

The staff needs additional information to clarify whether the program is consistent with the GALL-SLR Report recommendations:

1. The GALL-SLR Report "preventive actions" program element recommends that molybdenum disulfide and other lubricants containing sulfur should not be used. SLRA Section B.2.3.32 states that the program will be enhanced to explicitly state that lubricants cannot contain molybdenum disulfide, but this is not listed as an enhancement to the "preventive actions" program element, and also does not appear in Appendix A in the Updated Final Safety Analysis Report (UFSAR) supplement. In addition, the SLRA does not specify plans to exclude other lubricants containing sulfur.
2. The GALL-SLR Report "monitoring and trending" program element recommends that if a component support does not exceed the acceptance standards of IWF-3400 but is repaired to as-new condition, the sample is increased or modified to include another support that is representative of the remaining population of supports that were not repaired. SLRA Section B.2.3.32 does not address this recommendation and in its review of the current program, the staff did not find that this was a current practice. Therefore, considering the application states the program is consistent with the GALL-SLR Report it is unclear whether the program takes exception to the GALL-SLR Report recommendation, or if the program needs to be enhanced.

Request:

State whether the two areas discuss in the "Issue" section are considered Enhancements to the ASME Section XI, Subsection IWF AMP. If so, modify the SLRA (including the UFSAR supplement) to reflect the applicable enhancements in SLRA Section B.2.3.32 and Section 17.2.2.32, as necessary. In addition, clarify in the enhancement whether the program includes the prohibition of other lubricants containing sulfur (besides MoS_2) or provide justification to allow such lubricants.

FPL Response:

The two areas discussed in the "Issue" section are considered Enhancements to the PTN ASME Section XI, Subsection IWF AMP.

1. To help prevent the aging effect of cracking due to SCC in bolting, molybdenum disulfide and other lubricants containing sulfur will be explicitly prohibited at PTN.

The PTN SLRA is revised to clarify the enhancement.

2. The PTN ASME Section XI, Subsection IWF AMP will be enhanced to address modifying the sample population when component supports are repaired to as-new condition. The PTN SLRA is revised to reflect this enhancement.

References:

None

Associated SLRA Revisions:

SLRA Section 17.2.2.32, Table 17-3, and B.2.3.32 are amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions. Note these changes also include the changes associated with the response to RAI B.2.3.32-1 (Attachment 8).

Revise SLRA Appendix A Section 17.2.2.32 on pages A-36 and A-37 as follows:

The PTN ASME Section XI, Subsection IWF AMP is an existing condition monitoring program that consists of periodic visual examination of ASME Section XI Class 1, 2, and 3 piping and component support members for signs of degradation such as loss of material, **cracking,** and loss of mechanical function. Bolting for Class 1, 2, and 3, and MC piping and component supports is also included and inspected for loss of material and for loss of preload. Associated sliding surfaces, and vibration isolation elements are also inspected.

The PTN ASME Section XI, Subsection IWF AMP provides inspection and acceptance criteria and meets the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, 2007 edition with addenda through 2008, and 10 CFR 50.55a(b)(2) for Class 1, 2, 3 piping and components and their associated supports. The primary inspection method employed is visual examination. Non-destructive examination (NDE) indications are evaluated against the acceptance standards of ASME Code Section XI.

Examinations that reveal indications are evaluated. Examinations that reveal flaws or relevant conditions that exceed the referenced acceptance standard, are expanded to include additional examinations during the current outage. The scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). Tactile inspections of elastomeric vibration isolation elements to detect hardening if the vibration isolation function is suspect is also included.

The requirements of ASME Section XI, Subsection IWF are supplemented to include **the following:**

- **Identification of high-strength (maximum yield strength greater than or equal to 150 ksi) bolting greater than one inch nominal diameter within the boundaries of IWF-1300 and subsequent** volumetric examination **of 20%** of the high-strength bolting **at least once per interval** for cracking; **or replacement and inspection of the removed high-strength bolting (which may not be reused per plant procedure) using a technique capable of detecting cracking.**

  AND

- ~~a~~**A** one-time inspection within 5 years prior to the SPEO of an additional 5 percent of piping supports from the remaining IWF population that are considered most susceptible to age-related degradation.

Revise SLRA Appendix A Section 17.4 Table 17-3 on page A-104 as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
36	ASME Section XI, Subsection IWF (17.2.2.32)	XI.S3	Continue the existing PTN ASME Section XI, Subsection IWF AMP, including enhancement to: a) Store high strength bolts in accordance with Section 2 of Research Council for Structural Connections publication "Specification for Structural Joints Using High-Strength Bolts". b) Perform a one-time inspection, within 5 years prior to entering the SPEO, of an additional 5 percent of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports, which are not exempt from examination, that is focused on supports selected from the remaining IWF population that are considered most susceptible to age-related degradation. c) Include physical (tactile) examination of elastomeric vibration isolation	At 5 years prior to the SPEO, start one-time inspections~~.~~ Complete pre-SPEO inspection **<u>and enhancements</u>** no later than 6 months or the last refueling outage prior to SPEO. Corresponding dates are as follows: PTN3: 7/19/2027 - 1/19/2032 PTN4: 4/10/2028 - 10/10/2032

			elements to detect hardening if the vibration isolation function is suspect due to aging. **d) Identify the population of ASME Class 1, 2, 3 and MC high-strength structural bolting greater than 1 inch in nominal diameter within the boundaries of IWF-1300.** ~~d)~~**e)** Perform volumetric examination, comparable to Table IWB-2500-1, Examination Category B-G-1, at least once per interval for **20% of the identified high strength bolting within the boundaries of IWF-1300 up to a maximum of 25 bolts per unit** ~~a sample of high-strength bolting selected to provide reasonable assurance that SCC is not occurring for the entire population of high-strength bolts~~. Alternatively, **replacement and inspection of the removed bolting using a technique capable of detecting cracking may be performed** ~~a site-specific justification for waiving~~ **in place of the** volumetric examination ~~may be documented~~.	

			f) Revise procedures to note that lubricants cannot contain Molybdenum Disulfide, or other lubricants containing sulfur, in order to inhibit SCC. **g) Increase or modify the component support inspection population when a component is repaired to as-new condition by including another support that is representative of the remaining population of supports that were not repaired.**	

Revise SLRA Appendix B Section B.2.3.32 on page B-241 as follows:

B.2.3.32 ASME Section XI, Subsection IWF

Program Description

The ASME Section XI, Subsection IWF AMP is an existing condition monitoring program that consists of periodic visual examination of ASME Code Section XI Class 1, 2, and 3 supports for ASME piping and components for signs of degradation such as corrosion; cracking, deformation; misalignment of supports; missing, detached, or loosened support items; loss of integrity of welds; improper clearances of guides and stops; and improper hot or cold settings of spring supports and constant load supports. Bolting for Class 1, 2, and 3, piping and component supports is also included and inspected for corrosion, loss of integrity of bolted connections due to self-loosening, and material conditions that can affect structural integrity. Associated sliding surfaces, and vibration isolation elements are also inspected for loss of material or mechanical or isolation function.

The ASME Section XI, Subsection IWF AMP provides inspection and acceptance criteria and meets the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, 2007 edition with addenda through 2008, and 10 CFR 50.55a(b)(2) for Class 1, 2, 3 piping and components and their associated supports. The primary inspection method employed is visual examination. NDE indications are evaluated against the acceptance standards of ASME Code Section XI. Examinations that reveal indications are evaluated. Examinations that reveal flaws or relevant conditions that exceed the referenced acceptance standard, are expanded to include additional examinations during the current outage. The scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Code Class. The largest sample size is specified for the most critical supports (ASME Code Class 1). The sample size decreases for the less critical supports (ASME Code Class 2 and 3). Tactile inspections of elastomeric vibration isolation elements to detect hardening if the vibration isolation function is suspect is also included.

This AMP emphasizes proper selection of bolting material, lubricants, and installation torque or tension to prevent or minimize loss of bolting preload of structural bolting and cracking of high-strength bolting. As noted below in the enhancement discussion, the AMP also includes the preventive actions for storage requirements of high-strength bolts. The requirements of ASME Code Section XI, Subsection IWF are supplemented to include volumetric examination of high-strength bolting **<u>within the boundaries of IWF-1300</u>** for cracking, **<u>or replacement and inspection of the removed bolting using a technique capable of detecting cracking as an alternative to the volumetric examination</u>**. This AMP will also include a one-time inspection within 5 years prior to the SPEO of an additional 5 percent of piping supports from the remaining IWF population that are considered most susceptible to age-related degradation.

NUREG-2191 Consistency

The PTN ASME Section XI, Subsection IWF AMP, with exception and enhancements, will be consistent with the 10 elements of NUREG-2191, Section XI.S3, "ASME Section XI, Subsection IWF."

Exceptions to NUREG-2191

NUREG-2191 recommends using bolting material for structural applications that have an actual measured yield strength limited to less than 1,034 megapascals (MPa) (150 kilo-pounds per square inch) (NUREG-1339), as a preventive measure that can reduce the potential for SCC. High strength bolts, including ASTM A-325 and ASTM A-490 are used in structural applications at PTN and bolting replacement and maintenance activities. Site documentation indicates bolts conforming to ASTM A-490, "Heat Treated Structural Bolts, 150 ksi Minimum Tensile Strength" are installed on structural steel components. PTN performs visual inspection of high-strength bolting in accordance with ASTM A-325 and A-490. ~~PTN also limits the use of sulfur containing lubricants as a preventive measure to reduce SCC of high-yield bolting. Additionally, ASME Section XI, Subsection IWF AMP will be enhanced to explicitly state that lubricants cannot contain Molybdenum Disulfide.~~ The other preventive actions (use of appropriate lubricants, appropriate installation torque, and appropriate storage) in NUREG-2191 XI.S3 AMP that can reduce the potential for cracking are met by the PTN ASME Section XI, Subsection IWF AMP. The exception is acceptable because PTN meets all other program element requirements for structural bolting. Furthermore, the PTN AMP includes an enhancement for volumetric examination of high-strength bolts in order to provide reasonable assurance that SCC is not occurring.

Enhancements

The PTN ASME Section XI, Subsection IWF AMP will be enhanced for alignment with NUREG-2191 as discussed below. This AMP with the following enhancements is to be implemented and a one-time inspection of an additional 5 percent of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports is to be conducted within 5 years prior to the SPEO. Inspections that are to be completed prior to the SPEO are completed 6 months prior to the SPEO or no later than the last RFO prior to the SPEO.

Element Affected	Enhancement
2. Preventive Actions	Store high strength bolts in accordance with Section 2 of Research Council for Structural Connections publication "Specification for Structural Joints Using High-Strength Bolts". **Revise procedures to note that lubricants cannot contain Molybdenum Disulfide, or other lubricants containing sulfur, in order to inhibit SCC.**
3. Parameters Monitored or Inspected	Include **identification** ~~volumetric examination~~ of high strength bolting in sizes greater than 1 inch nominal diameter (including ASTM A490 bolts and equivalent bolts) **within the boundaries of IWF-1300 and subsequent volumetric examination or replacement and inspection of the bolting using a technique capable of detecting** ~~for~~ evidence of SCC.
4. Detection of Aging Effects	Include a one-time inspection, within 5 years of entering the SPEO, of an additional 5 percent of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports, which are not exempt from examination, that is focused on supports selected from the remaining IWF population that are considered most susceptible to age related degradation. Include tactile inspection (feeling, prodding) of elastomeric vibration isolation elements to detect hardening if the vibration isolation function is suspect. Include volumetric examination, comparable to **the techniques of** Table IWB-2500-1, Examination Category B-G-1, at least once per interval for **20% of the identified high strength bolting within the boundaries of IWF-1300 up to a maximum of 25 bolts per unit.** ~~a sample of high-strength bolting selected to provide reasonable assurance that SCC is not occurring for the entire population of high-strength bolts~~. Alternatively, **replacement and inspection of the removed bolting using a technique capable of detecting cracking may be performed** ~~a site-specific justification for waiving~~ **in place of the** volumetric examination ~~may be documented~~.

5. Monitoring and Trending	**Increase or modify the component support inspection population when a component is repaired to as-new condition such that the inspection population remains representative of the remaining population of supports that were not repaired.**

Associated Enclosures:

None

NRC RAI Letter No. ML18341A004 and ML18341A005 Dated January 15, 2019

5. Structures Monitoring Program, GALL AMP XI.S6

RAI B.2.3.35-3a

Background:

The "detection of aging effects" program element of GALL-SLR Report AMP XI.S6, "Structures Monitoring," recommends that a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the SPEO if the plant has an aggressive groundwater/soil environment. The GALL-SLR Report provides examples of what actions may be included as part of the plant-specific AMP. The SRP-SLR Appendix A provides the staff positions and guidance for implementing a plant-specific AMP.

The applicant response to RAI B.2.3.35-3, provides enhancements to the different program elements and establishes plant-specific actions within the Structures Monitoring Program, to ensure that the aging effects associated with structures exposed to aggressive groundwater/soil are adequately managed during the SPEO, as recommended by the GALL-SLR Report. In its response, the applicant proposes to perform a baseline inspection and evaluation of the results prior to the SPEO, and plans to use the inspection results to establish the subsequent periodic inspections (either focused or opportunistic) and evaluation requirements to adequately monitor the condition of inaccessible concrete exposed to aggressive groundwater/soil. The baseline inspection consists of the excavation of two inaccessible concrete locations, one in the main plant and another in a structure near the coastline, to allow a visual inspection and chemical analysis (pH and chloride concentration test) of the previously inaccessible concrete structural elements.

Issue:

Based on the response provided for RAI B.2.3.35-3, the staff identified the following issues requiring additional clarification:

A. The response and enhancements do not provide an adequate technical justification and establish the acceptance criteria that will be used, based on the baseline inspection results, to select the type of subsequent periodic inspections that will be performed during the SPEO (either focused, opportunistic, or both). The staff notes that when structures are exposed to an aggressive groundwater/soil environment, the use of opportunistic inspections may not be sufficient to adequately manage the structures. The staff also notes that the GALL-SLR Report recommends the use of opportunistic inspections of concrete when exposed via excavation for any reason, when plants are exposed to a nonaggressive groundwater/soil environment.

B. The AMP does not provide the criteria that will be used for establishing the sample size (quantity) and location(s) of structures that will be monitored during the SPEO using periodic inspections. The staff notes that SRP-SLR Section A.1.2.3.4 identifies the criteria for the selection of inspection population and sample size, and

the different aspects that need to be considered (i.e. material, environment, the specific aging effect, location and number of units) to adequately determine a representative sample of the population; and provides a provision for expanding the sample size when degradation is detected.

C. It is not clear if the locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.

Request:

1. Describe the criteria that will be used to determine the type of periodic inspection that will be performed (focused, opportunistic, or both), and to identify any additional actions following the baseline inspection and evaluation. Include any technical justification necessary to support the selected approach/criteria, and to demonstrate that the effects of aging on structures and components subject to an AMR will be adequately managed during the SPEO.
2. Describe the criteria that will be used to determine the sample size (quantity) and locations of inaccessible areas of structures that will be monitored following the baseline inspection; and state is there is a criteria for expanding the sample size when degradation is detected during the inspections (ref. SRP-SLR Section A.1.2.3.4).
3. Clarify if the locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.

FPL Response:

This response supplements FPL's December 14, 2018 revised response (Attachment 5 of Reference 6; FPL Letter L-2018-223) per discussion during the December 20, 2018 NRC public meeting with FPL (Reference 7) which supersedes FPL's November 2, 2018 RAI response (Attachment 16 of Reference 1; FPL Letter -2018-193) per discussion during the November 15, 2018 NRC public meeting with FPL (Reference 2).

The below response also considers the L-2018-191 Attachment 4 and 7 (Reference 5) and L-2018-223 Attachment 4 (Reference 6) responses.

This information addresses the following clarifications regarding the site-specific enhancement to the Structures Monitoring AMP:

1. The type of periodic inspection that will be performed (at intervals not to exceed 5 years) will be a focused inspection. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval. Following the baseline inspection and baseline evaluation or periodic inspection and periodic evaluation, any degradation relative to acceptance criteria specified in ACI 349.3 or chemical analysis results that indicate the potential for significant corrosion of reinforcing steel

will be entered into the corrective action program. Additional actions following the baseline inspection and evaluation are based on the corrective action program. The site-specific enhancement description in the SLRA is revised to clarify the periodic inspection and additional actions.

2. Following the baseline inspection and baseline evaluation or periodic inspection and periodic evaluation, any observed degradation will be entered into the corrective action program. The inspection, evaluation, and the corrective action program results (as applicable to degradation) will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years) for the SPEO. The minimum inspection interval of every 5 years will be reduced if the observed degradation could adversely affect structural function prior to the next scheduled inspection. Based on the corrective action results, the sample size may be expanded to include additional locations based on the aging effect (cracking, loss of material (spalling, scaling), increase in porosity and permeability, loss of strength, loss of bond), location (close to the coastline/intake or main plant area), existing technical information, structure design, material of construction (concrete), environment, operating conditions, and OE. For example, if degradation outside of the acceptance criteria is identified at the coastline/intake location inspected during the inspection interval, an additional location at the coastline should be inspected during the following inspection interval. The site-specific enhancement description in the SLRA is revised to clarify the criteria for expanding the sample size criteria when degradation is detected during the inspections.
3. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection. The site-specific enhancement description in the SLRA is revised for clarity.

For clarity, the existing enhancement to the detection of aging effects element will be replaced with a site-specific enhancement to the pertinent elements for management of inaccessible concrete (scope of program, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria). The site-specific enhancement includes the following:

1. A baseline inspection of inaccessible concrete will be conducted prior to the SPEO.
   a) The baseline inspection locations will consider site-specific OE. OE considered will include known degradation due to chlorides in ambient air and the potential for further degradation due to the aggressive groundwater as well as whether leaching and carbonation is occurring in the water-flowing environment.
   b) The baseline inspection will include excavation, visual inspection, and physical inspection of the inaccessible concrete through pH analysis and a chloride concentration test at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection of these two locations is a representative sample since the baseline sample is 20 percent of the population of structures most likely to experience degradation associated with

groundwater (Unit 3 and 4 intake structure, discharge structure, containment structure, and auxiliary building).

2. A baseline evaluation will be performed prior to the SPEO.
   a) The baseline evaluation will consider the baseline inspection results to determine the additional actions (if any) that are warranted. Any observed degradation will be entered into the corrective action program. The baseline inspection results are evaluated based on acceptance criteria provided in ACI 349.3R and will also consider the correlation between the chloride ion concentration necessary to induce corrosion and alkalinity level of the concrete (Reference 3). The highly alkaline environment of concrete protects the steel reinforcement from corrosion (Reference 4). Additional actions will be based on the baseline inspection results and corrective action program and may include: enhanced inspection techniques and/or frequency, destructive testing, and focused inspections of representative accessible concrete (leading indicator) or below grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil (or to leaching and carbonation in water-flowing if determined to impact intended function).
   b) The baseline inspection and evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years) for the SPEO. The minimum inspection interval of every 5 years will be reduced if the observed degradation could adversely affect structural function prior to the next scheduled inspection. The subsequent inspection sample size will include the two baseline inspection locations and may be expanded based on any corrective action program results (as applicable to degradation) to include additional locations. Additional locations may be based on the aging effect (cracking, loss of material (spalling, scaling), increase in porosity and permeability, loss of strength, loss of bond), location (close to the coastline/intake or main plant area), existing technical information, structure design, material of construction (concrete), environment, operating conditions, and OE. For example, if degradation outside of the acceptance criteria is identified at the coastline/intake location, an additional location at the coastline should be inspected.
3. Periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years) will be performed.
   a) Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.
   b) The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.

4. Periodic evaluation updates will be performed (not to exceed 5 years).
   a) Updates will be based on OE and focused periodic inspection results (and/or opportunistic inspection results if applicable) during the interval.
   b) The periodic evaluation results will update subsequent inspection requirements and inspection intervals (not to exceed 5 years) for the SPEO as required.

Accessible areas of in-scope concrete structures are inspected through the Structures Monitoring AMP for aging effects related to aggressive chemical attack such as loss of material (spalling, scaling), cracking, and other irregularities (increase in porosity and permeability. Issues related to accessible areas of concrete are entered into the corrective action program. Pertinent SLRA sections are revised to reflect the Structures Monitoring AMP site-specific enhancement.

References:

1. FPL Letter L-2018-193 to NRC Dated November 2, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review Requests for Additional Information (RAI) Set 6 Responses (ADAMS Accession Number ML18311A299)

2. NRC Public Meeting Agenda Dated November 5, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18315A004)

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

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

5. FPL Letter L-2018-191 to NRC Dated November 28, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review Requests for Additional Information (RAI) Set 7 Responses (ADAMS Accession No. ML18334A182)

6. FPL Letter L-2018-223 to NRC Dated December 14, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review – November 15, 2018 Public Meeting Action Item Responses (ADAMS Accession Number ML18352A885)

7. NRC Public Meeting Agenda Dated December 20, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18324A789)

Associated SLRA Revisions:

The following SLRA Associated Revisions discussion supersede the SLRA Revisions in Attachment 5 of the L-2018-223 (Reference 6).

SLRA Section 17.2.2.35, Table 17-3 Item 39, and Section B.2.3.35 Structures Monitoring AMP, as also amended by L-2018-191 Attachment 7 (Reference 5), are amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions. These revisions supersede the revisions provided in L-2018-193 Attachment 16 (Reference 1) and supersede the revisions provided in L-2018-223 Attachment 5 (Reference 6).

Additionally, unrelated revisions were made to SLRA Table 17-3, Item 39, and Section B.2.3.35 via L-2018-191 Attachment 4 (Reference 5) and via L-2018-223 Attachment 4 (Reference 6) which are not included in this markup. L-2018-191 Attachment 4 SLRA revisions are related to clarifying Structures Monitoring AMP inspection frequencies. L-2018-223 Attachment 4 SLRA revisions are related to clarifying the Structures Monitoring AMP management of groundwater infiltration and through-wall leakage. L-2018-191 Attachment 7 (Reference 5) revisions to SLRA Table 17-3, Item 39, and Section B.2.3.35 are directly related to this response are included in this markup. L-2018-191 Attachment 7 revisions are related to the site-specific enhancement to the Structures Monitoring AMP for inspections of inaccessible concrete.

Revise the Appendix A Section 17.2.2.35 on page A-37 as follows:

The PTN Structures Monitoring AMP is an existing condition monitoring program that consists primarily of periodic visual inspections of plant SCs for evidence of deterioration or degradation, such as described in the American Concrete Institute (ACI) Standards 349.3R, ACI 201.1R, and Structural Engineering Institute/American Society of Civil Engineers Standard (SEI/ASCE) 11. Quantitative acceptance criteria for concrete inspections are based on ACI 349.3R. Inspections and evaluations are performed using criteria derived from industry codes and standards contained in the plant CLB including but not limited to ACI 349.3R, ACI 318, SEI/ASCE 11, and the American Institute of Steel Construction (AISC) specifications. The AMP includes preventive actions to ensure structural bolting integrity. Results from periodic inspections are trended. Due the presence of aggressive groundwater chemistry (Chlorides > 500 parts per million (ppm)), ~~the AMP includes site-specific evaluations, destructive testing, if warranted, and/or focused inspections of representative accessible (leading indicator) or below-grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil, on an interval not to exceed five years.~~ **<u>the AMP includes a site-specific enhancement to conduct a baseline visual inspection, pH analysis, a chloride concentration test, and evaluation to address the potential degradation of concrete due to exposure of aggressive chemical attack in groundwater/soil or leaching and carbonation in water-flowing. The baseline evaluation will consider site-specific OE and the baseline inspection results and will determine the additional actions (if any) that are warranted. Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the</u>**

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

Revise the Table 17-3, Item 39 on page A-107 as follows:

Table 17-3

List of SLR Commitments and Implementation Schedule (Continued)

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
39	Structures Monitoring (17.2.2.35)	XI.S6	f) ~~Perform evaluations, destructive testing, and/or focused inspections of representative accessible (leading indicator) or below-grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil. The respective evaluation/inspection/ testing interval is not to exceed 5 years.~~ **Develop a new implementing procedure or attachment to an existing implementing procedure to address aging management of inaccessible areas exposed to groundwater/soil and water-flowing. The document will include guidance to conduct a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection results will be used to conduct a baseline evaluation that will determine the additional actions (if any) that are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**	

Revise the Structures Monitoring "Program Description" in Section B.2.3.35 on page B-256 as follows:

A dewatering system is not used or part of the CLB for PTN. Structures are monitored to confirm the absence of water in-leakage or signs of concrete leaching, chemical attack or steel reinforcement degradation. Due to the presence of high chloride levels in the groundwater a site-specific enhancement to manage the concrete aging during SPEO will ~~include evaluations, destructive testing, and/or focused inspections of representative accessible (leading indicator) or below-grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil, on an interval not to exceed 5 years~~. **include a baseline visual inspection, pH analysis, and a chloride concentration test prior to the SPEO. The inspection will include a location close to the coastline/intake and a location in the main plant area for comparison and consider site-specific OE. The baseline inspection results will be used to conduct a baseline evaluation that will determine if additional actions are warranted. Additionally, the baseline evaluation results will set the subsequent inspection requirements and inspection intervals (not to exceed 5 years). Functionality throughout the period of the next scheduled inspection will be monitored. Periodic inspections (focused) and evaluation updates (not to exceed 5 years) will be performed throughout the SPEO to ensure aging of inaccessible concrete is adequately managed. Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**

Revise the Exceptions to NUREG-2191 in Section B.2.3.35 on page B-256 and B-257 as follows:

The groundwater/soil at PTN is aggressive (chlorides > 500 ppm). Since the chloride levels for seawater are much greater than 500 ppm, there is reasonable certainty that any groundwater/soil chemistry tests will consistently result in chloride level readings that are greater than 500 ppm which indicates an aggressive groundwater/soil classification, and periodic sampling and testing is not necessary and of little value. Rather, the PTN Structures Monitoring AMP includes a site-specific enhancement to address aggressive groundwater/soil **and water-flowing.**~~, that may include evaluations, destructive testing if warranted, and/or focused inspections of representative accessible (leading indicator) or below-grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil, based on site OE but not to exceed 5 year intervals.~~ **The site-specific enhancement includes the following:**

1. **A baseline inspection of inaccessible concrete will be conducted prior to the SPEO.**
    a) **The baseline inspection locations will consider site-specific OE. OE considered will include known degradation due to chlorides in ambient air and the potential for further degradation due to the aggressive groundwater as well as whether leaching and carbonation is occurring in the water-flowing environment.**

b) The baseline inspection will include excavation, visual inspection, and physical inspection of the inaccessible concrete through pH analysis and a chloride concentration test at a location close to the coastline/intake and a location in the main plant area for comparison. The baseline inspection of these two locations is a representative sample since the baseline sample is 20 percent of the population of structures most likely to experience degradation associated with groundwater (Unit 3 and 4 intake structure, discharge structure, containment structure, and auxiliary building).

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

a) The baseline evaluation will consider the baseline inspection results to determine the additional actions (if any) that are warranted. Any observed degradation will be entered into the corrective action program. The baseline inspection results are evaluated based on acceptance criteria provided in ACI 349.3R and will also consider the correlation between the chloride ion concentration necessary to induce corrosion and alkalinity level of the concrete (Reference 3). The highly alkaline environment of concrete protects the steel reinforcement from corrosion (Reference 4). Additional actions will be based on the baseline inspection results and corrective action program and may include: enhanced inspection techniques and/or frequency, destructive testing, and focused inspections of representative accessible concrete (leading indicator) or below grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil (or to leaching and carbonation in water-flowing if determined to impact intended function).

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

3. **Periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years) will be performed.**
    a) **Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval.**
    b) **The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.**
4. **Periodic evaluation updates will be performed (not to exceed 5 years) throughout the SPEO.**
    a) **Updates will be based on OE and focused periodic inspections results (and/or opportunistic inspection results if applicable) during the interval.**
    b) **The periodic evaluation results will update subsequent inspection requirements and inspection intervals (not to exceed 5 years) for the SPEO as required.**

Revise the Enhancements in Section B.2.3.35 on page B-258 as follows:

Element Affected	Enhancement
4. Detection of Aging Effects	~~Update the governing AMP procedure with a site-specific enhancement that may include evaluations, destructive testing, and/or focused inspections of representative accessible (leading indicator) or below-grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil. The respective evaluation/inspection/ testing interval is not to exceed 5 years.~~ Update the governing AMP procedure with guidance on monitoring for indications of cracking and expansion due to reaction with aggregates in concrete structures. Update the governing AMP procedure to clarify that tactile inspection may be needed for detection of elastomer hardening.

A new implementing procedure, or new attachment to the AMP governing procedure, for management of concrete exposure to aggressive groundwater/soil and water-flowing will also be developed that addresses:

Element Affected	**Enhancement**
1. Scope	**Inaccessible concrete/foundations exposed to groundwater/soil and water-flowing in scope.**
3. Parameters Monitored or Inspected	**Monitoring of the condition of inaccessible concrete, including pH and chloride concentration, of concrete exposed to groundwater/soil and water-flowing environment for evidence of aggressive chemical attack or leaching and carbonation.**

Element Affected	Enhancement
4. Detection of Aging Effects	Guidance on baseline inspection with excavation, visual inspection, and physical inspection of the inaccessible concrete through pH analysis and a chloride concentration test of concrete exposed to groundwater/soil and water-flowing at a location near the coastline and a location in the main plant area for comparison prior to the SPEO. Include periodic inspections (focused) at a frequency determined in the baseline evaluation (not to exceed 5 years). Opportunistic inspections may be used to replace or supplement the focused inspections if the inspection location is excavated for other reasons during the periodic inspection interval. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection. Degradation of accessible areas due to aging mechanisms such as chemical attack, leaching of calcium hydroxide, and carbonation can be used as an indicator for reinforced concrete conditions in inaccessible areas. Carbonation test results from accessible areas may be used as an indicator to determine the extent of condition of inaccessible reinforced concrete that is exposed to similar environmental conditions and of similar age.

Element Affected	Enhancement
5. Monitoring and Trending	Guidance for the evaluation of the baseline inspection results and related OE, with concrete exposed to ambient air and to groundwater/soil, for concrete susceptible to aging effects related to an aggressive environment prior to the SPEO to determine subsequent inspection/evaluation requirements and intervals (not to exceed 5 years), with periodic updates based on periodic inspections and OE. Guidance for the evaluation of baseline inspection results and related OE of concrete exposed to water-flowing for evidence of leaching of calcium hydroxide and carbonation will be developed prior to the SPEO. The baseline evaluation will determine whether leaching and carbonation are occurring or causing adverse effects. Subsequent inspection/evaluation requirements and intervals (not to exceed 5 years), with periodic updates based on periodic inspections and OE, will be developed if leaching or carbonation is occurring in accessible or inaccessible areas that impacts intended function. The locations inspected during the baseline inspection will continue to be monitored during the periodic inspections along with any other locations established after the baseline inspection.

Element Affected	**Enhancement**
6. Acceptance Criteria	**Acceptance criteria for concrete inspections will be consistent with ACI 349.3R and consider the correlation between the chloride ion concentration within the concrete cover necessary to induce corrosion and alkalinity level of the concrete covering the rebar for inaccessible concrete exposed to groundwater/soil and water-flowing environments.**

Associated Enclosures:

None

NRC RAI Letter No. ML18341A004 and ML18341A005 Dated January 15, 2019

RAI 3.5.1.100-1a

Background:

SRP-SLR Table 3.5-1, item 099, recommends that Class 1, Class 2, Class 3 and Class MC aluminum and stainless steel support members, welds, bolted connections and support anchorage to building structures be managed for loss of material and cracking due to SCC by either the AMP XI.M32, "One Time Inspection;" AMP XI.S3, "ASME Section XI, Subsection IWF;" or AMP XI.M36, "External Surfaces Monitoring of Mechanical Components" program. This Table 1 line item is associated with a further evaluation, SRP-SLR Section 3.5.2.2.2.4, which states the acceptance criteria for the review and the recommended actions (including AMP enhancements) when loss of material or cracking has occurred and is sufficient to potentially affect the intended function of these components.

SLRA Table 3.5-1 item 3.5-1, 099 states that the aging effects for these components is addressed by item 3.5-1, 100 using the Structures Monitoring Program.

Issue:

In its response to RAI 3.5.1.100-1, the applicant stated that item 3.5-1, 099 in the SLRA Table 3.5-1 was not used by the applicant and that item 3.5-1, 100 will be used instead to manage loss of material for aluminum and stainless steel support members, welds, bolted connections, and support anchorages using the Structures Monitoring Program. However, it is not clear how aluminum and/or stainless steel Class 1, Class 2, Class 3 or Class MC components, associated with SLRA item 3.5-1, 099, are managed (or addressed) under item 3.5-1, 100 since a review of the associated SLRA Table 2 items does not include Class 1, Class 2, Class 3 or Class MC support members, welds, bolted connections, etc. as being managed for these aging effects.

Request:

Clarify if there are aluminum and/or stainless steel Class 1, Class 2, Class 3 and/or Class MC support members, welds, bolted connections, etc., and if so, how they will be managed for loss of material and cracking due to SCC using the item 3.5-1, 100, as stated.

FPL Response:

NOTE – Instances of "ASME Class" in this response, in the SLRA, or in other RAI responses address ASME Section XI Class 1, 2, 3 or MC components at Turkey Point.

ASME Class 1, 2, 3 and MC support members, welds, bolted connections and anchorage to structure at PTN are typically steel, as described in SLRA Table 3.5.2-1, rather than stainless steel. There are no aluminum support members, welds, bolted connections or anchorage to structure for ASME Class 1, 2, 3 or MC components at PTN. As such, there are no items currently in the SLRA structural aging management evaluation tables (3.5.2-1 to 3.5.2-18) aligned to SRP-SLR item 3.5-1, 099.

Aluminum structural component types with a structural support function include the ladders and stairways for the turbine building gantry cranes in SLRA Table 3.5.2-17. Aluminum structural component types with other functions are those listed in SLRA tables 3.5.2-2, 3.5.2-6, 3.5.2-9, 3.5.2-16, and 3.5.2-18. Aluminum component types include stop logs, doors, louvers, manufactured structure for the diesel driven fire pump, perimeter stop logs and ladders/stairways. These component types are not associated with ASME Class 1, 2, 3 or MC components. The above SLRA tables and associated table 3.5-1, item 100 were updated in Attachment 18 of Reference 1 to expressly include cracking managed by the Structures Monitoring AMP for these structural members, welds, bolted connections, and anchorage to structure. There are no aluminum support members, welds, bolted connections or anchorage to structure for ASME Class 1, 2, 3 or MC components at PTN.

Stainless steel structural component types are those listed in SLRA tables 3.5.2-1, 3.5.2-2, 3.5.2-4, 3.5.2-9, 3.5.2-14, and 3.5.2-18. Stainless steel structural component types include anchorage of racks, panels, cabinets, and enclosures for electrical equipment and instrumentation, sodium tetraborate (NaTB) pH control baskets, bolting, platform supports, pipe whip restraints, hoods, electrical/instrument panels/enclosures, instrument racks, stairs/platforms/grating, and intake screens. These structural component types are not associated with support of ASME Class 1, 2, 3 or MC components. These component types are also managed by the Structures Monitoring AMP per item 3.5-1, 100 as described in the SLRA and Attachment 18 of Reference 1.

Upon further review, stainless steel support members, welds, bolted connections or anchorage to structure are not precluded for ASME Class 1, 2, 3 or MC components. Also, bolting, instrument racks and enclosures, pipe whip restraints, and anchorage to structure for instrumentation enclosures could be associated with ASME Class 1, 2, 3 or MC components. Furthermore, consistent with SLRA Sections 3.2.2.2.4 and 3.4.2.2.2, the Turkey Point OE confirms the presence of halides in both the uncontrolled indoor and outdoor environments. As such, stainless steel exposed to uncontrolled indoor and outdoor air at PTN is susceptible to cracking due to SCC and requires management via an appropriate program.

As described in SLRA Sections 17.2.2.32 and B.2.3.32, the PTN ASME Section XI, Subsection IWF AMP provides inspection and acceptance criteria and meets the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, 2007 edition with addenda through 2008, and 10 CFR 50.55a(b)(2) for ASME Class 1, 2, 3 piping and components and their associated supports. This includes any stainless steel support members or bolting. Therefore, stainless steel support members, welds, bolted connections, and support anchorage for ASME Class 1, 2, 3, or MC components, though atypical, are managed by the PTN ASME Section XI, Subsection IWF AMP. This includes visual examination, as required by the Code, for evidence of loss of material or cracking that is sufficient to potentially affect the intended function of these components. In keeping with as-low-as-reasonably achievable (ALARA) and foreign material exclusion (FME) considerations, surface or volumetric examinations are not typically required or performed as part of the ASME Section XI, Subsection IWF AMP.

To provide reasonable assurance the aging effect of cracking is managed during the SPEO for stainless steel ASME Class 1, 2, 3 or MC support members, welds, bolts or anchorage to structure, the visual inspections of the ASME Section XI, Subsection IWF AMP will be supplemented by the Structures Monitoring AMP. Should cracking due to SCC from halogens in the uncontrolled indoor and outdoor air at PTN be identified for stainless steel mechanical or non-ASME structural components, engineering evaluation will determine the appropriate refined acceptance criteria, expansion criteria, examination frequency and corrective actions. This evaluation, conducted through the Structures Monitoring AMP, will also include stainless steel ASME Class 1, 2, 3 or MC support members, welds, bolted connections or anchorage to structure. If necessary based on the Structures Monitoring AMP evaluation results, an augmented examination plan for a representative sample of stainless steel ASME Class 1, 2, 3 or MC supports will be developed in accordance with IWF-2430 as a separate part of the ASME Section XI, Subsection IWF AMP.

The SLRA, including the amendment in Attachment 18 of Reference 1, is supplemented to clarify that loss of material and cracking for stainless steel ASME Class 1, 2, 3 or MC support members, welds, bolted connections and anchorage to structure are managed by the PTN ASME Section XI, Subsection IWF AMP supplemented by the Structures Monitoring AMP. Attachments 8 and 9 of this letter address management of cracking of high-strength bolting and preventive measures for high-strength and stainless steel bolting, as well as population adjustments when a support is repaired to as-new conditions.

References:

1. FPL Letter L-2018-193 to NRC Dated November 2, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application, Safety Review Requests for Additional Information (RAI) Set 6 Responses (ADAMS Accession No. ML18311A299)

2. FPL Letter L-2018-175 to NRC Dated October 17, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application, Safety Review Requests for Additional Information (RAI) Set 5 Responses (ADAMS Accession No. ML18292A642)

3. FPL Letter L-2018-191 to NRC Dated November 28, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review Requests for Additional Information (RAI) Set 7 Responses (ADAMS Accession No. ML18334A182)

4. FPL Letter L-2018-223 to NRC Dated December 14, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review – November 15, 2018 Public Meeting Action Item Responses (ADAMS Accession Number ML18352A885)

Associated SLRA Revisions:

Current SLRA Section 3.5 Aging Management Evaluation tables, Section 3.5.2.2.2.4, Sections B.2.3.32 and B.2.3.35, Table 17-3 items 36 and 39, and Sections 17.2.2.32 and 17.2.2.35 are amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions.

Attachment 9 (RAI B.2.3.32-2) of this letter includes revisions to SLRA Sections B.2.3.32, 17.2.2.32 and Table 17-3, item 36, for the ASME Section XI, Subsection IWF AMP, related to high strength bolting, preventive measures for SCC of bolting, and population adjustments when a support(s) is(are) repaired to as-new condition. Those revisions are not duplicated in this Attachment.

Unrelated revisions to SLRA Sections B.2.3.35, 17.2.2.35 and Table 17-3, item 39, for the Structures Monitoring AMP, are contained in Attachment 10 of this letter, L-2018-223 (Reference 4) Attachment 4, L-2018-191 (Reference 3) Attachments 4 and 7, and L-2018-193 (Reference 1) Attachments 14 and 17. Those revisions are also not duplicated in this Attachment.

Revise SLRA Table 3.5-1 items 087 and 099 on pages 3.5-78 and 3.5-85, respectively, as follows:

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

Item Number	Component	Aging Effect / Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5-1, 087	Structural bolting	Loss of preload due to self-loosening	AMP XI.S3, "ASME Section XI, Section IWF	No	Consistent with NUREG-2191. The ASME Section XI, Subsection IWF AMP will be used to manage loss of preload for structural bolting for ASME class 1, 2, and 3 **<u>or MC</u>** ~~piping~~ supports exposed to uncontrolled indoor air environments
3.5-1, 099	Aluminum, stainless steel support members; welds; bolted connections; support anchorage to building structure	Loss of material due to pitting and crevice corrosion, cracking due to SCC	AMP XI.M32, "One-Time Inspection," AMP XI.S3, "ASME Section XI, Section IWF," or AMP XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes (SRP-SLR Section 3.5.2.2.2.4)	~~The Turkey Point Structures Monitoring AMP will be used to manage loss of material for aluminum and stainless steel support members, welds, bolted connections, and support anchorage. This aging effect is addressed by 3.5-1, 100.~~ **<u>There are no aluminum supports for ASME Class 1,</u>**

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

Item Number	Component	Aging Effect / Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
					2, 3, or MC components at Turkey Point. Supports for ASME Class 1, 2, 3, or MC components at Turkey Point are typically steel but may include stainless steel. Stainless steel support members, welds, bolted connections and support anchorage are managed for loss of material and cracking by the PTN ASME Section XI, Subsection IWF AMP, supplemented by the Structures Monitoring AMP. Further evaluation is documented in Section 3.5.2.2.2.4.

Add the following rows to SLRA Table 3.5.2-1 on page 3.5-88:

Table 3.5.2-1: Containment Structure and Structural Components – 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**
ASME Class 1, 2, 3 supports	**Structural support**	**Stainless steel**	**Air – indoor uncontrolled**	**Loss of material** **Cracking**	**ASME Section XI, Subsection IWF**	**III.B1.1.T-36b**	**3.5-1, 099**	**B**
Structural bolting: ASME Class 1, 2, 3 or MC supports	**Structural support**	**Stainless steel**	**Air – indoor uncontrolled**	**Loss of preload**	**ASME Section XI, Subsection IWF**	**III.B1.1.TP-229**	**3.5-1, 087**	**B**

Revise SLRA Section 3.5.2.2.2.4 on page 3.5-31 as follows:

Cracking due to SCC and loss of material due to pitting and crevice corrosion are applicable aging effects in stainless steel and aluminum alloy support members, welds, bolted connections, or support anchorages to building structures exposed to any air, condensation, or underground environment where the presence of sufficient halides (e.g., chlorides) and moisture is possible. ~~Since seawater has a high chloride level~~**As described in SLRA Sections 3.2.2.2.4 and 3.4.2.2.4, the presence of halides is confirmed in both the indoor and outdoor environments. As such**, cracking due to SCC and loss of material due to pitting and crevice corrosion is an applicable aging effect at Turkey Point for stainless steel and aluminum alloys and is monitored with the **Turkey Point ASME Section XI, Subsection IWF AMP for stainless steel supports for Class 1, 2, 3 or MC components and** Turkey Point Structures Monitoring AMP for **non-ASME supports**. **To provide reasonable assurance the aging effect of cracking is managed during the SPEO for stainless steel ASME Class 1, 2, 3 or MC support members, welds, bolts or anchorage to structure, the visual inspections of the ASME Section XI, Subsection IWF AMP will be supplemented by the Structures Monitoring AMP. Should cracking due to SCC from halogens in the uncontrolled indoor and outdoor air at PTN be identified for stainless steel mechanical or non-ASME structural components, engineering evaluation will determine the appropriate refined acceptance criteria, expansion criteria, examination frequency and corrective actions. This evaluation, conducted through the Structures Monitoring AMP, will also include stainless steel Class 1, 2, 3 or MC support members, welds, bolted connections or anchorage to structure.**

Add a paragraph to the "Program Description" in Section B.2.3.32 on page 7 of 10 in Attachment 9 to this letter as follows:

The ASME Section XI, Subsection IWF AMP will be supplemented by the Structures Monitoring AMP should cracking due to SCC in the uncontrolled indoor and outdoor air at PTN be identified for mechanical or non-ASME structural components. If necessary based on the Structures Monitoring AMP evaluation results, an augmented examination plan for a representative sample of stainless steel ASME Class 1, 2, 3 or MC supports will be developed in accordance with IWF-2430 as a separate part of the ASME Section XI, Subsection IWF AMP.

Supplement the "Enhancements" in Section B.2.3.32 on page 8 of 10 of Attachment 9 of this letter as follows:

Element Affected	Enhancement
1. Scope	**If necessary based on related Structures Monitoring AMP evaluation results (of stainless steel cracking in the uncontrolled indoor and outdoor air at PTN), develop an augmented examination plan in accordance with IWF-2430 for surface examination of a representative sample of stainless steel ASME Class 1, 2, 3 or MC supports as a separate part of the ASME Section XI, Subsection IWF AMP.**

Supplement the "Enhancements" in Section B.2.3.35 on page 3 of 7 for Attachment 18 of Reference 1 as follows:

Element Affected	Enhancement
3. Parameters Monitored or Inspected	Update the governing AMP procedure to include monitoring for cracking due to SCC for stainless steel and aluminum components.
4. Detection of Aging Effects	Update the governing AMP procedure to include guidance on surface examination inspections for cracking due to SCC for stainless steel and aluminum components.
7. Corrective Actions	**Update the governing AMP procedure to include stainless steel ASME Class 1, 2, 3 or MC support members, welds, bolted connections or anchorage in the engineering evaluation of acceptance criteria, expansion criteria, and examination frequency if cracking due to SCC in the uncontrolled indoor and outdoor air at PTN is detected for stainless steel mechanical or non-ASME structural components.**

Supplement the 3rd paragraph of current SLRA Section 17.2.2.32 on page 2 of 10 of Attachment 9 to this letter as follows:

Examinations that reveal indications are evaluated. Examinations that reveal flaws or relevant conditions that exceed the referenced acceptance standard, are expanded to

include additional examinations during the current outage. The scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). Tactile inspections of elastomeric vibration isolation elements to detect hardening if the vibration isolation function is suspect is also included. **If necessary based on related Structures Monitoring AMP evaluation results (of stainless steel cracking in the uncontrolled indoor and outdoor air at PTN), develop an augmented examination plan in accordance with IWF-2430 for a representative sample of stainless steel ASME Class 1, 2, 3 or MC supports as a separate part of the ASME Section XI, Subsection IWF AMP.**

Revise SLRA Table 17-3, item 36, on pages 4-6 of 10 for Attachment 9 of this letter as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
36	ASME Section XI, Subsection IWF (17.2.2.32)	XI.S3	Continue the existing PTN ASME Section XI, Subsection IWF AMP, including enhancement to: a) Store high strength bolts in accordance with Section 2 of Research Council for Structural Connections publication "Specification for Structural Joints Using High-Strength Bolts". b) Perform a one-time inspection, within 5 years prior to entering the SPEO, of an additional 5 percent of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports, which are not exempt from examination, that is focused on supports selected from the remaining IWF population that are considered most susceptible to age-related degradation. c) Include physical (tactile) examination of elastomeric vibration isolation elements to detect hardening if the vibration isolation function is suspect due to aging. d) {See Attachment 9 of this Letter (RAI B.2.3.32-2)} e) {See Attachment 9 of this Letter (RAI B.2.3.32-2)}	At 5 years prior to the SPEO, start one-time inspections. Complete pre-SPEO inspection and enhancements no later than 6 months or the last refueling outage prior to SPEO. Corresponding dates are as follows: PTN3: 7/19/2027 - 1/19/2032 PTN4: 4/10/2028 - 10/10/2032

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
			f) {See Attachment 9 of this Letter (RAI B.2.3.32-2)} g) {See Attachment 9 of this Letter (RAI B.2.3.32-2)} **h) If necessary based on related Structures Monitoring AMP evaluation results (of stainless steel cracking in the uncontrolled indoor and outdoor air at PTN), develop an augmented examination plan in accordance with IWF-2430 for a representative sample of stainless steel ASME Class 1, 2, 3 or MC supports as a separate part of the ASME Section XI, Subsection IWF AMP.**	

Supplement SLRA Table 17-3, item 39, on page 4 of 7 for Attachment 18 of L-2018-193 (Reference 1) as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
39	Structures Monitoring (17.2.2.35)	XI.S6	i) Revise inspection procedures to include guidance on inspection for cracking due to SCC for stainless steel and aluminum components. **j) Revise governing AMP procedure to include stainless steel ASME Class 1, 2, 3 or MC support members, welds, bolted connections or anchorage in the engineering evaluation of acceptance criteria, expansion criteria, and examination frequency if cracking due to SCC in the uncontrolled indoor and outdoor at PTN is detected for stainless steel mechanical or non-ASME structural components.**	No later than 6 months prior to the SPEO, i.e.: PTN3: 1/19/2032 PTN4: 10/10/2032

Associated Enclosures:

None

NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

6. ASME Section XI, Subsection IWE, GALL AMP XI.S1

Regulatory Basis:

Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function will be maintained consistent with the current licensing basis for the PEO. As described in SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report (the technical basis for SRP-SLR) and when evaluation of the matter in the GALL-SLR Report applies to the plant.

As indicated in SRP-SLR Section 3.5.3.2.4, both plant-specific and industry OE concerning age-related degradation are used to ensure that the AMPs are effective to manage the aging effects for which they are created. The AMPs are either enhanced, or new AMPs are developed, when it is determined through the evaluation of OE that the effects of aging may not be adequately managed.

RAI B.2.3.30-1a

Background:

SLRA Section B.2.3.30 states that the Turkey Point ASME Section XI, Subsection IWE AMP, with enhancements, will be consistent with the 10 elements of NUREG-2191 AMP XI.S1.

RAI B.2.3.30-1 requested information regarding the enhancement to the "detection of aging effects" program element of SLRA Section B.2.3.30 AMP to verify consistency with the corresponding provision in GALL-SLR AMP XI.S1 that recommends a one-time supplemental volumetric examination of containment liner areas inaccessible from one side, if triggered by plant-specific OE of occurrence or recurrence of measurable liner corrosion (material loss exceeding 10 percent of nominal thickness) initiated on the inaccessible side identified since the issuance of the first renewed license. Refer to GALL-SLR AMP XI.S1 for complete description. The technical bases stated in Table 2-30 (page 2-380) of NUREG-2221 notes that the one-time trigger-based provision for supplemental volumetric examination provides a means of plant-specific verification and confirmation of the expected effectiveness of the AMP in managing corrosion degradation in inaccessible areas of containment shell/liner for long term operation.

NextEra's response to RAI B.2.3.30-1 was provided in Attachment 8 of letter L-2018-193 dated November 2, 2018 (ADAMS Accesssion No. ML18311A299). The response on page 3 of 8 to Request 2, which requested FPL to state if there has been OE of containment liner corrosion initiated on the inaccessible (concrete) side at Turkey Point Unit 3 and/or Unit 4 since the June 6, 2002, issuance of the first renewed license, states, in part:

> The only containment liner corrosion partially attributed to the inaccessible (concrete) side is the small hole in the floor of the Unit 4 reactor cavity sump liner plate in 2006 that is described in SLRA Sections B.2.3.4 (page B-7) and B.2.3.30 (page B-235) ...
>
> Otherwise, there has been no operating experience of liner corrosion that initiated on the inaccessible side since the issuance of the PTN [Turkey Point] renewed licenses ...

Further, the revision to SLRA Section B.2.3.30 on page 7 of the response states:

> There has been no evidence of corrosion degradation on the concrete side of the liner plate, apart from the pin hole identified in the Unit 4 cavity sump area in 2006, which was partially attributed to water trapped under the liner plate ...

The revised description of this OE in SLRA Section B.2.3.4, "Boric Acid Corrosion" under Item 1 on SLRA page B-72 provided on page 5 of the response states, in part:

> In November 2006, a small hole was found in the floor of the Unit 4 reactor cavity sump liner plate. The corrosion was attributed to water trapped behind the liner plate when high pressure water was used to cut a hole in the Containment building to facilitate reactor vessel closure head (RVCH) replacement. Bulges in the liner plate provided a path for retained water to collect beneath the reactor sump. The hole was plugged and welded and the area was left with stainless steel shims on for a stainless steel support plate. ... The repair was leak tested successfully. ... It appeared to be attributed to boric acid.

Request 3 to RAI B.2.3.30-1 requested that if the response to Request 2 is yes, then (i) describe the OE and how it was addressed in the corrective action program, and (ii) explain how the conduct of the "triggered" supplemental volumetric examination, including schedule, is sufficiently captured in the revised enhancement in the response. The RAI response that appears to address Request 3(ii) states: "... This apparent localized corrosion that may have originated on the inaccessible (concrete) side, as a result of trapped water associated with RVCH temporary modification in 2005, does not affect the ASME Section XI, Subsection IWE AMP for SLR beyond the operating experience discussion ... and identifying cavity sump pit as a likely area for focused inspection."

The revised Commitment 34(b) provided on page 8 of the response reads as follows:

> Implement a one-time inspection of metal liner surfaces that samples randomly selected as well as focused (such as cavity sump pit) locations susceptible to loss of thickness due to corrosion from the concrete side if triggered by site-specific OE identified through code inspections, or other maintenance/testing activities performed since June 6, 2002. This sampling is conducted to demonstrate, with 95% confidence, that 95 percent of the accessible portion of the liner is not experiencing greater that 10 percent wall loss.
>
> Implementation schedule: Complete any applicable pre-SPEO one-time

inspections no later than 6 months or the last refueling outage prior to SPEO. Corresponding dates are as follows: PTN3: 1/19/2032, PTN4: 10/10/2032

The Turkey Point OE indicates instances of through-thickness corrosion of containment liner for Unit 3 (2010) and Unit 4 (2006); and intrusion and accumulation of water behind the liner during hydro-demolition activities for creation of containment construction openings during RV head replacement projects, which increases potential for liner corrosion from the inaccessible side of the liner.

Issue:

A. From the response to Request 2 and Request 3 of RAI B.2.3.30-1, and the associated changes to SLRA Section B.2.3.30 (last paragraph on page 7 of the response), it appears that the described OE of a small hole found in the Unit 4 reactor cavity sump liner plate has triggered or invoked the provision for the conduct of a one-time supplemental volumetric examination in GALL-SLR AMP XI.S1. However, NextEra's response does not state if Nextera plans to conduct the supplemental volumetric examination and how the two units will be treated for the examination. Further, the associated Commitment 34(b) and the implementation schedule lacks clarity and continues to use the phrase "if triggered by site-specific OE identified ...since June 6, 2002," which appears to indicate the applicant believes the provision for one-time supplemental volumetric examination may not have been triggered yet.

B. The revised Commitment 34(b) does not state what type or method of one-time examination (i.e., no mention of supplemental volumetric) is intended by the action stated in the commitment; therefore, the commitment does not demonstrate consistency with the GALL-SLR provision for supplemental volumetric examination.

C. Issue 2 in RAI B.2.3.30-1 stated that the trigger specified in the provision for supplemental volumetric examination in GALL-SLR AMP XI.S1 is the occurrence of the stated plant-specific OE since the issuance of the first renewed license without regard to how or when (PEO or SPEO) the OE is identified. Contrary to this, the revised Commitment 34(b) continues to focus on methods of identification and the implementation schedule is associated with prior to entering the SPEO (which may or may not be the case), rather than the identification of the OE. These are inconsistent with the GALL-SLR AMP XI.S1 recommendations.

Additional information and clarity with regard to the above issues is needed to ensure effectiveness of Nextera's IWE AMP in identifying and managing potential corrosion degradation from the inaccessible side of the containment liner and consistency with GALL-SLR Report.

Request:

1) State if the OE described in the response to RAI B.2.3.30-1 of a hole in the Unit 4 reactor cavity sump liner plate has triggered or invoked, for Turkey Point Units 3 and 4, the provision for the conduct of a one-time supplemental volumetric examination in GALL-SLR AMP XI.S1.
2) If the provision for conduct of a one-time supplemental volumetric examination is met, provide a revised Commitment 34(b) and implementation schedule that addresses Issues 1 thru 3 and adequately captures the conduct of the "triggered" supplemental volumetric examination. Explain NextEra's considerations and justification of the treatment of the two Units in the conduct of the examination, and how it is appropriately captured in Commitment 34(b) and its implementation schedule.
3) If the response to Request 1 above is that the one-time volumetric examination provision is not currently met, provide supporting technical justification (e.g., the loss of containment liner thickness attributed to corrosion from the inaccessible (concrete) side did not exceed 10 percent of the nominal thickness). Also, provide a revised Commitment 34(b) and implementation schedule (relative to the date of occurrence of the triggering OE, which could occur in the PEO or SPEO) that addresses Issues 1 thru 3 and appropriately captures this case, including treatment of the two Units when OE occurs in one, and considering the fact that the provision for supplemental volumetric examination can be triggered by OE anytime since the issuance of the first renewed license through the end of the SPEO.

FPL Revised Response:

The following numbered items respond to the comparable numbered follow-on requests above:

1. The OE described in the response to RAI B.2.3.30-1 of a hole in the Unit 4 reactor cavity sump liner plate has not triggered or invoked the committed one-time supplemental volumetric examination for the Turkey Point Units 3 and 4 containment liners. Attachment 6 of FPL letter L-2018-223 (Reference 3) superseded the response to RAI B.2.3.30-1, Attachment 8 of L-2018-193 (Reference 1), in its entirety. This letter was transmitted December 14, 2018 to address related items discussed at the November 5, 2018 public meeting (Reference 2) and again at the December 20, 2018 public meeting (Reference 4).

   Per the revised response to RAI B.2.3.30-1 (Attachment 6 of Reference 3), "containment liner plate operating experience at Turkey Point has not initiated the conduct of the "triggered" supplemental volumetric examination specified in GALL-SLR AMP XI.S1." The further review summarized in Attachment 6 of Reference 3 determined that the description of the Unit 4 reactor cavity sump liner plate in 2006 required clarification and was found to originate on the accessible side, versus the inaccessible side, of the liner plate. The corrosion was attributed to a combination of boric acid and galvanic corrosion. A walkdown revealed water trickling out of the

hole below a steel plate used to support one of the sump pumps, when it was displaced. The evaluation considered the shim material used for the plate. The hole was plugged and welded, the area was left with steel shims, and the steel support plate returned. The repair was leak tested successfully.

2. As described for numbered item 1 above, the provision for conduct of a one-time supplemental volumetric examination is currently not met. As such, refer to the response to numbered item 3 below.
3. The technical justification, supporting that the one-time volumetric examination provision is not currently met, is summarized in the revised response to B.2.3.30-1 (Attachment 6 of Reference 3).

   There is one reactor cavity sump pit for each containment building. The sump pit is located at the <u>lowest elevation</u> of the containment building (-17' – 8"). VT-1 inspections resulted in increasing the size of the repair plug. Nominal thickness of the liner plate in the pit is ¼" (0.25"). The reduced thickness was evaluated and determined to be acceptable. Furthermore, the underlying concrete was observed to be in sound condition, without any gap between the liner plate and the concrete, when the hole was drilled out for the repair. The cause of the reduced thickness was also considered to be corrosion on the top surface (air side) of the liner plate. After removal of the section, the bottom (concrete) side was found to be relatively smooth (no material loss). Thickness readings did not indicate reduced thickness in locations where corrosion on the top surface (air side) was not visible. Therefore, the reduced containment liner thickness is <u>not</u> attributed to corrosion from the inaccessible (concrete) side, as was previously indicated in SLRA Section B.2.3.4.

   The reactor cavity sump pits for both Units 3 and 4 were considered in the extent of condition review and historical searches were performed to ensure this instance was not a repeat or recurrent condition in either unit. The sumps have been coated, which will provide protection to the sump liner and greatly reduce the likelihood of recurrence. In addition, the sumps are periodically inspected in accordance with the ASME Section XI, IWE program. The revised response to RAI B.2.3.30-1 (Attachment 6 of Reference 3) included revisions to SLRA Sections 17.2.2.30, B.2.3.4, and B.2.3.30 to clarify the discussion of the pinhole and to provide details of the inspections to be performed if liner corrosion that originates from the inaccessible (concrete) side were to occur.

   The inspection method described in Sections 17.2.2.30, and B.2.3.30 of the SLRA, as well as Attachment 6 of Reference 3, is volumetric examination. As a supplement to Attachment 6 of Reference 3, the as-amended SLRA Table 17-3, item 34 is revised herein to clarify the implementation schedule to reflect that:

   - Site-specific OE of liner degradation originating on the inaccessible (concrete) side, for either unit, triggers the on-time volumetric examination of sample locations in both units.

- If triggered, either before or during the SPEO, the one-time inspection of liner locations in both units should be completed within 2 outages of the site-specific OE in either unit to provide reasonable assurance that loss of liner material originating on the inaccessible (concrete) side will not result in loss of component intended function.

References:

1. FPL Letter L-2018-193 to NRC Dated November 2, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application, Safety Review Requests for Additional Information (RAI) Set 6 Responses (ADAMS Accession No. ML18292A642)
2. NRC Public Meeting Agenda Dated November 5, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18315A004)
3. FPL Letter L-2018-223 to NRC Dated December 14, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review – November 15, 2018 Public Meeting Action Item Responses (ADAMS Accession No. ML18352A885)
4. NRC Public Meeting Agenda Dated December 20, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18324A789)

Associated SLRA Revisions:

As-amended SLRA Table 17-3 (item 34), in Attachments 1, 2, 6, and 7 of Reference 3, is supplemented as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions.

Supplement the as-amended commitments for the ASME Section XI Subsection IWE AMP in Table 17-3, item 34, in Attachments 1 (pg 8, 9 of 12); 2 (pg 8 of 9), 6 (pg 8, 9 of 10), and 7 (pg 6 of 7) of Reference 3 as follows (previously submitted changes in underlined font):

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
34	ASME Section XI, Subsection IWE AMP (17.2.2.30)	XI.S1	Continue the existing PTN ASME Section XI, Subsection IWE AMP, including enhancement to: a) (See SLRA pg A-103) b) Implement a one-time <u>**volumetric**</u> inspection of metal liner surfaces <u>**for both units**</u> that samples randomly selected as well as focused <u>(such as cavity sump pit)</u> locations susceptible to loss of thickness due to corrosion from the concrete side if triggered by site-specific OE identified through code inspections <u>or other maintenance/testing activities</u> performed <u>**in either unit**</u> since June 6, 2002. <u>This sampling is conducted to demonstrate, with 95% confidence, that 95% of the accessible portion of the liner is not experiencing greater than 10% wall loss.</u> c) (See response to follow-on RAI 3.5.2.1.2-1a, Attachment 14 to this letter) d) (See response to follow-on RAI B.2.3.30-2a, Attachment 13 to this letter) e) (See response to RAI 3.5.1.9-1, Attachment 2 to Reference 3)	<u>**Complete one-time inspection of containment liner locations in both units if degradation from inaccessible (concrete) side is identified, in either unit, within 2 outages of such identification prior to or during the SPEO**</u> <u>**and**</u> Complete ~~any applicable~~ pre-SPEO one-time inspections, for SCC<u>**, and other enhancements**</u> no later than 6 months or the last RFO prior to SPEO. Corresponding dates are as follows: PTN3: 1/19/2032 PTN4: 10/10/2032

Associated Enclosures:

None

NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

RAI B.2.3.30-2a

Background:

SLRA Section B.2.3.30 states that the Turkey Point ASME Section XI, Subsection IWE AMP, with enhancements, will be consistent with the 10 elements of NUREG-2191 AMP XI.S1.

RAI B.2.3.30-2 requested information related to the adequacy of aging management of Turkey Point air chase test connection interfaces at the containment floor, the degradation of which may provide pathways for intrusion of moisture into inaccessible areas of the containment liner as communicated to the industry in NRC Information Notice (IN) 2014-07, included in the "operating experience (OE)" program element of GALL-SLR AMP XI.S1, and reiterated in NRC Regulatory Issue Summary (RIS) 2016-07. NextEra's response to RAI B.2.3.30-2 in Attachment 9 of letter L-2018-193 dated November 2, 2018, indicates that no air chase test connections are currently being inspected by the IWE program and includes a new program enhancement and corresponding License Renewal Commitment 34(d) which states:

> Update inspection procedure/plan to formally include accessible air chase system test connection at the containment floor-level.

Further, new or revised commitments/enhancements have been added to the AMP in SLRA Section B.2.3.30 as a result of Set 5 response dated October 17, 2018, to RAI 3.5.2.1.2-1, and Set 6 responses dated November 2, 2018, to RAIs B.2.3.30-1, B.2.3.30-2, 3.5.1.9-1 and 3.5.1.9-2.

Issue:

A. With regard to updating the [IWE] inspection procedure/plan, the program enhancement and Commitment 34(d) does not specify the inspection actions that will be performed by the update, namely (i) the examination method (e.g., general visual, VT-1 augmented), (ii) the code provision based on which the examination will be performed (e.g., Table IWE-2500-1, Examination Category and Item No.), (iii) frequency of inspection (periodic, one-time), (iv) program elements to which applicable, and (v) Turkey Point Unit applicability

B. The response does not include an implementation schedule for program enhancements associated with: (a) Commitment 34(d) specifically, and (b) other commitments with regard to the IWE AMP, especially those that do not involve one-time inspections (general).

C. The response does not state whether or not there has been any OE at Turkey Point, Units 3 and/or 4, of moisture intrusion into inaccessible containment liner areas through the air chase test system interfaces. This information is needed to determine the adequacy of the inspection method with regard to Issue 1.

D. The response does not include Table 2 AMR results associated with components

that will be inspected in accordance with Commitment 34(d).

The staff needs the above information: (a) to determine the adequacy of the Commitment 34(d) to assure aging management of the inaccessible containment liner areas that interface with the air chase system; (b) to verify that plant-specific and industry OE is incorporated through adequate program enhancements that ensure AMP effectiveness; and (c) to assure that a specific implementation schedule is provided for all commitments associated with the AMP in SLRA Section B.2.3.30.

Request:

1. Provide a revised program enhancement and Commitment 34(d) that specifies the inspection actions that will be performed by the [IWE] inspection procedure/plan update, namely, (i) the inspection or examination method, (ii) the code provision based on which the examination will be performed (e.g., Table IWE-2500-1, Examination Category and Item No.), (iii) examination frequency, (iv) applicable program elements, and (v) applicability to Turkey Point Unit 3 and Unit 4.
2. Provide the implementation schedule for program enhancements associated with: (a) Commitment 34(d); and (b) all other commitments related to the SLRA Section B.2.3.30 AMP, especially those that do not involve one-time inspections.
3. Identify whether there has been any past OE at Turkey Point, Units 3 and/or 4, of moisture intrusion into inaccessible containment liner areas through the air chase test system interfaces that could cause degradation of the inaccessible liner areas. If OE is identified, provide a summary of the OE and justify the adequacy of the inspection method and frequency proposed in response to Request 1, considering the past OE.
4. Provide AMR results associated with SLRA Table 3.5.2-1 for components that will be subject to aging management in accordance with Commitment 34(d).

FPL Revised Response:

The following numbered items respond to the comparable numbered follow-on requests above:

1. A revised program enhancement and commitment 34(d) that specifies the inspection actions that will be performed by the ASME Section XI, Subsection IWE AMP were provided in Attachment 7 of FPL Letter L-2018-223 (Reference 1). Attachment 7 of Reference 1 supersedes in its entirety the response to RAI B.2.3.30-2 submitted via Attachment 9 of L-2018-193 (Reference 2) and includes clarifications related to topics addressed at the November 15, 2018 public meeting (Reference 3). FPL Letter L-2018-223 (Reference 1) was submitted on December 14, 2018. The RAI was also briefly addressed at the December 20, 2018 public meeting (Reference 4). The revised enhancement and commitment 34(d), as contained in Attachment 7 of Reference 1, are further clarified below –

(i) Upon further review and consistent with the recommendation of IN 2014-07, the IWE inspection procedure/plan has been adjusted to include general visual inspection of 100% of the accessible air chase test connections at the containment floor-level interfaces in each unit. Inspection procedures will be enhanced to clarify that "if a loose or degraded test connection is discovered, it will be opened for internal visual inspection of the test connection and channel/angle to confirm no water intrusion into the air chase system."

(ii) The general visual examination of accessible air chase connections are per Table IWE-2500-1, Examination Category E-A, Item E1.30. Visual inspection of air chase test connections at the containment floor-level interfaces is included in the procedure/plan, to complement the evaluation of IN 2014-07 OE for PTN, and procedure/plan will be clarified to indicate the inspections serve to confirm that no water has entered the air chase channels to support reasonable assurance that water will not degrade the containment liner from underneath (inaccessible/concrete side) and the intended function is maintained through the SPEO.

(iii) The periodic air chase test connection examinations are performed on the same frequency as other IWE inspections during the inspection interval, i.e. 10 years.

(iv) As the examination complements the PTN evaluation of IN 2014-07 OE applicability, the enhancement is to clarify the operating experience program element of the PTN ASME Section XI, Subsection IWE AMP.

(v) The examination is of accessible air chase test connections in both PTN units.

2. The implementation schedule for the five enhancements to the PTN ASME Section XI, Subsection IWE AMP is clarified in response to follow-on RAI B.2.3.30-1a (Attachment 12 of this letter). The implementation schedule for commitment 34(d), accessible air chase test connection visual inspection, is no later than 6 months or the last RFO prior to the SPEO.

3. There is no past OE at Turkey Point Units 3 and/or 4 of moisture intrusion into inaccessible containment liner areas through the air chase test system interfaces that could cause degradation of the inaccessible liner areas.

   As described in Attachment 7 of Reference 1, there have been no instances of loose or degraded air chase test connections at PTN. Some instances of moisture barrier degradation in the early 2000s included evaluation of air chase angle (toe plate) which is considered an interference surface for installation of the containment floor moisture barrier sealant. These instances included confirmation of no degradation, or only surface degradation of the angle (or seal weld), no degradation of the liner, and repair of the moisture barrier. An instance of degraded air chase angle evaluated in 2006 included removal of a portion of the angle and inspection, and thickness measurement of the liner beneath with acceptable

results. The removed air chase angle was repaired and the moisture barrier restored.

4. The 'Liner plate, anchors and attachments (accessible areas)' and "Liner plate, anchors and attachments (inaccessible areas)' component types in SLRA Table 3.5.2-1 are subject to AMR and include, but are not limited to, the air chase channels and angles. Program discussions in SLRA Sections 17.2.2.30, B.2.3.30, and Table 17-3, item 34 describe the portions affected by the individual enhancements, rather than the AMR results.

References:

1. FPL Letter L-2018-223 to NRC Dated December 14, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review – November 15, 2018 Public Meeting Action Item Responses (ADAMS Accession No. ML18352A885)
2. FPL Letter L-2018-193 to NRC Dated November 2, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application, Safety Review Requests for Additional Information (RAI) Set 6 Responses (ADAMS Accession No. ML18292A642)
3. NRC Public Meeting Agenda Dated November 5, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18315A004)
4. NRC Public Meeting Agenda Dated December 20, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18324A789)

Associated SLRA Revisions:

Current SLRA Sections B.2.3.30, 17.2.2.30 and Table 17-3 (Item 34) are further amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions. These SLRA revisions replace those provided in Attachment 7 of L-2018-223 (Reference 1) in their entirety and consider the other revisions to those sections in this letter. The SLA revisions in Attachment 7 of L-2018-223 (Reference 1) previously replaced those provided in Attachment 9 of L-2018-193 (Reference 2)

Replace the enhancement and operating experience discussions in SLRA Section B.2.3.30 on pages 3 and 4 of 7 in Attachment 7 of L-2018-223 (Reference 1) as follows:

Element Affected	Enhancement
10. Operating Experience	**<u>Update inspection procedure/plan to clarify for the inspection of accessible air chase system test connections in each unit that:</u>** • **<u>Acceptance criterion for this inspection is no evidence of loose or degraded air chase test connections, and</u>** • **<u>If a loose or degraded test connection is identified, it will be opened prior to repair and the test connection and air chase channel inspected internally to confirm no water intrusion to the air chase.</u>**

Operating Experience

<u>Industry Operating Experience</u>

NRC IN 2010-12 was issued to inform addressees of the then-recent issues involving the corrosion of the steel reactor containing building liner. The NRC expected recipients to review the information for applicability of their facilities and to consider actions, as appropriate, to avoid similar problems. In response, PTN issued an AR which evaluated that the containment liner inspection programs in effect at PTN are effective in detecting and addressing any found degradation of the containment liner due to corrosion, and ensure that the structural integrity and design function of the component are maintained. Additionally, the planned ASME Section XI Subsection IWE inspection in 2010 effectively located and corrected liner plate corrosion.

<u>NRC IN 2014-07 was issued to inform addressees of identified issues concerning degradation of floor weld leak channel systems of steel containment shell and</u>

concrete containment metallic liner that could affect leak-tightness and aging management of containment structures. This IN provides examples of operating experience at some plants of water accumulation and corrosion degradation in the leak-chase channel system that has the potential to affect the leak-tight integrity of the containment shell or liner plate. In each of the examples, the licensee had no provisions in its ISI plan to inspect any portion of the leak-chase channel system for evidence of moisture intrusion and degradation of the containment metallic shell or liner within it. The moisture intrusion and associated degradation found within leak chase channels, if left uncorrected, could have resulted in more significant corrosion degradation of the containment shell or liner and associated seam welds.

Turkey Point does have an air chase system inside the Unit 3 and Unit 4 containment structures, similar to the leak chase system discussed in IN 2014-07. Walk downs for accessible air chase test connection condition were conducted during a recent outage (PT3/4-28). Test connection (grouted pipe cap) condition was determined to be satisfactory or indeterminate (inaccessible). Subsequent reevaluation/inspection identified accessible capped air chase system test connections in each unit. The inspection procedure/plan was updated to include the accessible air chase system test connections in future IWE inspections per Table IWE-2500-1, Category E-A, item E.1.30. As described above, the ASME Section IX, Subsection IWE AMP will be updated to clarify the acceptance criterion and that any identified loose or degraded test connection will be opened for internal inspection of the test connection and channel/angle to ensure no moisture intrusion to the air chase.

Replace the paragraph added to the end of the as-amended SLRA Section 17.2.2.30 on page 4 of 7 of Attachment 7 of Reference 1 as follows:

The PTN ASME Section XI, Subsection IWE AMP will also be updated to clarify a) the acceptance criterion for the accessible air chase system test connections of "no loose or degraded connections" and b) that if a loose or degraded test connection is discovered, it will be opened prior to repair for internal inspection of the test connection and channel/angle to confirm no water intrusion to the air chase.

Replace the as-amended commitments for the ASME Section XI, Subsection IWE AMP in Table 17-3, item 34, on page 6 of 7 in Attachment 7 of Reference 1 as follows:

(Note: See Attachments 2, 3 and 6 of Reference 1 for revised Items b), and e) to this commitment; See Attachments 12 and 14 of this letter for current items b) and c), respectively, to this commitment)

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
34	ASME Section XI, Subsection IWE AMP (17.2.2.30)	XI.S1	Continue the PTN ASME Section XI, Subsection IWE AMP, including enhancement to: **d) Update inspection procedure/plan to clarify the acceptance criterion for examination of accessible air chase system test connections in each unit at the containment floor-level, and that loose or degraded test connections, if discovered, will be opened prior to repair for internal inspection of the test connection and channel/angle to confirm no water intrusion to the air chase.**	(See Attachment 12 of this letter)

Associated Enclosures:

None

NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

7. Stress Corrosion Cracking

RAI 3.5.2.1.2-1a

Regulatory Basis:

Section 54.21(a)(3) of 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 SPEO.

Background:

Nextera's revised response to RAI 3.5.2.1.2-1 in letter dated December 14, 2018, states that "if stress corrosion cracking (SCC) is detected as a result of the supplemental one-time inspections, additional inspections will be conducted in accordance with the site's corrective action process." As indicated in the revised response, the proposed supplemental one-time inspection is intended to confirm the absence of the SCC aging effect, and will be conducted using a representative sample consistent with the guidance in NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report".

As described in NUREG-2192 (SRP-SLR), an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report and when evaluation of the matter in the GALL-SLR Report applies to the plant. The GALL-SLR Report AMP XI.M32, "One-Time Inspection," which can be used to verify the absence or lack of significance of an aging effect, recommends as a corrective action to develop a periodic inspection program for the specific material, environment, and aging effect combination when an aging effect identified during an inspection does not meet acceptance criteria or projected results of the inspections do not meet the acceptance criteria described in GALL-SLR AMP XI.M32. The further evaluation in SRP-SLR Section 3.5.2.2.1.6, associated with SRP-SLR Table 3.5-1, item 010, recommends the implementation of additional appropriate examination/evaluation methods (e.g. surface examination or enhanced visual examination), as part of the AMP, to detect SCC in stainless steel (SS) components and dissimilar metal welds of the containment pressure retaining boundary.

Issue:

It is not clear how the licensee's assertion that "additional inspections [that] will be conducted in accordance with the site's corrective action process" demonstrates compliance with 10 CFR § 54.21(a)(3). The staff needs additional information to evaluate the adequacy of Nextera's corrective action(s) to manage the aging effect if the one-time inspection does not confirm the absence of SCC aging effects.

Request:

Describe the additional inspection action(s), including inspection method and interval that will be conducted following one-time inspections that identify SCC, and justify its adequacy to manage the aging effect during the SPEO. Clarify if these additional inspections will implement a periodic inspection program under IWE using additional appropriate examination/evaluation methods (e.g. surface examination or enhanced visual examination) to adequately manage and detect SCC for these components during the SPEO, as recommended by the GALL-SLR Report. Otherwise, provide adequate technical justification for the exception taken to the GALL-SLR Report recommendations.

FPL Supplemental Response:

This RAI response supplements the response provided in Attachment 1 of Reference 1 discussed during the December 20, 2018 NRC public meeting with FPL (Reference 2). For the 1 stainless steel fuel transfer tube per unit and 7 penetrations per unit with dissimilar metal welds for stainless steel piping systems with normal operating temperatures above 140°F, the respective additional inspection actions if SCC is detected by the one-time inspections will include:

1. one additional dissimilar metal weld for penetration of a stainless steel piping system above 140°F per unit until SCC is no longer detected.

   For example, if SCC is detected in the dissimilar metal weld for an RHR penetration, the other RHR penetration would be inspected. If SCC is detected by the additional inspection, a dissimilar metal weld for a CVCS penetration would also be inspected and so on.

2. periodic inspections of the stainless steel transfer tube and/or penetration dissimilar metal welds for stainless steel piping systems above 140°F if warranted based on the inspection results. These periodic inspections, if warranted, will be established through the PTN ASME Section XI, Subsection IWE AMP.

The inspection intervals established, through the site's corrective action process, for the stainless steel fuel transfer tube and/or penetration dissimilar metal welds will be appropriate based on the inspection results and will be no greater than the interval for other IWE inspections. Furthermore, appropriate enhanced visual and/or surface examination methods will be used depending on the inspection results and transfer tube/penetration configurations to detect SCC for the components during the SPEO, if necessary.

The as-amended SLRA in Attachment 1 of Reference 1 is supplemented herein to clarify the additional actions if SCC is detected by the one-time inspections.

References:

1. FPL Letter L-2018-223 to NRC Dated December 14, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application Safety Review – November 15, 2018 Public Meeting Action Item Responses (ADAMS Accession Number ML18352A885)
2. NRC Public Meeting Agenda dated December 20, 2018, Telecon Between NRC and FPL to Discuss Items Associated with the Safety Review of the Turkey Point Subsequent License Renewal Application (ADAMS Accession No. ML18324A789)

Associated SLRA Revisions:

Current SLRA Section 3.5.2.2.1.6, Section 17.2.2.30, Table 17-3 (Item 34), and Section B.2.3.30 in Reference 1 are supplemented as indicated by the following text deletion (strikethrough) and text addition (red underlined font).

Revise the 1st paragraph of the as-amended further evaluation in Section 3.5.2.2.1.6 on Attachment 1 of Reference 1, page 6 of 12 as follows:

The penetration sleeves (assemblies) penetrating the containment at Turkey Point are carbon steel. As such, SCC is not an applicable aging mechanism for penetration sleeves at Turkey Point. High-temperature piping systems that are stainless steel and penetrate the containment include dissimilar metal welds of the flued head of the steel penetration assembly to the outside of the pipe. These dissimilar metal welds are not considered susceptible to SCC. SCC requires a concentration of chloride contaminants, which are not normally present in significant quantities in containment, as well as high stress and temperatures greater than 140°F. The containment bulk ambient temperature during operation is between 50°F and 120°F, and localized temperatures at penetrations are less than 200°F by design. Furthermore, there has been no site OE of cracking of these dissimilar metal welds. Therefore, cracking of dissimilar metal welds for containment penetrations will be managed by the ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J AMPs. A supplemental one-time inspection of the stainless steel fuel transfer tube on each unit, and a representative sample of penetrations with dissimilar metal welds associated with high-temperature stainless steel piping systems, will be included as an enhancement to the ASME Section XI, Subsection IWE AMP to provide confirmation that no additional examinations/evaluations are required. Consistent with the guidance of NUREG-2191, a representative sample size is 20 percent of the ~~population up a maximum of 25~~ components at each unit. As a result, two of the penetrations with dissimilar metal welds associated with **greater than 140°F**~~high-temperature~~ stainless steel piping systems will be inspected on each unit. Additionally, if SCC is detected as a result of the supplemental one-time inspections, additional inspections will be conducted in accordance with the site's corrective action process. **This will include 1 additional penetration with dissimilar metal welds associated with greater than 140°F stainless steel piping systems on each unit until SCC is no longer detected. Periodic inspection of the stainless steel transfer tube and/or subject**

penetrations with dissimilar metal welds will be added to the PTN ASME Section XI, Subsection IWE AMP if necessary depending on the inspection results.

Add the following to the 5th paragraph of the as-amended SLRA Section 17.2.2.30 on Attachment 1 of Reference 1, page 7 of 12:

If triggered by site-specific OE, this AMP also includes a one-time supplemental volumetric examination by sampling both randomly selected and focused liner locations susceptible to corrosion that are inaccessible from one side. This AMP also includes a supplemental one-time inspection of the stainless steel fuel transfer tube on each unit, and a representative sample of penetrations with dissimilar metal welds associated with high-temperature stainless steel piping systems, will be included as an enhancement to the ASME Section XI, Subsection IWE AMP to provide confirmation that no additional examinations/evaluations are required. Consistent with the guidance of NUREG-2191, a representative sample size is 20 percent of the population at each unit. As a result, two of the penetrations with dissimilar metal welds associated with high-temperature stainless steel piping systems will be inspected on each unit. Additionally, if SCC is detected as a result of the supplemental one-time inspections, additional inspections will be conducted in accordance with the site's corrective action process. **This will include 1 additional penetration with dissimilar metal welds associated with high-temperature stainless steel piping systems on each unit until SCC is no longer detected. Periodic inspection of the stainless steel transfer tube and/or subject penetrations with dissimilar metal welds will be added to the PTN ASME Section XI, Subsection IWE AMP if necessary depending on the inspection results.**

Supplement Table 17-3 item 34 c) on Attachment 1 of Reference 1 page 8, 9 of 12 as follows:

No.		Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
34		ASME Section XI, Subsection IWE (17.2.2.30)	XI.S1	Continue the existing PTN ASME Section XI, Subsection IWE AMP, including enhancement to: c) Implement a one-time surface or enhanced visual examination of the stainless steel fuel transfer tube (including penetration sleeve and expansion joints) on each unit, and a representative sample of penetrations (two) associated with high-temperature stainless steel piping systems in frequent use on each unit. Additionally, if stress corrosion cracking (SCC) is detected as a result of the supplemental one-time inspections, additional inspections will be conducted in accordance with the site's corrective action process. **This will include 1 additional penetration with dissimilar metal welds associated with greater than 140°F stainless steel piping systems for each unit until SCC is no longer detected. Periodic inspection of the stainless steel transfer tube and/or subject penetrations with dissimilar metal welds will be added to the PTN ASME Section XI, Subsection IWE AMP if necessary depending on the inspection results.**	Complete any applicable pre-SPEO one-time inspections no later than 6 months or the last RFO prior to SPEO. Corresponding dates are as follows: PTN3: 1/19/2032 PTN4: 10/10/2032

Revise the corrective action enhancement to the as-amended SLRA Section B.2.3.30 on Attachment 1 of Reference 1, page 12 of 12 as follows:

Enhancements

The PTN ASME Section XI, Subsection IWE AMP will be enhanced as follows for alignment with NUREG-2191. The changes and enhancements will be implemented no later than six months prior to entering the SPEO.

Element Affected	Enhancement
7. Corrective Actions	If SCC is detected as a result of the supplemental one-time inspections, additional inspections will be conducted in accordance with the site's corrective action process. ~~As a minimum, two additional penetrations per unit will be inspected upon detection of SCC~~ **This includes an additional penetration per unit until SCC is no longer detected**. **If SCC is detected by the one-time inspection of a stainless steel transfer tube per unit or the inspections of representative penetrations with dissimilar metal welds associated with greater than 140°F stainless steel piping systems (population of 7 per unit), periodic inspections at the appropriate intervals will be added to the program.** **These periodic inspections, if necessary, will use the enhanced visual and/or surface examination method that is best suited for configuration and inspection results and will be at intervals no greater than those for other IWE inspections.**

Associated Enclosures:

None

NRC RAI Letter Nos. ML18341A004 and ML18341A005 Dated January 15, 2019

8. Reactor Pressure Vessel Underclad Cracking, TLAA

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 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. In accordance with 10 CFR 54.21(c)(1)(ii), the applicant has proposed to demonstrate that the TLAA for RPV underclad cracking, as described in SLRA Section 4.3.4, has been projected to the end of the subsequent period of extended operation (SPEO).

Background:
To support its 10 CFR 54.21(c)(1)(ii) disposition of the RPV underclad cracking TLAA, the SLRA included PWR Owners Group (PWROG) Report PWROG-17031-NP, Revision (Rev.) 0 in Enclosure 4 of the SLRA. The PWROG-17031-NP, Rev. 0 report provides a generic methodology for analysis of underclad cracks in Westinghouse RPVs, applicable to 80-years of plant operation. PWROG-17031-NP, Rev. 0 is not generically approved by the NRC staff for use SLR applications. Therefore, the staff is reviewing the PWROG-17031-NP, Rev. 0 report, as included in the SLRA, to determine whether this supports the applicant's TLAA disposition of 10 CFR 54.21(c)(1)(ii).

RAI 4.3.4-1a

Issue:

PWROG-17031-NP, Rev. 0 shows that the most limiting Code-allowable flaw size (per IWB-3610) for emergency and faulted conditions (Level C and D) is determined by the Large Steamline Break (LSB) transient. The Code-allowable flaw size for the LSB transient remains the same for 80 year applications as that defined in the 2002 version of this methodology, WCAP-15338-A, which is approved for 60 year applications.

PWROG-17031-NP, Rev. 0 and WCAP-15338-A indicate that if the RPV material is in the upper shelf temperature regime, a value of 200 ksi√in. is used for determining Code-allowable flaw size. The response to RAI 4.3.4-1 states that for analysis of emergency and faulted conditions, if metal temperature (T) minus RTNDT at the flaw depth is greater than 104.25 °F, 200 ksi√in is used; otherwise the KIC equation per A-4200 is used. Based on its review of this information, the NRC staff has no basis to assume that the limiting RPV beltline material is in the upper shelf temperature regime throughout the LSB transient.

Request:

1. Please provide temperature profile for the LSB transient at Turkey Point.
2. Is the material on the upper shelf for the entire transient.
3. If so, is 200 ksi in used as the limiting KIC value for the entire transient? If not, provide the KIC and RTNDT values used for determining the allowable crack length for the entire transient and confirm calculations were performed in accordance with ASME Code IWB-3610.

FPL Response:

Portions of this response contain proprietary information. Proprietary information is identified and bracketed. For each of the bracketed sections, the reasons for the proprietary classification are provided using superscripted letters "a", "c", and "e". These letters stand for:

a. The information reveals the distinguishing aspects of a process or component, structure, tool, method, etc. The prevention of its use by Westinghouse's competitors, without license from Westinghouse, gives Westinghouse a competitive economic advantage.

c. The information, if used by a competitor, would reduce the competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

e. The information reveals aspects of past, present, or future Westinghouse- or customer funded development plans and programs of potential commercial value to Westinghouse.

The numbered responses below correspond with numbered requests above.

1. The LSB transient temperature profile based on "Systems Standard Design Criteria 1.3" is representative of the temperature profile for the LSB transient at Turkey point and is shown below in Figure 1.

a, c, e

Figure 1: Westinghouse SSDC 1.3, Rev. 1 Main Steam Line Break Transient (Ref. 1)

2. The material is not on the upper shelf for the entire LSB transient. See further discussion for this transient in the response to 3 below.

3. For the large steam line break, the transient temperatures are not exclusively in the upper shelf regime. Thus, K_{Ic} calculated per ASME Section XI, Appendix A, A-4200 is used to determine the critical flaw size. The allowable flaw sizes were calculated per ASME Section XI, IWB-3600, as discussed in PWROG-17031-NP (Ref. 2), Section 5.3.2. The critical flaw sizes for the Level C and D transients are based upon a typical Westinghouse Pressurized Water Reactor (PWR) for 60 years, as referenced in PWROG-17031-NP and described in WCAP-15338-A, Section A-1 (Ref. 3). Consistent with the discussion in PWROG-17031-NP, Rev. 1 Section 5.6, RT_{NDT} is not expected to change significantly from 60 to 80 years as the rate of material embrittlement decreases at higher fluence levels. This "saturation" effect is evidenced by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," Fig. 1.

   A small increase in RT_{NDT} as a result of any additional neutron embrittlement can be accommodated given that the maximum flaw depth due to fatigue crack growth for 80 years is 0.4267 inches as shown in PWROG 17031-NP, Section 5.4. This represents a significant margin compared to the Normal/Upset/Test allowable flaw depth of 0.67 inches. As a further conservatism, underclad cracks are assumed to be surface flaws

which results in a conservative K_I. The surface flaw assumption also results in a higher calculated fatigue crack growth rate as it considers a water environment.

Regardless of the RT_{NDT} value utilized for the critical flaw size determination in WCAP-15338-A, protecting the beltline region of a PWR Reactor Vessel (RV) from fracture during a large steam line break and small loss-of-coolant accident outside of the upper-shelf regime is ultimately ensured through compliance with 10 CFR 50.61. This regulation requires licensees of all operating PWRs to maintain licensed values of the reference temperature for pressurized thermal shock (RT_{PTS}) for each beltline material. These values must be below the screening values of 270°F for plates, forgings, and axial welds or below 300°F for circumferential welds. If RT_{PTS} values are projected to exceed the screening criteria, "the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion." Additionally, licensees may subject the RV to thermal annealing or demonstrate compliance to PTS regulations via evaluation consistent with 10 CFR 50.61a. Per the Turkey Point Subsequent License Renewal Application, Turkey Point Units 3 and 4 do not exceed the 10 CFR 50.61 screening criteria at the end of the SPEO.

The NRC's original position on Pressurized Thermal Shock is summarized in Policy Issue SECY-82-465 (Ref. 4), which affirms that the risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criterion is acceptable. It also provides, in significant detail, the basis for this conclusion, which includes an analysis of PTS transients. The PTS transients analyzed include main steam line break and small LOCA, amongst others.

A subsequent NRC study of PTS was published in NUREG-1874 (Ref. 5), which stated that "It is now widely recognized that the state of knowledge and data limitations in the early 1980s necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis for the current PTS Rule." NUREG-1874 confirms, through additional analysis of PTS transients, that the 10 CFR 50.61 methods and screening criteria are conservative.

Thus, since a main steam line break transient is considered a PTS transient, mandatory compliance with 10 CFR 50.61 or 10 CFR 50.61a inherently ensures beltline vessel integrity during this transient.

References:

1. Westinghouse Report, SSDC 1.3, Rev. 1, "Systems Standard Design Criteria Nuclear Steam Supply System Design Transients," April 5, 1971.
2. Pressurized Water Reactor Owners Group (PWROG) Report, PWROG-17031-NP, Revision 1, "Update for Subsequent License Renewal: WCAP-15338-A, 'A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants'," May 2018.

3. Westinghouse Report, WCAP-15338-A, Rev. 0, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," October 2002
4. NRC Policy Issue SECY-82-465, "Pressurized Thermal Shock," November 23, 1982. ([ADAMS Accession No. ML16232A574)
5. U.S. NRC Report NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 2010

Associated SLRA Revisions:

None

Associated Enclosures:

Westinghouse Letter CAW-19-4855 dated January 22, 2019, Application for Withholding Proprietary Information from Public Disclosure

End of Non-Proprietary Attachments

Proprietary Attachment 15P is Inserted Beginning on the Following Page

Enclosure

Westinghouse Letter CAW-19-4855 dated January 22, 2019

Application for Withholding Proprietary Information from Public Disclosure

Westinghouse Affidavit CAW-19-4855

Proprietary Information Notice and Copyright Notice

Regarding

LTR-SDA-18-131-P, Rev. 0, "Westinghouse Response to NRC Request for Additional Information on Turkey Point Subsequent License Renewal (RAI 4.3.4-1a)" (Proprietary)

Westinghouse Non-Proprietary Class 3

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CAW-19-4855

January 22, 2019

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-SDA-18-131-P, Rev. 0, "Westinghouse Response to NRC Request for Additional Information on Turkey Point Subsequent License Renewal (RAI 4.3.4-1a)" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-19-4855 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Florida Power and Light Company.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-19-4855, and should be addressed to Camille Zozula, Manager, Facilities and Infrastructure Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2, Suite 259, Cranberry Township, Pennsylvania 16066.

Korey L. Hosack, Manager
Product Line Regulatory Support

Enclosures:

1. Affidavit CAW-19-4855
2. Proprietary Information Notice and Copyright Notice
3. LTR-SDA-18-131-P, Rev. 0, "Westinghouse Response to NRC Request for Additional Information on Turkey Point Subsequent License Renewal (RAI 4.3.4-1a)" (Proprietary)

CAW-19-4855

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Korey L. Hosack, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 20190122

Korey L. Hosack
Product Line Regulatory Support

(1) I am Manager, Product Line Regulatory Support, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SDA-18-131-P, Rev. 0, "Westinghouse Response to NRC Request for Additional Information on Turkey Point Subsequent License Renewal (RAI 4.3.4-1a)" (Proprietary), for submittal to the Commission, being transmitted by Florida Power and Light Company letter. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's request for NRC approval of PWROG-17031, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to update the 60 year fatigue crack growth analysis in WCAP-15338-A and confirm that the analysis is applicable to subsequent license renewal (SLR), up to 80 years of operation.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of continuing life extension support.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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