TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4 (TURKEY POINT)
SUBSEQUENT LICENSE RENEWAL APPLICATION (SLRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAIS)
SAFETY - SET 8

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

RAI B.2.3.9-1a:

<u>Background:</u>

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.

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.

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.

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).

3. Concrete, GALL AMR XI.S6

Background:

In SLRA Section 3.5.2.2.2.6, “Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation,” (Rev. 1) (ADAMS Accession No. ML18283A308), the applicant describes calculations performed to determine the projected peak neutron fluence and gamma dose within the Turkey Point reactor cavity for 80 years of plant operation for downstream use in structural analysis calculations, which are used to demonstrate sufficient margin exists for the reactor vessel (RV) supports to carry various design basis loads; RAIs 3.5.2.2.2.6-1 through 3.5.2.2.2.6-3 below relate to these calculations. The applicant also describes calculations

performed to estimate RV structural support steel irradiation damage to demonstrate that sufficient ductility exists in RV structural support steel to support the RV inlet and outlet nozzles; RAI 3.5.2.2.2.6-4 relates to these calculations.

Regulatory Basis:

NUREG-2192 (SRP-SLR), Section 3.5.2.2.2.6, “Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation,” describes a method for determining whether the applicant has met the requirements of the NRC regulations in 10 CFR 54.21 by providing the acceptance criterion for the aging management of the reduction of strength and mechanical properties of concrete due to irradiation as it pertains to the reactor biological shield (or bioshield) wall. NUREG-2192 (SRP-SLR), Section 3.5.2.2.2.6 states:

> Reduction of strength, loss of mechanical properties, and cracking due to irradiation could occur in PWR and BWR Group 4 concrete structures that are exposed to high levels of neutron and gamma radiation. These structures include the reactor (primary/biological) shield wall, the sacrificial shield wall, and the reactor vessel support/pedestal structure. Data related to the effects and significance of neutron and gamma radiation on concrete mechanical and physical properties is limited, especially for conditions (dose, temperature, etc.) representative of light-water reactor (LWR) plants. However, based on literature review of existing research, radiation fluence limits of 1×10^{19} [neutrons per square centimeter (n/cm^2)] neutron radiation and 1×10^{8} [Gray (Gy)] (1×10^{10} rad) gamma dose are considered conservative radiation exposure levels beyond which concrete material properties may begin to degrade markedly.
>
> Further evaluation is recommended of a plant-specific program to manage aging effects of irradiation if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron (fluence cutoff [energy greater than 0.1 million-electron-volts [(E > 0.1 MeV)] or gamma radiation exceeds the respective threshold level during the subsequent period of extended operation or if plant-specific [operating experience] of concrete irradiation degradation exists that may impact intended functions. 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 acceptance criteria are described in BTP [Branch Technical Position] RLSB-1 (Appendix A.1 of this SRP-SLR).

Additionally, 10 CFR 54.21 requires SLR applicants to perform an integrated plant assessment. For Turkey Point, the applicant has determined that this includes assessing the effects of irradiation damage resulting in a loss of fracture toughness of RV structural steel supports. Some of the RV structural steel support elements are partially embedded in the bioshield wall, but have exposed beams protruding from the bioshield wall including saddles that support the RV inlet and outlet nozzles.

Further, 10 CFR 54.21(a)(3) 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 all structures and components (SCs) that have been scoped and screened-in for subsequent license renewal, for the SPEO. As

described in 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. SRP-SLR Section 1.2.1 notes that the SRP-SLR and GALL-SLR Reports do not provide a comprehensive list of all potential aging effects that may be applicable to structures subject to an AMR. Therefore, applicants should perform plant-specific AMRs for additional aging effects that are applicable. Branch Technical Position A.1.2, in Appendix A of the SRP-SLR, provides additional guidance on identifying applicable aging effects.

Finally, SRP-SLR Section 3.5.2.2.2.6 states that reduction of strength, loss of mechanical properties, and cracking due to irradiation could occur in PWR Group 4 concrete structures (e.g., reactor (primary/biological) shield wall, the sacrificial shield wall, and the RV support/pedestal structure) that are exposed to high levels of neutron and gamma radiation. The SRP-SLR recommends further evaluation of a plant-specific program to manage aging effects of irradiation if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete (also referred to as reinforced/composite concrete) from neutron (fluence cutoff energy E > 0.1 MeV) or gamma radiation exceeds the SRP-SLR threshold levels during the SPEO or if plant-specific OE of concrete irradiation degradation exists that may impact intended functions.

RAI 3.5.2.2.2.6-1

The basis for SLRA Section 3.5.2.2.2.6 (Rev. 1) is documented in Audit Document FPLCORP020-REPT-130, Rev. 1, “Primary Shield Wall Irradiation Evaluation,” October 2018. As explained in Audit Document FPLCORP020-REPT-130, Rev. 1, Appendix G, “Radiation Analysis Support on Turkey Point Irradiated Concrete Exposures for Subsequent License Renewal Application,” on pages G-7 and G-10 of G-11, the peak fluence determined by the applicant is based on values reported by Westinghouse in Audit Document Westinghouse Letter FPL-09-41, “Turkey Point Units 3 and 4 - Extended Power Uprate (EPU),” Response to Shaw Request for Radiological Information, February 2009. These values are: (1) based on an azimuthally averaged value instead of the peak azimuthal value and (2) reported at a location 8 centimeters (cm) into the shield wall concrete instead of at the surface.

Provide a justification for using the azimuthally averaged value 8 cm into the shield wall concrete instead of the peak surface fluence value given that the stated intent of SLRA Section 3.5.2.2.2.6 (Rev. 1) is to determine maximum fluence values incident on the shield wall.

RAI 3.5.2.2.2.6-2

Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence,” Section 1.4, “Methodology Qualification and Uncertainty Estimates” (ADAMS No. ML010890301) is germane to reactor pressure vessel applications. However, it provides some general guidance useful for fluence method qualification. In order to establish the accuracy of the fluence estimates supporting SLRA Section 3.5.2.2.2.6:

a. Validate the fluence methods chosen to estimate neutron and gamma fluence incident on and throughout the shield wall for the energy ranges of interest (i.e., E > 0.1 MeV for neutrons and for all gamma energies). Include comparisons with applicable measurement and calculational benchmarks. Include additional margin for uncertainty as appropriate if no applicable measurement or calculational benchmarks are available.

b. Analytic uncertainty has not been quantified for the peak 80 year fluence values provided. Provide analytic uncertainty estimates for the reported fluence values, including all relevant sources of uncertainty, to demonstrate the accuracy of the methodology.

RAI 3.5.2.2.2.6-3

Will be issued with RAI Set 9.

RAI 3.5.2.2.2.6-4

Regarding the RV support displacements per atom (dpa) calculation as reviewed in Audit Document FPLCORP020-REPT-130, Rev. 1, Appendix E, "Irradiated Reactor Vessel Supports Evaluation," pages. E-5 and E-6 of E-9, supporting SLRA Section 3.5.2.2.2.6:

a. The calculation of the dpa rate was performed incorrectly based on the method chosen. Correct the dpa rate calculation by:

   i. Using the total integrated neutron flux given by Equation 12 in the dpa rate calculation method reference.

   ii. Using the average neutron energy as given by Equation 13 in the dpa rate calculation method reference.

b. The model used to determine dpa has not been validated by comparison to an appropriate benchmark or standard (e.g., ASTM E693-17, "Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA)") and no consideration of dpa cross-section uncertainty was considered. Validate the accuracy of the dpa estimate by:

   i. Comparing the dpa calculational model to an appropriate benchmark or standard and determining if application of a bias and/or uncertainty is warranted.

   ii. Accounting for additional uncertainty in the dpa calculation due to:

      1. Total fluence uncertainty affecting the total fluence term in Equation 11 of the dpa rate calculation method reference. Note: (1) that this request is related to RAI 3.5.2.2.2.6-3 and (2) any changes in the peak fluence due to the response to RAI 3.5.2.2.2.6-1 may necessitate an update to the total fluence term used in Equation 11.

      2. Fluence spectrum uncertainty affecting the average energy term (which is based on a weighting function equal to the fluence spectrum) in Equation 13 of the dpa rate calculation method reference.

      3. Nuclear data uncertainty affecting the cross-section term in Equation 11 of the dpa rate calculation method reference.

c. SLRA Section 3.5.2.2.2.6 references a generic E > 1 MeV axial neutron flux profile corresponding to the neutron flux incident on a shield wall. The applicant explains that the profile shows that the flux at the top of active fuel region is 40% of the peak neutron

> flux at the top of the active fuel region. This 0.4 factor is combined with the Turkey Point peak E > 0.1 MeV and E > 1 MeV neutron fluxes incident on the Turkey Point shield wall and are used as inputs to the dpa rate calculation method.
>
> Verify that the assumption of 0.4 for the axial peaking factor is bounding (or sufficiently representative) of past actual and future expected axial peaking factors corresponding to the most influential peripheral fuel assemblies with respect to neutron fluence incident on the shield wall at Turkey Point for 80 years of operation.

RAI 3.5.2.2.2.6-5

In the discussion of the bolting, SLRA Section 3.5.2.2.2.6 (Rev. 1) states:

> Based on the review of NUREG-1509 and the design documentation of the PTN [Turkey Point] RV support bolting, no further evaluation for reduction in fracture toughness for the bolting is required.

Describe in sufficient detail the analysis which led to the stated conclusion. Include descriptions of the specific information from NUREG-1509 and the applicable design documentation utilized in the analysis that provide the basis for the conclusions. This should include identification of the bolt material, neutron fluence at the location of the support bolting, estimation of the radiation embrittlement, and a description of the analyses following the flow charts in NUREG-1509, Figures 4-1, 4-2, 4-3, 4-4 and 4-5, as appropriate.

RAI 3.5.2.2.2.6-6

The analysis for ∆NDT in SLRA Section 3.5.2.2.2.6 (Rev. 1) uses the fitted curve from Figure 3-1 of NUREG-1509 citing:

> The fitted curve was utilized because the test data points associated with the dpa being evaluated (in the 1 to 5 x 10^{-3} range) are all below the fitted data curve. Additionally, the upper-bound curve in the region of interest included points that combine neutron and gamma exposure and are based on only 2 worst case data points.

The trend curve in Figure 3-1 of NUREG-1509 was determined based on all of the data identified in Figure 3-1. Using the methodology of NUREG-1509, ∆NDT is determined using the bounding curve in Figure 3-1; the methodology does not include options for using the fitted curve, or restricting consideration of the data to a limited range of neutron fluence values. Section 4.4 ("Accurate Analysis") of NUREG-1509 states:

> The initial NDT temperature of the RPV support material should be evaluated in accordance with the notes pertinent to Fig. 4-2. The radiation-induced ∆NDT should be estimated from the upper bound correlation curve from Fig. 3-1.

This is consistent with the example provided in Section 3.3 ("Trojan Dosimetry"). Given the data and bounding curve fit in Figure 3-1, the basis for deviating from the methodology in NUREG-1509, using the fitted curve instead of the bounding curve, is not sufficient or clear. Provide a thorough technical basis for the use of the fitted curve, including appropriate consideration of the uncertainty in the estimate of ∆NDT.

RAI 3.5.2.2.2.6-7

The transition temperature analysis described in Figure 4-4 of NUREG-1509 features a step described in the flowchart box labeled "Evaluate TT_{EOL} + ***Margin*** ≤ LST", where TT_{EOL} is determined in the example cases based on the upper bound curve. Section 4.3.4.2 of NUREG-1509 states:

> Uncertainties related to NDT determinations demand that a margin of safety be maintained between the LST and the NDT temperature, such as provided in Appendix R, Figure R-1200-1, Ref. 18.

Identify the appropriate margin that was used in the evaluation as addressed in NUREG-1509. Describe the analysis following the flowchart in Figure 4-4, particularly the flowchart box to determine "Evaluate TT_{EOL} + ***Margin*** ≤ LST" and the subsequent actions (following the flowchart in Fig. 4-4 of NUREG-1509) that may result.

RAI 3.5.2.2.2.6-8

Will be issued with RAI Set 9.

RAI 3.5.2.2.2.6-9

Section 54.21(1) of 10 CFR states that an integrated plant assessment (IPA) must--For those systems, structures, and components within the scope of this part, as delineated in 10 CFR 54.4, identify and list those structures and components subject to an AMR. Determine if an AMR item is required. If so, identify AMR items that address degradation of the RPV supports in the presence of a neutron environment, including plans for aging management.

a. Describe any inspection, examinations or other activities that would provide an indication of the condition of the supports of interest at Turkey Point.

b. Provide the findings of the activities described in (a), in particular any degradation that could adversely affect the performance of the supports in the presence of a high radiation environment.

Conversely, provide a justification for why an AMR item is not required.

RAI 3.5.2.2.2.6-10

<u>Background:</u>

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 $3.57x10^{19}$ 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 $1x10^{19}$ 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 $1x10^{19}$ 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.

RAI 3.5.2.2.2.6-11

Will be issued with RAI Set 9.

RAI 3.5.2.2.2.6-12

<u>Background:</u>

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 Turkey Point 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 original 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 re-calculated for the reduced concrete section due to the crack under the CLB loading and considering the reduced strengths and modulus of the irradiated concrete. 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 has 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 [PSW] concrete will be significantly reduced. For the [PSW], 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.

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 RV 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.

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

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(s) 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 and when evaluation of the matter in the GALL-SLR Report applies to the plant.

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.

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.

5. Structures Monitoring Program, GALL AMP XI.S6

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 PEO. As described in 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.

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.

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.

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 actionstated 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.

RAI B.2.3.30-2a

<u>Background:</u>

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.

<u>Issue:</u>

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).

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.

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 RT_{NDT} at the flaw depth is greater than 104.25 °F, 200 ksi√in is used; otherwise the K_{IC} 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 K_{IC} value for the entire transient? If not, provide the K_{IC} and RT_{NDT} values used for determining the allowable crack length for the entire transient and confirm calculations were performed in accordance with ASME Code IWB-3610.