NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

Section I – Fluence

Background

In SLRA Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation," (Rev. 1) (Agencywide Documents Access and Management System (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.

6-2Regulatory Basis

NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," (or 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 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.

RAI 3.5.2.2.2.6-1

<u>Additional Background</u>

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.

<u>Issue</u>

(1) The SLRA states that “calculations performed to determine the projected peak neutron fluence and gamma dose within the PTN Unit 3 and Unit 4 reactor cavity for 80 years of plant operation have shown that they are above the radiation exposure thresholds [stated in the SRP-SLR]…” However, these values used in the analysis are based on an azimuthally averaged value instead of the peak azimuthal value indicated in the SLRA.

(2) The SLRA states that “neutron fluence and gamma dose incident on the primary shield wall were determined as follows...” However, the fluence values in the audited documents are reported at a location 8 centimeters (cm) into the shield wall concrete instead of at the surface.

<u>Request</u>

Clarify the apparent discrepancies between the audit documents which use the azimuthally averaged value 8 cm into the shield wall concrete instead of the peak surface fluence value as stated in SLRA Section 3.5.2.2.2.6 (Rev. 1). Provide justification supporting the chosen approach.

FPL Response:

As noted in the attachment to Reference 1, neutron flux values reported as incident on the primary shield wall for the extended power uprate (EPU) were used to determine the end of SPEO fluence on the primary shield wall. Westinghouse provided additional details regarding the fluence analysis performed for EPU indicating that the fluence values were: (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. Upon receipt of this information, and to ensure that the end of SPEO fluence value incident on the primary shield wall utilized for the concrete degradation evaluation was conservative, Westinghouse performed additional PTN specific SLR calculations which satisfy the guidance set forth in Regulatory Guide 1.190 (Reference 2), using the NRC approved methodology in WCAP-14040-A (Reference 3), to determine the maximum neutron fluence (E > 0.1 MeV) incident on the primary shield wall at the end of the SPEO.

Westinghouse calculated the maximum neutron fluence (E > 0.1 MeV) at 72 Effective Full-Power Years (EFPY) (end of SPEO) on the primary shield wall at Turkey Point Units 3 and 4 based on the reactor models and radiation transport calculations performed for the SLR reactor pressure vessel (RPV) neutron exposure for the Turkey Point Units 3 and 4 SPEO (summarized in SLRA Section 4.2). These discrete ordinates radiation transport calculations were performed on a fuel-cycle-specific basis at Turkey Point Units 3 and 4. Plant-specific forward transport calculations were carried out using the two-dimensional (2D) / one-dimensional (1D) fluence rate synthesis technique. All of the transport calculations were carried out using the DORT discrete ordinates code (Reference 4) coupled with the BUGLE-96 cross-section library (Reference 5). An investigation of the RPV exposure results summarized in SLRA Section 4.2 indicated that for the traditional beltline region (RPV region directly surrounding the height of the active fuel) Turkey Point Units 3 and 4 gave very similar results. Based on this similarity, the use of either Turkey Point Unit 3 or Unit 4 in the analysis would provide results applicable to both units at the maximum neutron fluence (E > 0.1 MeV) location on the primary shield wall. Accordingly, the Turkey Point Unit 4 calculations were utilized to calculate the maximum neutron fluence (E > 0.1 MeV) on the primary shield wall and this result was applied to both Turkey Point Units 3 and 4.

The maximum neutron fluence (E > 0.1 MeV) on the primary shield wall was extracted at the radial location corresponding to the air-side surface of (incident on) the primary shield wall liner, at an azimuthal location of 0° (the azimuthal location where the maximum fast neutron fluence (E > 0.1 MeV) occurs on the concrete biological shield considering 0°, 15°, 30°, and 45°), and at an elevation providing the maximum exposure. Similar to the analysis supporting RPV neutron exposure, future projections included a 20% positive bias on the peripheral and re-entrant corner assemblies on the projection fuel cycle. Peripheral assemblies have one or more faces exposed to the core baffle plates and re-entrant corner assemblies have one corner exposed the core baffle plates.

The Westinghouse calculations described above determined that the maximum neutron fluence (E > 0.1 MeV) at the air-side surface of (incident on) the primary shield wall liner was 3.24×10^{19} n/cm^2 at 72 EFPY. Thus, the fluence value (E > 0.1 MeV) of 3.57×10^{19} n/cm^2 incident on the primary shield wall used for the concrete degradation evaluation is conservative. Note that the calculation summary for the additional Westinghouse calculations is available on the ePortal.

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).

2. Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” Office of Nuclear Regulatory Research, March 2001.

3. Westinghouse Electric Company Document WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

4. RSICC Computer Code Collection CCC-650, “DOORS 3.2, One-, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System,” Radiation Safety Information Computational Center, Oak Ridge National Laboratory, April 1998.

5. RSICC Data Library Collection DLC-185, “BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” Radiation Safety Information Computational Center, Oak Ridge National Laboratory, July 1999.

Associated SLRA Revisions:

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

Insert the following as the third paragraph on page 2 of 19 of the attachment to Reference 1 as follows:

Westinghouse provided additional details regarding the fluence analysis performed for EPU indicating that the fluence values were: (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. Upon receipt of this information, and to ensure that the end of SPEO fluence value incident on the primary shield wall utilized for the concrete degradation evaluation was conservative, Westinghouse performed additional PTN specific SLR calculations which satisfy the requirements set forth in Regulatory Guide 1.190, using the NRC approved methodology in WCAP-14040-A, to determine the maximum neutron fluence (E > 0.1 MeV) incident on the primary shield wall at the end of the SPEO. This calculation determined that the maximum neutron fluence (E > 0.1 MeV) incident on the primary shield wall surface was 3.24 x 10^{19} n/cm^2 at 72 EFPY. Thus, the fluence value (E > 0.1 MeV) of 3.57 x 10^{19} n/cm^2 incident on the primary shield wall used for the concrete degradation evaluation is conservative.

Associated Enclosures:

None

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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.

<u>Issue</u>

In order to assess the methodology for determining fluence, the staff needs additional information that establishes the accuracy of the fluence estimates supporting SLRA Section 3.5.2.2.2.6.

<u>Request</u>

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. Quantify analytic uncertainty estimates for the reported fluence values of peak 80 year fluence, including all relevant sources of uncertainty, to demonstrate the accuracy of the methodology or provide a basis for not doing so.

FPL Response:

The following lettered items respond to the comparable lettered requests above, and refer to the additional calculations performed by Westinghouse described in the response to RAI 3.5.2.2.2.6-1 in Attachment 1 to this letter:

a. Westinghouse has an ex-vessel neutron dosimetry (EVND) program, where neutron dosimeters are placed in the cavity region in front of the primary shield wall for one or more fuel cycles for irradiation. Typical fast neutron reaction rates analyzed from EVND are Cu-63(n,α)Co-60, Ti-46(n,p)Sc-46, Fe-54(n,p)Mn-54, Ni-58(n,p)Co-58, U-238(n,f)FP, Nb-93(n,n')Nb-93m, and Np-237(n,f)FP. Among these reactions, cross sections of U-238(n,f)FP, Nb-93(n,n')Nb-93m, and Np-237(n,f)FP cover energy ranges greater than 0.1 MeV, and these three reaction rates are selected to validate the Westinghouse fluence methodology in calculating neutron fluences (E > 0.1 MeV) incident on the primary shield wall.

Table 1 gives the number of data points for U-238(n,f)FP, Nb-93(n,n')Nb-93m, and Np-237(n,f)FP from a database of 11 three-loop neutron pad (neutron shield panel) plants. All EVND locations listed in Table 1 below are from core midplane locations.

Table 2 below lists the EVND measurement-to-calculation reaction rate ratios (M/C) averaged over each data point and the associated percent standard deviation (% std dev). The M/C results listed in Table 2 provide a validation of the results of the neutron transport calculations for the primary shield wall. These data comparisons show that the measurements and calculations agree within 11% and are well within the 20% criterion specified in Regulatory Guide 1.190 for RPV neutron fluence analysis.

Table 1
EVND Data for Neutron (E > 0.1 MeV) Reaction Rates

EVND Location	Number of Data Points per Reaction		
	U-238(n,f)FP	Nb-93(n,n')Nb-93m	Np-237(n,f)FP
0^o Midplane	17	6	16
15^o Midplane	8	5	8
30^o Midplane	8	5	8
45^o Midplane	7	5	7

Table 2
EVND Measurement-to-Calculation Reaction Rates Ratios

Reaction	0^o Midplane		15^o Midplane		30^o Midplane		45^o Midplane	
	Average	% std dev	Average	% std dev	Average	% std dev	Average	% std dev
U-238(n,f)FP	0.96	3.7	0.89	4.7	0.91	8.7	0.95	8.5
Nb-93(n,n')Nb-93m	0.92	10.2	0.92	6.8	0.96	11.7	0.90	10.7
Np-237(n,f)FP	0.99	4.9	0.92	8.7	0.92	3.7	0.99	4.5

In terms of measurement benchmarks, comparisons of calculations with in-vessel surveillance capsule and ex-vessel reactor cavity measurements from the H. B. Robinson reactor benchmark experiment were used in the RPV neutron fluence uncertainty estimate in WCAP-14040-A, Revision 4 (Ref. 1). The H. B. Robinson in-vessel surveillance capsules contain the analysis of U-238(n,f)FP and Np- 37(n,f)FP among other nuclides and the ex-vessel cavity capsules contain the analysis of U-238(n,f)FP among other nuclides. The uncertainty obtained from the H. B. Robinson benchmark is given as 3% in WCAP-14040-A, Revision 4. A calculational benchmark specific to the primary shield wall was not found; however, comparisons with the measurement database and the H. B. Robinson benchmark provide confidence in the validation of the neutron fluence (E > 0.1 MeV) calculations for the primary shield wall.

For the gamma fluence validation, the VENUS-1 benchmark has been used to compare calculated-to-measured gamma heating rates at the inner baffle, outer baffle, core barrel, and neutron pad (neutron shield panel) regions (Ref. 2). WCAP-14040-A, Revision 4 indicates that:

> *In some extreme cases where part length poisons or shielded fuel assemblies have been inserted into the reactor core to reduce the fluence locally in the vicinity of key vessel materials, the calculational approach may be modified to use either a multi-channel synthesis approach or a fully three-dimensional technique.*

In order to model the heterogeneous regions of the VENUS-1 benchmark accurately, the TORT code was used in the analysis of VENUS-1. Table 5 of Reference 2 demonstrates that the zone-averaged (from the inner baffle, outer baffle, core barrel, and neutron pad zones) calculated-to-measured gamma heating rate is 1.08 ± 7.3%. While the gamma heating rates for VENUS-1 were calculated for the reactor internals, there were no other measurement benchmarks found for the primary shield wall. A calculational benchmark for gamma fluences was also not available.

b. An analytic uncertainty calculation was not performed for the maximum neutron fluence (E > 0.1 MeV) and gamma fluences at the end of the SPEO for the primary shield wall. An analytic uncertainty for the EVND has been quantified as 12% in Reference 3. This uncertainty is associated with calculations for fast neutron fluence (E > 1.0 MeV) and can be used for the primary shield wall for this energy range. A comparison of the EVND location uncertainties associated with best-estimate neutron fluences from least squares evaluations for the H. B. Robinson benchmark is 7% for the fast neutron fluence rate (E > 1.0 MeV) and 13% for the neutron fluence rate (E > 0.1 MeV) (Ref. 3). The difference in uncertainties between neutron fluence rates for E > 1.0 MeV and E > 0.1 MeV from least squares evaluations indicates that the analytic uncertainties for neutron fluences E > 0.1 MeV could be higher compared to those for E > 1.0 MeV. However, the neutron fluence at the end of SPEO was calculated conservatively by Westinghouse by using a 20% positive bias on the peripheral and re-entrant corner assemblies on the projection fuel cycle and is expected cover the additional analytic uncertainties in the neutron fluence (E > 0.1 MeV) calculation.

For the maximum gamma dose on the primary shield wall, Westinghouse calculated a conservative value assuming that the most conservative out-in fuel cycle from Turkey Point Unit 4 applied through the end of the SPEO. This conservative maximum gamma dose on the concrete biological shield was shown to be lower than the threshold used in the FPL analysis. The Westinghouse-calculated value and the value used by FPL for the gamma dose are 1.44×10^{10} rads and 1.9×10^{10} rads, respectively, at the end of the SPEO. Given that a very conservative approach was used in calculating the gamma dose at the end of the SPEO by Westinghouse, and being ~32% below the gamma dose of 1.9×10^{10} rads used in the irradiated concrete degradation evaluation as noted on page 2 of 19 of the attachment to Reference 4, performing an analytic uncertainty analysis for the gamma dose was considered unnecessary.

References:

1. Westinghouse Electric Company Document WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

2. Proceedings of the 11th International Symposium on Reactor Dosimetry, "Analysis of the VENUS-1 Benchmark using TORT and BUGLE-96," Reactor Dosimetry in the 21st Century, Brussels, Belgium, 18-23 August 2002, World Scientific Publishing Co. Pte. Ltd., 2003.

3. Westinghouse Report WCAP-15557, Rev. 0, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," September 2000.

4. 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:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

RAI 3.5.2.2.2.6-3

Additional Background

SLRA Section 3.5.2.2.2.6 (Rev. 1) explains that the relative radial neutron fluence profile used to determine the relative neutron fluence throughout the PTN shield wall was based on the results in Figure 4-2, "Neutron flux (n/cm^{2}s – normalized per source neutrons) attenuation in Portland concrete (two-loop model)," of Audit Document EPRI Report 3002002676, "Expected Condition of Reactor Cavity Concrete After 80 Years of Radiation Exposure."

Issue

It is not clear that the model used to generate the data in Figure 4-2 is relevant to PTN given that audit document EPRI Report 3002002676 explains that the model used approximates an actual reactor geometry and spatial source distribution based on "an infinite two-dimensional (2-D) cylinder with a point source at the center with a typical U-235 fission spectrum." It is not clear whether the applicant considered: (1) a detailed 3-D spatial source specification and (2) a fission spectrum specific to the more important and highly burned peripheral fuel assemblies, which are necessary to estimate an accurate fluence profile throughout the shield wall concrete due to the need to account for energy-dependent neutron transport pathways that originate at various points throughout the reactor rather than originating from a single point at the center of a geometrically simplified representation of the reactor. Furthermore, publicly available Ref. 6 cited in audit document EPRI 3002002676, simulating a more realistic reactor-shield wall configuration, indicates that the attenuation profile used by the applicant non-conservatively overestimates the actual attenuation. Justification for using that attenuation profile is not provided in the SLRA.

The NRC staff notes that no comparisons were provided between the simplified model and more detailed models in the concrete region and takes exception to the following statement in audit document EPRI Report 3002002676, which is based only on how the simplified model predicts attenuation throughout the reactor pressure vessel: "The variation between models was considered small enough that the [simplified] model can provide a reasonable spectrum for evaluation of attenuation in the concrete."

Request

Justify use of the simplified model to determine the radial neutron fluence profile throughout the PTN shield wall.

Explain the basis for not using a concrete specific to PTN as this may have a significant impact on the concrete attenuation characteristics. Concrete characteristics include not only the concrete composition based on the Miami oolite concrete used at PTN, but the amount of concrete drying that has occurred with aging (e.g., due to elevated temperatures, migration of water away from the concrete surface inward, and drying due to any other environmental conditions).

FPL Response:

With regard to justification of use of the simplified model to determine the radial neutron fluence profile throughout the PTN shield wall, FPL provides the following.

The process to establish the neutron fluence throughout the PTN shield wall was performed in two steps. First, the neutron fluence incident on the primary shield wall was calculated utilizing NRC approved Westinghouse methodology. Second, neutron attenuation through the shield wall was determined using the calculated fluence incident on the primary shield wall and the figure on page 5 of 19 of the attachment to Reference 1, which is Figure 2-3 of EPRI Report No. 3002011710.

As noted in the attachment to Reference 1, neutron fluence incident on the primary shield wall at the end of plant life was determined as follows. Westinghouse performed PTN plant specific analyses in 2009 to determine neutron fluxes incident on the primary shield wall before and after the extended power uprate (EPU) which was implemented in 2012. The flux values incident on the primary shield wall were calculated for energy groups corresponding to the energy groups provided in the original PTN design basis. The predicted EPU flux values on the primary shield wall were derived from the reactor vessel fluence evaluation performed for the EPU which satisfied the requirements set forth in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". The methodology used for the EPU fluence evaluation was approved by the NRC and is described in detail in WCAP-14040-A (Reference 2) and WCAP-16083-NP-A (Reference 3). The neutron fluxes incident on the primary shield wall were calculated based on the loading pattern of Cycle 28C that was provided for the reactor pressure vessel (RPV) neutron fluence evaluation for the Turkey Point EPU. Cycle 28C contained a core thermal power of 2644 MWt (i.e. Turkey Point Unit 3 EPU power level) and a low-leakage loading pattern with no fresh fuel on the core periphery. This takes into consideration the fission spectrum specific to the more important and highly burned peripheral fuel assemblies. The fluxes incident on the primary shield wall were calculated by taking the two-dimensional 2D planar flux solution for the radial and azimuthal (r Ө) geometry of the reactor vessel fluence evaluation with application of an axial factor of 1.288, which is based on the peak axial relative power for the cycle. Application of this axial factor converts the 2D planar flux to a 3D flux. Thus, a PTN plant specific analysis was performed to establish the neutron fluence profile incident on the primary shield wall including considerations for the PTN reactor configuration and core pattern.

More recently, Westinghouse performed additional PTN specific SLR calculations. A summary of the additional Westinghouse calculations is included in the responses to 3.5.2.2.2.6-1 and -2 (Attachments 1 and 2 to this letter). The calculations satisfy the requirements set forth in Regulatory Guide 1.190 by using the NRC approved methodology in WCAP-14040-A (Reference 3) and were used to determine the maximum neutron fluence (E > 0.1 MeV) incident on the primary shield wall at the end of the SPEO. Energy and space-dependent core power distributions as well as system operating temperatures were treated on a fuel-cycle specific basis. This calculation determined that the maximum neutron fluence (E > 0.1 MeV) on the primary shield wall was 3.24 x 10^{19} n/cm^2 at 72 EFPY, which is 9.24% below the value used in the concrete degradation evaluation of 3.57 x 10^{19} n/cm^2.

For neutron attenuation, the neutron fluence profile in the primary shield wall was determined utilizing the results from the above calculations as incident on the primary shield wall, and then applying the figure on page 5 of 19 of the attachment to Reference 1, which is Figure 2-3 of EPRI Report No. 3002011710. A comparison of this curve to Figure 1 (blue curve, neutron flux > 0.1 MeV) of Reference 6 cited in EPRI Report No. 3002002676 indicates the curves are essentially identical with the neutron flux being reduced by one order of magnitude in the first 5 inches of the primary shield wall. Thus, the attenuation profile used is appropriate and does not over estimate attenuation.

With regard to the statement taken from EPRI Report 3002002676 on variation between models, the statement was addressing variations in models determining neutron energy profiles inside and outside of the reactor vessel. As summarized above, the neutron fluence incident on the primary shield wall utilized in the PTN concrete degradation evaluation was based on a plant specific calculation. Thus this statement is not applicable to the PTN evaluation. Additionally, although the calculated fluence (E > 0.1 MeV) for PTN incident on the primary shield wall at the end of the SPEO is approximately half of the bounding number determined in EPRI report numbers 3002002676 and 3002011710, the more important parameter in determination of attenuation through the concrete is the ratio of fluence at E > 0.1 Mev to fluence at E > 1.0 MeV. For PTN, whose reactor vessel wall thickness is 7.75” at the nuclear fuel mid-plane, this ratio was determined to be 8.62 which is consistent with the relationship presented in Figure 2-1 of EPRI report number 3002011710 (same as Figure 3-3 in EPRI report number 3002002676). Therefore, the use of concrete neutron attenuation curves provided by EPRI in report numbers 3002002676 and 3002011710 is reasonable. Consideration should also be given to the conservatisms in the fluence projected to be incident on the primary shield wall as noted above, and the use of the two loop attenuation curve versus the three loop curve. Additionally, future fluence projections included a 20% positive bias on the peripheral and re-entrant corner assemblies on the selected projection cycle.

Thus, use of the simplified model to determine the radial neutron fluence profile throughout the primary shield wall is reasonable and conservative.

The basis for not using a concrete specific to PTN in determination of attenuation through the primary shield wall is that the attributes of the concrete utilized in the development of the EPRI attenuation models are comparable to and in some cases bounding of the PTN concrete attributes. Specific comparisons of these attributes including cement type, water to cement ratio, aggregate and temperature are provided in the responses to RAIs 3.5.2.2.2.6-10 and -11 in Attachment 5 to FPL letter L-2019-012 (Reference 5) and Attachment10 to this letter, respectively. Additional information supporting use of the EPRI report information in determining the neutron attenuation in the PTN primary shield wall is provided below:

1. EPRI prepared models for 2-loop and 3-loop pressurized water reactors that focused on radiation in the concrete. These were prepared in collaboration with Professor Benoit Forget from MIT. These models are described in EPRI report 3002002676. The concrete was typical of concrete constructed with Portland cement. The concrete properties were taken from PNNL-15870 entitled “Compendium of Material Composition Data for Radiation Transport Modeling”. Concrete for two reference types in PNNL-15870 were considered, Concrete, Ordinary (NIST) and Concrete, Portland. The bounding attenuation of these material types was considered. Based on the use of Portland cement with Miami oolite (limestone) aggregate at PTN, the EPRI attenuation was used in the PTN calculations.

2. The analyses of generic 2 and 3 loop biological shields shown in EPRI Report 30020011710 indicated that the attenuation was slightly more rapid in the 3-loop model, thus the 2-loop model was more conservative. The more conservative 2-loop results were used for the PTN evaluation.

3. The expected PTN water/cement ratio is relatively high in relation to higher strength concretes used to establish the attenuation curves. Additionally, the use of the steel liner at the inside diameter of the primary shield wall at PTN will assist in reducing evaporative dehydration. As stated by Maruyama (Reference 4), water loss should relate to the reduction of both heat capacity and shielding performance. Water, which is the main source of the hydrogen atom in the concrete, is considered to have a large impact on the neutron shielding performance of concrete. As such, for PTN with the relatively high water/cement ratio the concrete attenuation model used for PTN based on the EPRI model in EPRI Report 30020011710 is considered conservative.

4. There are significant conservatisms in the calculated end of SPEO life fluence (E > 0.1 MeV incident on the PTN primary shield wall.

Thus, use of the EPRI model to establish neutron attenuation in the PTN primary shield wall is reasonable and conservative.

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)

2. Westinghouse Electric Company Document WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

3. Westinghouse Electric Company Document WCAP-16083-NP-A, Revision 0, “Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,” May 2006.

4. Maruyama, I, K. Haba, O Sato, S. Ishikawa, O. Kontani, M. Takizawa “A numerical model for concrete strength change under neutron and gamma-ray irradiation”, Journal of advanced Concrete Technology, Materials, Structures and Environment. Vol. 14 (2016), pp 144-162.

5. FPL Letter L-2019-012 to NRC dated February 14, 2019, 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. MLxxxxxxxxx).

Associated SLRA Revisions:

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

Insert the following as the third paragraph of SLRA Section 3.5.2.2.2.6, page 2 of 19 of the attachment to Reference 1 as follows:

Westinghouse provided additional details regarding the fluence analysis performed for EPU indicating that the fluence values were: (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. Upon receipt of this information, and to ensure that the end of SPEO fluence value incident on the primary shield wall utilized for the concrete degradation evaluation was conservative, Westinghouse performed an additional PTN specific SLR neutron fluence calculation. This calculation satisfies the requirements set forth in Regulatory Guide 1.190 by using the NRC approved methodology in WCAP-14040-A to determine the maximum neutron fluence (E > 0.1 MeV) incident on the primary shield wall at the end of the SPEO. This calculation determined that the maximum neutron fluence (E > 0.1 MeV) on the primary shield wall was 3.24 x 10^{19} n/cm^2 at 72 EFPY. Thus, the fluence value (E > 0.1 MeV) of 3.57 x 10^{19} n/cm^2 incident on the primary shield wall used for the concrete degradation evaluation is conservative.

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

RAI 3.5.2.2.2.6-4

Additional Background

The applicant provided its calculation for reactor vessel support displacements per atom (dpa) 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:

Issue

a. The audit document describes the dpa calculation. The reference cited for the calculation is from a textbook. Reviewing the textbook equations used: (1) the total integrated neutron flux term in Equation 12 of the textbook was not used, and (2) the average neutron energy term in Equation 13 of the textbook was not used (other values were used instead). It is not clear that the terms, as defined by the equations in the textbook being used as the reference defining the method by which the dpa is calculated, are being used.

b. The accuracy and precision of the model used to determine dpa is not clear because it 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 uncertainty was considered.

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 PTN peak E > 0.1 MeV and E > 1 MeV neutron fluxes incident on the PTN shield wall and are used as inputs to the dpa rate calculation method.

Request

a. Provide justification of the method used, or 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. Validate 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. 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.

FPL Response:

The following lettered items respond to the comparable lettered requests above, and refer to the additional calculations performed by Westinghouse described in the response to RAI 3.5.2.2.2.6-1 in Attachment 1 to this letter:

a. Regarding the RV support displacements per atom (dpa) calculation presented in Audit Document FPLCORP020-REPT-130, Rev. 1, Appendix E, "Irradiated Reactor Vessel Supports Evaluation," pages E-5 and E-6 of E-9, the calculation was simplified based on using the total neutron flux (n/cm^2-sec) for two energy groups, E = 0.1MeV and E = 1.0 MeV, as representative of the total flux incident on the primary shield wall for a best-estimate evaluation. This was due to the limited neutron energy distribution available for the primary shield wall. The input to the neutron fluence calculation described in SLRA Section 3.5.2.2.6, Rev. 1, reported PTN primary shield wall total flux for E > 1.0 MeV and for 5.53 keV < E ≤ 1.0 MeV energy groups, both design basis and post EPU. The pre and post EPU neutron flux values for these average energies were used to separately estimate (calculate) a corresponding dpa rate using equations 10 and 11 from the dpa rate method reference. These dpa rates were converted to dpa for both average energies before and after the EPU, then totaled. This simplification is considered reasonable as total flux and average energies were used. Furthermore, the total flux used for the average energy groups was determined to be 8 centimeters into, rather than incident on, the primary shield wall. However, as described in response to RAI 3.5.2.2.2.6-1 (Attachment 1 to this letter), the fluence determined from the reported total flux values is conservative. The same is true of the dpa determined from those same total flux values to the two energy groups, as clarified below. In addition, the total flux for the entire 5.53 keV < E ≤ 1.0 MeV energy group was applied to the 0.1MeV energy. Lastly, the weighted average of neutron energies > 1.0 MeV is expected to be close to 1 MeV.

To further support the conservatism of the dpa calculation used as an input to the reactor vessel support embrittlement evaluation, information from NUREG/CR-5320 (Reference 1) was reviewed. Section 7.5 of NUREG/CR-5320 presents the results of what are characterized as extensive sophisticated calculations of neutron flux and dpa rate for the Turkey Point reactor vessel supports. On page 147 of Reference 1, Table 7.2 provides 32 EFPY dpa projections (E > 0.1 MeV) for various locations/data points of the Turkey Point reactor vessel supports. The data point in the table comparable to the top of the active fuel region is data point number four which has a projected 32 EFPY dpa value of 2.30×10^{-4}. To project this dpa value to the end of the SPEO, a ratio of 72 EFPY/32 EFPY was applied. Then, an additional 75% was added for conservatism to bound the effects of the EPU and the positive bias associated with the peripheral and re-entrant corner assemblies as described in b. below. This resulted in a dpa value of 9.06×10^{-4} ($2.30 \times 10^{-4} \times 72/32 \times 1.75$). This projected dpa is significantly below the dpa value of 4.89×10^{-3} (page 16 of 19 of the attachment to Reference 2), used in the reactor vessel support embrittlement analysis. Thus, the simplified determination is reasonable for the best estimate evaluation of the RPV support members and does not require correction.

b. The model used to determine dpa was not validated by comparison to an appropriate benchmark or standard. Estimates were used in the evaluation consistent with NUREG-2192 Section 3.5.2.2.2.6. However, to provide reasonable assurance that the estimated dpa is conservative, a separate evaluation was consulted relative to accuracy and uncertainties as described below:

   i. Westinghouse has calculated the neutron-induced iron dpa (E > 0.1 MeV) to be used in the RPV supports critical flaw size analysis. The dpa (E > 0.1 MeV) was calculated at two locations: (1) the RPV outer surface and (2) the air-side surface of (incident on) the primary shield wall liner. An investigation of the RPV exposure results summarized in SLRA Section 4.2 indicated that for the traditional beltline region (RPV region directly surrounding the height of the active fuel) Turkey Point

Units 3 and 4 gave very similar results. Based on this similarity, the use of either Turkey Point Unit 3 or Unit 4 in the analysis would provide results applicable to both units at the maximum neutron fluence (E > 0.1 MeV) location on the primary shield wall. Accordingly, the dpa (E > 0.1 MeV) calculation used the neutron fluence results for Turkey Point Unit 4 and this result was applied to both units. The dpa calculation utilized the fluence rate synthesis technique (Reference 3) and the dpa cross sections from ASTM Standard E693-94 (Reference 4) which uses ENDF/B-IV-based iron dpa cross sections. The newer standard, ASTM E693-17 (Reference 5), uses ENDF/B-VI dpa cross sections. A review of Table 2 in ASTM E693-17 shows that there is up to 4% difference in spectrum-averaged dpa cross sections for the Arkansas Nuclear ONE-1 (ANO) cavity and for the H.B. Robinson Unit 2 RPV at the 3⁄4T vessel wall. The % difference comparison was calculated using the formula ([Current - Old]/Old), where "Old" uses the ASTM Standard E693-94 and "Current" uses the ASTM Standard E693-17 recommended iron dpa cross sections. In order to adjust the iron dpa (E > 0.1 MeV) results based on the ASTM Standard E693-17 recommendation, the iron dpa data (E > 0.1 MeV) were increased by 4%.

The Westinghouse-calculated dpa (E > 0.1 MeV) contains the same conservatism described in RAI 3.5.2.2.2.6-1 for the neutron fluence (E > 0.1 MeV) calculation at the air-side surface of (incident on) the primary shield wall liner, where future projections included a 20% positive bias on the peripheral and re-entrant corner assemblies on the projection fuel cycle to reach the end of the SPEO. Furthermore, the dpa (E > 0.1 MeV) value was conservatively reported at the core midplane elevation to be used for the RPV supports, as the available Turkey Point reactor models created for the RPV analysis that are described in Section 4.2 of the SLRA, and also used in the dpa analysis, did not include the RPV support structures.

An analytic uncertainty for the ex-vessel neutron dosimetry (EVND) has been quantified as 12% in Reference 6. This uncertainty is associated with calculations for fast neutron fluence (E > 1.0 MeV) and can be used for the concrete biological shield for this energy range. A comparison of the EVND location uncertainties associated with best-estimate neutron fluences from least squares evaluations for the H. B. Robinson benchmark is 7% for the fast neutron fluence rate (E > 1.0 MeV) and 11% for dpa per second (dpa/s) (Reference 6). The difference in uncertainties between the neutron fluence rate (E > 1.0 MeV) and dpa/s from least squares evaluations indicates that the analytic uncertainties for dpa/s could be higher compared to those from E > 1.0 MeV. However, the neutron fluence at the end of SPEO was calculated conservatively by Westinghouse by using a 20% positive bias on the peripheral and re-entrant corner assemblies on the projection fuel cycle and using a conservative location (i.e. core midplane elevation) for the RPV supports. Therefore, these conservatisms are expected to cover the additional analytic uncertainties in the dpa/s (E > 0.1 MeV) calculation.

The results indicate that the dpa at the core midplane incident on the primary shield wall is 1.05×10^{-2} dpa (E > 0.1 MeV). Applying the 40% value (see further discussion in items a. and c.) for comparison to the dpa value in Reference 2 (4.89×10^{-3}), the value becomes 4.20×10^{-3}, approximately 16% lower. Thus, the dpa value used in the RV support embrittlement evaluation is reasonable and conservative and application of a bias or uncertainty is not warranted. The calculation summary for this calculation is available on the ePortal.

ii. No additional uncertainties need to be applied to the dpa calculation. Based on the additional calculations performed by Westinghouse as summarized in the response to RAI 3.5.2.2.2.6-1 (Attachment 1 to this letter) and i. above, and comparison to the values presented in NUREG/CR-5320, there is sufficient conservatism in the dpa value of 4.89×10^{-3} (page 16 of 19 of the attachment to Reference 2) used in the reactor vessel support embrittlement analysis to account for uncertainties.

c. The figure on page 13 of 19 of the attachment to Reference 2 presents the expected neutron flux (E > 1 MeV n/(cm^2 sec)) variation relative to the active fuel region. Additional figures presented in References 7 and 8 show the total neutron flux normalized to the maximum flux within the belt-line region. This information indicates that the normalized neutron flux at the top of the active fuel region is approximately 40% of the maximum neutron flux at the belt-line region which is consistent with the figure in Reference 2.

Additional references (References 7 and 8) were reviewed to verify that the use of 0.4 for the axial peaking factor is bounding (or sufficiently representative). As noted in the response to a. above, Section 7.5 of NUREG/CR-5320 (Reference 1) presents the results of what are characterized as extensive sophisticated calculations of neutron flux and dpa rate for the Turkey Point reactor vessel supports. On page 147 of Reference 1, Table 7.2 provides 32 EFPY dpa projections (E > 0.1 MeV) for various locations of the Turkey Point reactor vessel supports. The data point number in the table comparable to the top of the active fuel region is data point number four which has a projected 32 EFPY dpa value of 2.30×10^{-4}. To project this dpa value to the end of the SPEO, a ratio of 72 EFPY/32 EFPY) was applied. Then, an additional 75% was added for conservatism to bound the effects of the EPU and the positive bias associated with the peripheral and re-entrant corner assemblies as described in b. above. This resulted in a dpa value of 9.06×10^{-4} ($2.30 \times 10^{-4} \times 72/32 \times 1.75$). This projected dpa is significantly below the dpa value of 4.89×10^{-3} (page 16 of 19 of the attachment to Reference 2) used in the reactor vessel support embrittlement analysis. Thus, the use of 0.4 for the axial peaking factor is bounding of past actual and future expected axial peaking factors corresponding to the most influential peripheral fuel assemblies for 80 years of operation.

References:

1. NUREG/CR-5320, Impact of Radiation Embrittlement on Integrity of Pressure Vessel Supports for Two PWR Plants

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. Westinghouse Electric Company Document WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

4. ASTM Designation E693, 1994, “Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E706(ID)” ASTM, West Conshohocken, PA, 1994, www.astm.org.

5. ASTM Designation E693, 2017, “Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA),” ASTM International, West Conshohocken, PA, 2017, DOI: 10.1520/E0693-17, www.astm.org.

6. Westinghouse Report WCAP-15557, Rev. 0, “Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology,” September 2000.

7. Remec, I., “Radiation Environment in Biological Shields of Nuclear Power Plants”, Oak Ridge National Laboratory (ORNL), March 22, 2013.

8. TransWare Enterprises (TwE) Report No. TWE-LPI1-001-R-001, Rev. 0 “An Evaluation of Neutron, Gamma, and Temperature Profiles in a Three Loop PWR Biological Shield”, February, 2013.

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

Section II – Decrease in Fracture Toughness of RPV Supports

Background

In SLRA Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation," (Rev. 1) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18283A308), the applicant describes evaluations performed to determine decrease in fracture toughness due to the effects of neutron irradiation on the reactor vessel structural steel supports. The applicant opted to follow the methodology in NUREG-1509 to assess the aging effects due to neutron embrittlement of the reactor vessel (RV) supports. RAIs 3.5.2.2.2.6-5 through 3.5.2.2.2.6-9 address the reduction in toughness of the steel RV supports.

Regulatory Basis

10 CFR 54.21 requires SLR applicants to perform an integrated plant assessment. For PTN, 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 concrete bioshield wall, but have exposed beams protruding from this wall, including saddles that support the RV inlet and outlet nozzles.

RAI 3.5.2.2.2.6-5

Issue

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.

However, the applicant did not articulate what information was used to reach that conclusion. The staff needs additional information to evaluate the adequacy of the applicant's assertion that no further evaluation is required per its citation of NUREG-1509.

Request

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.

FPL Response:

The RV support bolting material was evaluated following the same process as that for the beam material summarized in updated SLRA Section 3.5.2.2.2.6 (Attachment to Reference 1). The evaluation was performed in accordance with the transition temperature approach indicated in Table 4-4 of NUREG-1509 and considered design stresses, the existing condition of the supports (see Attachment 8 to this letter), and a comparison of the estimated end of life NDT to the normal operating temperature.

The bolting for the PTN RV supports are composed of ASTM A354, Grade BC material. ASTM A354, Grade BC material composition is primarily carbon, manganese, and chrome with no specified range for copper. As noted in Reference 1, certified mill test reports (CMTRs) for the bolting were reviewed to confirm the material composition and properties. Since this bolting is not ASME Section III, per NUREG-1509, an estimated NDT should be obtained from Table 4-1 as identified in Item (3)b of "Explanatory Notes for Figures 4-1 Through 4-3". The material is included in the "Quenched & Tempered" classification in Table 4-2 of NUREG-1509. However, A-354 is not listed in Table 4-1 under "Quenched & Tempered". As a result, a numerical average of the six NDT+2σ values listed in Table 4-1 under the "Quenched & Tempered" classification was taken. Utilizing the numerical average was considered reasonable and conservative based on a review of NUREG-0577 (Ref. 2). In NUREG-0577, primary component support materials were evaluated for the potential of low fracture toughness and lamellar tearing. As part of that evaluation, quenched and tempered bolting materials, including ASTM A354, were assessed relative to brittle fracture characteristics. NUREG-0577 indicated that because these materials were of high strength, and contained well-tempered martensitic microstructures, they would not be expected to show an abrupt ductile-brittle transition. For this reason, NUREG-0577 concluded that the quenched and tempered bolting materials required no further evaluation regarding brittle fracture characteristics. Note that item (3)b allows for the estimation of initial NDT from Table 4-1. The numerical average resulted in an initial NDT+2σ value of -15ºF for the bolting material [40+(-10)+(-20)+20+(-60)+(-60)=-90, -90/6=-15]. Additionally, based on the location of the bolting and its material content, the projected neutron fluence and calculated displacements per atom (dpa) will the same as that identified for the cantilever and cross beams in Reference 1, or 1.43×10^{19} n/cm^2 and 4.89×10^{-3}, respectively.

Thus, utilizing the fitted curve in Figure 3-1 of NUREG-1509 for 4.89×10^{-3} dpa, the ΔNDT would be the same as that identified for the beam material in Reference 1 of 70 ºC, or 126ºF. As noted in Reference 1, the fitted curve was utilized because the test data points associated with the dpa being evaluated (in the 1 to 5×10^{-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 end-of-life NDT for the bolting material would be ~111ºF which is below the normal operating temperature (Modes 1 through 4) in the reactor cavity of ~120ºF. This provides an NDT margin of 9ºF at the end of the SPEO. When considering the actual normal operating temperature of the bolting which is estimated to be 150 ºF, the NDT margin at the end of the SPEO would be 39 ºF. The actual ΔNDT for the bolting material would be expected to be much less than the beam material based on the information presented in NUREG-0577 and the fact that the initial NDT is based on NDT plus 2 standard deviations.

Finally, based on the SLR stress evaluation of the RV supports with implementation of auxiliary line leak-before-break (see reference documents on the portal), the critical component stress interaction ratio (% allowable) of the bolting material is ~8% versus ~23% for the cantilever beams. Thus, based on the approximated initial bolting NDT and expected ΔNDT, and the fact that the cantilever beams are more limiting from a stress standpoint, the embrittlement evaluation of the cantilever beams bounds the evaluation of the bolting material.

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 (ML18283A308).

2. NUREG-0577, Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports.

Associated SLRA Revisions:

SLRA Section 3.5.2.2.2.6 (attachment to Reference 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 16 of 19 of Reference 1 as follows:

Based on review of NUREG-1509 and the design documentation of the PTN RV support bolting, ~~no further~~ **the** evaluation for reduction in fracture toughness for the bolting ~~is~~ **will be bounded by that of the beam material** ~~required~~.

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

RAI 3.5.2.2.2.6-6

<u>Additional Background</u>

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.

<u>Issue</u>

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").

<u>Request</u>

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.

FPL Response:

Additional technical basis for using the fitted curve versus the upper bound correlation curve in Figure 3-1 of NUREG-1509 is as follows:

- With the exception of A350LF3 and A105 (two of the materials exposed to both neutron and gamma), the rest of the data presented in Figure 3-1 of NUREG-1509 is based on A212B material. NUREG-0577 (Ref. 1) was prepared to address a number of reactor coolant system support materials with regard to their susceptibility to brittle fracture. Per NUREG-0577 (Ref. 1), A212B material was classified as "Group I" due to its coarser grained microstructure and is considered to have the highest susceptibility to brittle failure. ASTM A-588 as-hot rolled material was classified as "Group II, with medium susceptibility to brittle failure, but the fine grained normalized form would be classified "Group III", having the least susceptibility to brittle failure, based on the classification of evaluated plants having normalized A-588 materials. As indicated in SLRA 3.5.2.2.2.6 Rev. 1, the PTN cantilever beams are normalized ASTM A-588 B material. Thus, the data in Figure 3-1 of NUREG-1509 tends to over predict the ∆NDT for the A-588 B normalized fine grained material. With regard to the PTN RV support bolting (ASTM A-354), NUREG-0577 indicated that because these materials contained well-tempered martensitic microstructures, they would not be expected to show an abrupt ductile-brittle transition.

- All of the data points on Figure 3-1 of NUREG-1509 that are above the fitted curve between 1 x 10^{-4} and 6 x 10^{-3} are from high-flux isotope reactor (HFIR) testing with materials exposed to both neutron and gamma radiation. As noted on page 19 of NUREG-1509, the measured gamma flux in HFIR was reported as 36.4 Gy/sec, or 131,040 Gy/hr. In comparison the gamma flux value at the RV supports for PTN is approximately 130 Gy/hr. NUREG-1509 goes on to state that the gamma flux in cavities of operating reactors is much less than that in HFIR, and should not induce a significant increase in embrittlement (i.e., ΔNDT) of the RV supports. Since no attempt was made to establish the effect of this significantly higher gamma flux on the measured ΔNDT, use of these data points in the curve fit tends to cause an over prediction of the ΔNDT. By removing these data points, the shape of the bounding curve would closely match that of the fitted curve in the region of between 1 x 10^{-4} and 6 x 10^{-3} dpa.

- To support the conservatism of the dpa calculation used as an input to the reactor vessel support embrittlement evaluation, information from NUREG/CR-5320 (Reference 2) was reviewed. Section 7.5 of NUREG/CR-5320 presents the results of what are characterized as extensive sophisticated calculations of neutron flux and dpa rate for the Turkey Point reactor vessel supports. On page 147 of NUREG/CR-5320, Table 7.2 provides 32 EFPY dpa projections (E > 0.1 MeV) for various locations of the Turkey Point reactor vessel supports. The data point number in the table comparable to the top of the active fuel region is data point four which has a projected 32 EFPY dpa value of 2.30 x 10^{-4}. Applying a ratio to project the dpa to the end of the SPEO (72 EFPY/32 EFPY), and then an additional 75% for conservatism to bound the effects of the EPU and the positive bias associated with the peripheral and re-entrant corner assemblies, results in a dpa value of 9.06 x 10^{-4} (2.30 x 10^{-4} x 72/32 x 1.75). This projected dpa is significantly below the dpa value of

4.89×10^{-3} (page 16 of 19 of the attachment to Reference 3) used in the reactor vessel support embrittlement analysis (see response to RAI 3.5.2.2.2.6-4 in Attachment 7 to this letter for further discussion of dpa). This would more than compensate for the uncertainty in the estimate of ΔNDT.

Based on the above, use of the fitted curve provides a reasonable estimate of the ΔNDT of the PTN RV support beam material at the end of the SPEO.

References:

3. NUREG-0577, “Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports,” NRC, October 1979 (Draft).

4. NUREG/CR-5320, “Impact of Radiation Embrittlement on Integrity of Pressure Vessel Supports for Two PWR Plants,” Oak Ridge National Laboratory, January 1989.

5. 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 (ML18283A308).

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

RAI 3.5.2.2.2.6-7

Additional Background

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.

Issue

It is not clear how the margin component of the equation was considered or calculated for this step.

Request

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.

FPL Response:

The first paragraph on page 17 of 19, of the attachment to Reference 1 identified the normal operating temperature (Modes 1 through 4) in the reactor cavity as ~120°F. To clarify, this is the normal air temperature in the reactor cavity and not the actual operating temperature of the RV support members. This value was used for conservatism. When addressing margins, the actual normal operating temperature of the components should be considered. Per Westinghouse calculations, the normal operating temperature of the cantilever beams and bolting is 150°F. The margin curve presented in Appendix R, Figure R-1200-1 of Reference 18 of NUREG-1509 is based on material thickness. For the cantilever beams, the limiting location is the upper flange with a thickness of 2.47” which requires a margin of 30°F. For the bolting material, whose diameter is 2.25”, the required margin is also 30°F.

Using the end-of-life NDT for the cantilever beam of 99°F from the first paragraph on page 17 of 19 of the attachment to Reference 1 (value based on the fitted curve from Figure 3-1 of NUREG-1509, see discussion provided in the response to RAI 3.5.2.2.2.6-6, Attachment 6 to this letter), the margin equation is satisfied as follows:

$$99°F + 30°F \leq 150°F$$

$$129°F \leq 150°F$$

Using the end-of-life NDT for the bolting material of 111°F from Attachment 5 to this letter (see note above on use of fitted curve from Figure 3-1 of NUREG-1509), the margin equation is also satisfied as follows:

$$111^\circ\text{F} + 30^\circ\text{F} \leq 150^\circ\text{F}$$

$$141^\circ\text{F} \leq 150^\circ\text{F}$$

Thus, the margin equation is satisfied and no additional subsequent actions are required per the flowchart in Fig. 4-4 of NUREG-1509.

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 (attachment to Reference 1) is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions.

Revise the text at the top of page 17 of 19 as follows:

The end-of-life NDT**s** for the beam material **and bolting are** ~~would be~~ ~99°F **and 111°F, respectively,** which ~~is~~ **are both** below ~~the~~ **their** normal operating temperature ~~(Modes 1 through 4) in the reactor cavity~~ of ~~~120°F~~**150°F**. This ~~provides an~~ **satisfies the *Margin* term described in Figure 4-4 of NUREG-1509 of 30°F for the beam and 30°F for the bolting with additional margin available** ~~NDT margin of 21°F~~ at the end of the SPEO. **Thus, no subsequent actions are required.** ~~Also note that the initial NDT is based on mean plus 1.3 standard deviation. This could provide a further margin of 23ºF based on the actual material properties from the CMTRs.~~

Additional conservatisms associated with stress analysis of RV support steel are described below.

(1) Per the CMTRs for the beams, the yield strength of the beam is reported as 58.16 ksi, which is about 20% larger than 48.75 ksi (used in the determination of the maximum IR of the RV support steel).

~~(2) (The yield strength of 48.75 ksi was calculated based on the operating temperature of 150 ºF, which is higher temperature than the calculated ambient temperature of 120 ºF. Lower temperature results in more capacity for the RPV support steel.~~

(2) Based on the span depth ratio of the cantilever portion of the RPV support steel, the beam is considered as a deep beam where the shear is typically governing. Per the stress analysis, the shear capacity of the beam was calculated by considering only the web area. The beam is reinforced by using 1” thick stiffener plates. Thus, the

shear stress flows not only in the web but also in the stiffeners and the top and bottom flanges of the beam. The effective area of the beam for the shear should be larger than the web-only area.

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

RAI 3.5.2.2.2.6-8

Additional Background

Section 4.3.1.1 of NUREG-1509 states:

> Physical examination of the RPV supports is essential to the reevaluation. As mentioned previously, the purpose of the examination is to detect visible signs of degradation of the supports, including, but not limited to, rust, corrosion, cracks or permanent deformation of the members.

Figure 4-2 of NUREG-1509 identifies “evaluate existing physical condition” as one of the key inputs to the “preliminary evaluation” prior to performing the transition temperature approach described in Appendix E. The visual inspections described in Appendix E “have not identified dimensional shifts or changes in the RV support steel,” but there is no mention of rust, corrosion, cracks, or permanent deformation of the members as cited in NUREG-1509.

Issue

The SLRA does not describe the visual inspections described in Appendix E of NUREG-1509.

Request

Describe the examinations that have been performed to assess degradation of the RV supports due to rust, corrosion, and cracks to provide a justifiable basis for the analysis.

FPL Response:

ASME Class I, 2 and 3 structural supports and associated bolting are managed for loss of material, loss of preload, and cracking (high strength Class 1 bolting only) by the PTN ASME Section XI, Subsection IWF AMP as described in SLRA Table 3.5-1, items 068, 081, 087, 089, and 091. The aging management review of the PTN Class 1 RV supports for SLR was performed consistent with NUREG-2191, Section III, Table B1.1, Structures and Component Supports, Class 1, and NUREG-2192, Table 3.5-1, and presented in Tables 3.5-1 and 3.5.2-1 of the SLRA. Thus, as presented in NUREG-2191 and -2192 and in the PTN SLRA, visual inspections performed under ASME Section XI, Subsection IWF are suitable for detecting rust and corrosion. The beam attachment bolting material used for the PTN RV supports is ASTM A-354, Grade BC. Actual yield and tensile stresses based on the Certified Material Test Reports (CMTRs) for this PTN bolting material are both less than 150 ksi. Thus the bolting material is not considered high-strength, and cracking of the bolting is not an aging effect requiring management for the RV supports. As such, SLRA Table 3.5-1, item 068 is not applicable to the ASME Class 1 RV supports. Additionally, the support beams that makeup the RV supports are ASTM A-588 Type B, more commonly known as Cor-Ten steel. Cor-Ten steel is categorized as a “weathering” steel. The material is a corrosion resistant steel, that left uncoated develops an outer layer of patina. This patina protects the steel from additional corrosion.

Based on the above, the aging effects requiring management for the PTN RV supports include loss of material and loss of preload (bolting material). Cracking was not identified as an aging effect requiring management for the PTN RV supports. The programs credited for managing these aging effects are the PTN ASME Section XI, Subsection IWF and Boric Acid Corrosion AMPs. See FPL response to RAI No. 3.5.2.2.2.6-9 in Attachment 9 to this letter for the proposed initial (baseline) inspection for cracking to address condition evaluation identified in NUREG-1509.

As noted in SLR Section B.2.3.32, the PTN ASME Section XI, Subsection IWF AMP is consistent with the requirements of NUREG-2191, XI.S3. XI.S3 indicates that parameters monitored or inspected include corrosion; cracking, deformation; misalignment of supports; missing, detached, or loosened support items; general structural condition of weld joints and weld connections to building structure for loss of integrity.

Additionally, XI.S3 indicates for detection of aging effects that the VT-3 examination method specified by the program can reveal loss of material due to corrosion and wear, cracks, verification of clearances, settings, physical displacements or loss of integrity at bolted connections.

Based on the above, the PTN ASME Section XI, Subsection IWF and Boric Acid Corrosion AMPs are considered adequate to manage the aging effects of loss of material and loss of preload of the PTN RV supports.

Examinations performed to date on the PTN RV supports as part of the current PTN ASME Section XI, Subsection IWF Inservice Inspection Program consist of VT-3 visual inspections. These inspections are summarized below with the specifics provided on the portal:

Unit 3

VT-3 inspections of accessible portions of the Unit 3 PTN RV supports were performed on March 2, 2003 and November 2, 2004 utilizing photographic equipment on extension poles. These inspections were performed from the lower cavity area below the reactor vessel on all accessible areas to the extent possible. On March 14, 2012, based on ALARA and lessons learned, all accessible areas to the extent possible were examined on the Unit 3 RV supports through the reactor coolant piping penetrations in the primary shield wall using a fiber optic video probe. The VT-3 inspection data sheet from the 2012 inspection indicated acceptable results meeting the acceptance criteria of IWF-3410, and did not identify any areas requiring further evaluation.

The acceptance criteria specified in IWF-3410 is as follows:

"*(a) Component support conditions which are unacceptable for continued service shall include:*

- *(1) Deformations or structural degradations of fasteners, springs, clamps, or other items;*
- *(2) Missing detached, or loosened support items;*
- *(3) Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance machined or sliding surfaces;*
- *(4) Improper hot or cold settings of spring supports and constant load supports;*
- *(5) Misalignment of supports;*
- *(6) Improper clearances of guides and stops.*

(b) Except as noted in IWF-3410(a), the following are examples of non-relevant conditions:

- *(1) Fabrication marks (e.g., from punching, layout, bending, rolling, and machining);*
- *(2) Chipped or discolored paint;*
- *(3) Weld spatter on other than close tolerance machined or sliding surfaces;*
- *(4) Scratches and surface abrasion marks;*
- *(5) Roughness or general corrosion which does not reduce the load bearing capacity of the support;*
- *(6) General conditions acceptable by the material Design, and/or Construction Specifications."*

Unit 4

VT-3 inspections of accessible portions of the Unit 4 PTN RV supports were performed on March 25, 2002 and October 29, 2003 utilizing photographic equipment on extension poles on all accessible areas to the extent possible. These inspections were performed from the lower cavity area below the reactor vessel. On December 8, 2012, based on ALARA and lessons learned, all accessible areas to the extent possible were examined on the Unit 4 RV supports through the reactor coolant piping penetrations in the primary shield wall using a fiber optic video probe. The VT-3 inspection data sheet from the 2012 inspection indicated acceptable results meeting the acceptance criteria of IWF-3410 (see above), and did not identify any areas requiring further evaluation.

None of the inspections performed would invalidate the SLR evaluation summarized in SLRA Section 3.5.2.2.2.6 of the PTN RV supports. Details of the above inspections, including inspection reports and photographs, are available on the ePortal.

References:

None

Associated SLRA Revisions:

See response to RAI 3.5.2.2.2.6-9, Attachment 9 to this letter

Associated Enclosures:

None

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RAI 3.5.2.2.2.6-9

As stated in the regulatory basis above, 10 CFR 54.21(1) 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 aging management review (AMR).

Issue

The applicant did not address RPV supports in an irradiated environment in its AMR tables.

Request

For RPV supports in an irradiated environment, 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. If not, justify how the requirements of 10 CFR 54.21 to perform an integrated plant assessment are being met.

FPL Response:

SLRA Table 2.4.1-1 in Section 2.4.1.2 lists “ASME Class 1, 2, and 3 supports” and “Structural bolting: ASME Class 1, 2, and 3 supports” as component types that require an aging management review. The RV supports and associated bolting are included as part of these component types as they are ASME Class 1 supports. SLRA Section 3.5.2.1.1, Table 3.5-1, and Table 3.5.2-1 address the aging management review for the containment structures and internal structural components. NUREG-2192, Table 3.5-1 line item numbers identified in SLRA Table 3.5-1 applicable to the RV supports include 081, 087, 089, and 091. AMR items in SLRA Table 3.5.2-1 applicable to the RV supports include two entries for ASME Class 1, 2, and 3 supports on page 3.5-88, and two entries for Structural bolting: ASME Class 1, 2, and 3 supports on page 3.5-97. The beam attachment bolting material used for the PTN RV supports is ASTM A-354, Grade BC. Actual yield and tensile stresses based on the Certified Material Test Reports (CMTRs) for this bolting material are both less than 150 ksi. Thus the bolting material is not considered high-strength, and cracking of the bolting is not an aging effect requiring management for the RV supports. As such, SLRA Table 3.5-1, item 068 is not applicable. As noted in the response to RAI 3.5.2.2.2.6-8, (Attachment 8 to this letter), aging effects requiring management for the RV supports include loss of material and loss of preload. These aging effects are managed by the PTN ASME Section XI, Subsection IWF and Boric Acid Corrosion AMPs. Although there is not a line item in NUREG-2192, Table 3.5-1 specifically for loss of fracture toughness of steel RV supports due to a neutron environment, SLRA Tables 3.5-1 and 3.5.2-1 are revised to address this AMR item.

As noted in the response to RAI 3.5.2.2.2.6-8, (Attachment 8 to this letter), VT-3 visual inspections of the RV supports to date as part of the current PTN ASME Section XI, Subsection IWF Inservice Inspection Program have not identified unacceptable degradation or loss of material associated with the supports, or provided any indication that the performance of the supports has been adversely affected by their location in a radiation environment. More frequent visual inspections of the RV supports than what are currently performed would provide additional assurance that the supports would continue to perform their component intended function throughout the SPEO. Accordingly, FPL commits to performing a baseline VT-3 visual inspection of the RV supports (6 supports per unit) as part of the PTN ASME Section XI, Subsection IWF AMP during the last scheduled refueling outage prior to entry into the SPEO for each unit. This will establish the general condition of the RV supports and their readiness for the SPEO. Subsequently, the same inspections of the RV supports will be performed on each unit on a five year frequency (more often than is currently performed) during the SPEO as part of the PTN ASME Section XI, Subsection IWF AMP. The acceptance criteria and corrective actions for the RV support inspections will be consistent with the requirements of IWF. The SLRA is revised as noted below including adding AMR items to Table 3.5.2-1 for a radiation environment.

Generic Safety Issue 15 (GSI-15) Considerations in NUREG-0933

Resolution of Generic Safety Issue 15 (GSI-15), “Radiation Effects on Reactor Pressure Vessel Supports,” in 1996, as reported in NUREG-0933 states in part:

> “The preliminary conclusion indicated that the potential problem did not pose an immediate threat to public safety. The tentative results indicated that plant safety could be maintained despite reactor vessel support structures (RVSS) radiation damage. In order to encompass the uncertainties in the various analyses and provide an overall conservative assessment, several structural analyses conducted demonstrated the following:
>
> (1) Postulating that one of the four RPV supports was broken in a typical PWR, the remaining supports would carry the reactor vessel and the load even under safe-shutdown earthquake (SSE) seismic loads;
>
> (2) If all supports were assumed to be totally removed (i.e., broken), the short span of piping between the vessel and the shield wall would support the load of the vessel.”

In summary, based on the proposed RV support inspection plan, and the GSI-15 considerations above, there is reasonable assurance that the Turkey Point Units 3 and 4 reactor vessel support steel will perform the license renewal intended function during the SPEO.

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, Table 3.5-1, Table 3.5.2-1, Section 17.2.2.32, Table 17-3 Item 36, and Section B.2.3.32 are 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 by inserting the following paragraph after the fourth paragraph on page 17 of 19 of the attachment to Reference 1 as follows:

The requirements of ASME Section XI, Subsection IWF are enhanced to include a VT-3 visual inspection of the RV supports (6 per unit) as part of the PTN ASME Section XI, Subsection IWF AMP prior to or during the last scheduled refueling outage prior to entry into the SPEO for each unit. Subsequently, the same inspections of the RV supports will be performed on each unit on a five year frequency during the SPEO as part of the PTN ASME Section XI, Subsection IWF AMP to provide reasonable assurance that the RV supports will continue to perform their intended function.

Revise SLRA Table 3.5 -1 on pages 3.5-71 and 3.5-84 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, 068	High-strength steel structural bolting	Cracking due to SCC	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 cracking for high strength steel bolting exposed to an uncontrolled indoor air environment. **RV support bolting is not high strength, so this item is not applicable to RV supports.**
3.5-1, 097	Group 4: Concrete (reactor cavity area proximate to the reactor vessel): reactor (primary/ biological) shield wall; sacrificial shield wall; reactor vessel support/pedestal structure	Reduction of strength; loss of mechanical properties due to irradiation (i.e., radiation interactions with material and radiation-induced heating)	Plant-specific AMP	Yes (SRP-SLR Section 3.5.2.2.2.6)	A plant-specific AMP is not required for SLR **for the reactor cavity concrete and embedded steel. For the reactor vessel supports in the reactor cavity, AMP XI.S3, "ASME Section XI, Subsection IWF" with enhancement is proposed in lieu of a plant specific AMP**. Further evaluation is documented in Section 3.5.2.2.2.6.

Revise SLRA Table 3.5.2-1 on pages 3.5-88, 3.5-97, and 3.5-100 as follows:

Table 3.5.2-1: Containment Structure and Internal 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 RV supports	Structural support	Carbon Steel	Neutron Flux	Loss of fracture toughness	ASME Section XI, Subsection IWF	III.A4.T-35	3.5-1, 097	E, 2
Structural bolting: ASME Class 1 RV supports	Structural support	Carbon Steel	Neutron Flux	Loss of fracture toughness	ASME Section XI, Subsection IWF	III.A4.T-35	3.5-1, 097	E, 2

E. Consistent with NUREG-2191 environment and aging effect but a different AMP is credited or NUREG-2191 identifies a plant-specific AMP. Although carbon steel is not included in NUREG-2191, Item III.A4.T-35, and NUREG-2192, Table 3.5-1, Item 097, reactor vessel support /pedestal structure is listed.

Plant-Specific Notes for Table 3.5.2-1

1. Note B applies to the Structures Monitoring AMP only.
2. ASME Section XI, Subsection IWF with enhancement is proposed in lieu of a plant specific AMP. Further evaluation of the RV supports for loss of fracture toughness is included in SLRA Section 3.5.2.2.2.6.

Revise SLRA Section 17.2.2.32 on page A-37 as follows:

The requirements of ASME Section XI, Subsection IWF are supplemented to include volumetric examination of high-strength bolting for cracking and 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.

The requirements of ASME Section XI, Subsection IWF are also enhanced to include a VT-3 visual inspection of the RV supports (6 supports per unit) as part of the PTN ASME Section XI, Subsection IWF AMP prior to or during the last scheduled refueling outage prior to entry into the SPEO for each unit. Subsequently, the same inspections will be performed on each unit on a five year frequency during the SPEO as part of the PTN ASME Section XI, Subsection IWF AMP.

Revise SLRA Table 17-3 Item 36 on page A-104 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
36	ASME Section XI, Subsection IWF (17.2.2.32)	XI.S3	i) **Perform a VT-3 visual inspection of the RV supports (6 supports per unit) as part of the PTN ASME Section XI, Subsection IWF AMP prior to or during the last scheduled refueling outage prior to entry into the SPEO for each unit. Subsequently, the same inspections will be performed on each unit on a five year frequency during the SPEO as part of the PTN ASME Section XI, Subsection IWF AMP.**	At 5 years prior to the SPEO, start one-time inspections. Complete pre-SPEO inspections 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

Revise SLRA Section B.2.3.32 on page B-243 as follows:

Element Affected	Enhancement
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 Table IWB-2500-1, Examination Category B-G-1, at least once per interval for a sample of high-strength bolting selected to provide reasonable assurance that SCC is not occurring for the entire population of high-strength bolts. **Include a VT-3 visual inspection of the RV supports (6 supports per unit) as part of the PTN ASME Section XI, Subsection IWF AMP prior to or during the last scheduled as refueling outage prior to entry into the SPEO for each unit. Subsequently, the same inspections will be performed on a five year frequency as part of the PTN ASME Section XI, Subsection IWF AMP.**

Associated Enclosures:

None

NRC RAI Letter Nos. ML19032A396 and ML19032A397 Dated February 1, 2019

RAI 3.5.2.2.2.6-11

Regulatory Basis

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 subsequent period of extended operation. 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 Aging Management Review (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.

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 reactor vessel 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 subsequent period of extended operation or if plant-specific operating experience (OE) of concrete irradiation degradation exists that may impact intended functions.

Background:

The SRP-SLR states that data related to the effects and significance of gamma radiation on concrete mechanical and physical properties is limited, especially for conditions (e.g., dose, temperature) representative of light-water reactor (LWR) plants. The SRP-SLR also states that based on literature review of existing research, a radiation limit of 1×10^{8} Gy (1×10^{10} rad) gamma dose is considered a conservative radiation exposure level beyond which concrete material properties may begin to degrade markedly.

SLRA Section 3.5.2.2.2.6, as supplemented by letter dated October 5, 2018, states that Figure 7 of Hilsdorf (1978) paper “presented the change in compressive strength versus gamma dose for a limited amount of data." The SLRA also states that "[t]he mean interpolated value of the trend of this data would indicate a decrease in compressive strength for a dose between 2.0 x 10^{10} rads to 3.0 x 10^{10} rads. However, the data that was used to derive the plot is varied and not considered as fully representative of commercial reactor conditions." The SLRA further states the following:

> The Maruyama (2017) paper (Reference 1) suggested that either the threshold reference value for gamma exposure be raised to a high level or abandoned entirely. With consideration of the prior Hilsdorf data and the available test data presented by Maruyama, the test data indicates gamma irradiation up to and beyond a threshold of 2.3x10^{10} rads has no effect on material properties.
>
> Based on the above discussion, and considering the 80 year gamma dose incident on the primary shield wall at PTN is 1.9 x 10^{10} rads, there will be no degradation of the primary shield concrete at PTN due gamma radiation.

<u>Issue</u>:

The staff noted that the SLRA references Maruyama's (2017) study, which uses a gamma dose threshold of 2.3x10^{10} rads, as the basis to conclude that there will be no degradation of the PTN PSW concrete due to gamma radiation since PTN's gamma dose is 1.9 x 10^{10} rads. It is not clear to the staff how Maruyama (2017) findings are a justified approach for screening out the effects of gamma radiation on PTN's PSW concrete when the SRP-SLR threshold for damage is 1.0 x 10^{10} rads. In its review of the referenced study, the staff noted the following:

- **<u>gamma dose rate</u>:** the staff noted uncertainties on the applicability of test results relevant to gamma dose radiation of concrete in nuclear power plants (NPPs). For example, in Maruyama (2017) studies, concrete specimens were exposed to gamma ray dose rates 2-20 times greater (1.25 to 10 kGy/h) than expected at concrete components near a PWR reactor vessel (approx. 500 Gy/h). In that regard, Murayama's paper (2017) states that "it is impossible, in principle, to determine whether the obtained data can be applied to commercial reactors without assessing the effects of dose-rate." Also, in its conclusion summary of results for gamma-ray impact on concrete Maruyama (2017) states that "the findings reported in this work must be validated to ensure their reproducibility. […] After validation, the reference values for gamma rays should be abandoned."

- **carbonation of concrete:** Murayama's paper (2017) notes that "[o]nly when concrete is carbonated under irradiation will its strength increase," and that "[t]here was almost no difference between gamma ray-irradiated and heat dried specimens exposed to conditions under which carbonation typically proceeds," noting the need for "[a]dditional gamma-ray irradiation tests on concrete without carbonation." The paper then concludes that "[w]hen supplemental drying tests under non-CO_2 conditions were performed […] strength of those specimens quickly fell as the mass reduction rate increased, faster than the strengths of gamma-ray-irradiated and heat dried specimens." It is not clear to staff whether the applicant has taken into consideration the impact of carbonation, heating and drying on the results in Maruyama's paper (2017) and if and how that relates to the conditions on the concrete at PTN PSW concrete. Absent conditions suitable for carbonation of the concrete it is not clear how the applicant concludes that no loss of strength is expected due to gamma irradiation.

- **temperature:** The staff also noted that the test temperature of the Maruyama study specimens was lower (10 to 30 degrees Celsius) than the operating temperature for the concrete at PTN's PSW (approximately 49 degrees Celsius). The staff notes that at higher temperatures more degradation due to irradiation is expected due to an increase in thermal stresses of concrete.

- **cement type and w/c ratio:** The staff noted that PTN's concrete is composed of ASTM C-150-64 Florida Type II cement with a w/c ratio of 0.59, as reported in the SLRA and Supplement FPLCORP020-REPT-130, Revision 1, while Maruyama's gamma radiation tested concrete specimens used early high strength Type I cements with a much lower w/c ratio of 0.50. It is not clear if and how the gamma radiation induced aging effects (e.g.., radiolysis) of the Maruyama tested concrete specimens with a lower w/c ratio compare to the cast concrete of higher w/c ratio at PTN's PSW.

Based on both apparent dissimilarities between PTN concrete and that used in the Maruyama study and lack of consideration of some factors, it is not clear to staff if and how the Maruyama (2017) study test results for gamma dose aging effects on concrete are relatable and applicable to PTN PSW concrete. The staff needs additional information to justify the applicant's assumption that there are no aging effects on concrete due to interactions of gamma rays with cement paste and aggregate used in PTN concrete during the SPEO.

<u>Request</u>:

With regard to considerations such as the gamma dose rate, carbonation of concrete, aggregate, cement type, w/c ratio, and operating temperature of the concrete at PTN PSW versus the test specimens used in the Maruyama study: explain how the conclusions in Maruyama's paper can be used to assume that there is no degradation in material properties due to gamma dose, or provide justification for why such comparison between the PTN PSW and Maruyama study is unnecessary.

FPL Response:

As noted in PTN SLRA 3.5.2.2.2.6, Rev. 1 (page 12 of 19 of the attachment to Reference 1), gamma irradiation up to 2.3 x 10^{10} rad likely has no effect on material properties of concrete. Additionally, NUREG/CR-7171 (Reference 2), Section 5.2 used 2.0 x 10^{10} rad (i.e., 2.0 x 10^{5} kGy) as the critical level for gamma exposure when assessing the reduction of concrete strength under nuclear irradiation (as shown in Figure 5.7 excerpted and reprinted below from Reference 2). Concrete structures were considered to be sound as long as the level of radiation did not exceed this critical level.

This figure presents a plot, prepared by Kontani et al. (2010), of the same test data collected by Hilsdorf et al. (1978), showing the effect of gamma rays on the residual compressive and tensile strengths of concrete. In the figure, the reference dose is shown as 2.0 x 10^{10} rad.

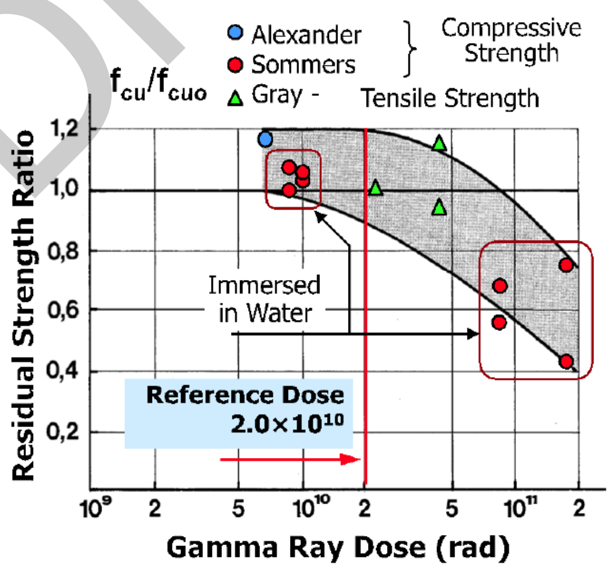

Figure 5.7. Effect of gamma radiation on the residual compressive strength of concrete. (Kontani et al. 2010)

<u>[Excerpted from NUREG/CR-7171]</u>

EPRI report number 3002011710 (Reference 3) drew a similar conclusion indicating that for primary shield wall (PSW) evaluations a value of 2.3 x 10^{10} rad should be utilized as the gamma radiation threshold. This was based on the experimental results in the Maruyama paper (2017, Reference 4).

Additionally, as noted in the response to RAI 3.5.2.2.2.6-2 (Attachment 2 to this letter), the Westinghouse-calculated gamma dose incident on the PTN PSW at the end of the SPEO is 1.44 x 10^{10} rad. The dose calculated by FPL for the gamma dose used in the concrete degradation evaluation was 1.9 × 10^{10} rad incident on the PSW at the end of the SPEO. Considering the very conservative gamma dose used in the concrete degradation evaluation, and the 2 x 10^{10} rad critical level from NUREG/CR-7171, concrete degradation due to gamma radiation of the PTN PSW is not expected through the SPEO.

In addition, the considerations (i.e., gamma dose rate, carbonation of concrete, temperature, cement type, and w/c ratio) for the PTN PSW in relation to the Maruyama paper are discussed below.

- **<u>Gamma dose rate</u>**

  Concrete specimens were exposed to gamma ray dose rates of 1.25 to 10 kGy/h for 32 months, which is 2-20 times greater than the expected gamma dose rate at concrete components near a PWR reactor vessel (approximately 500 Gy/h). This was to achieve an integrated gamma dose of 2.0 x 10^{10} rad within the 32 months (for instance, 10 kGy/h x 23,040 hr > 2.0 x 10^{10} rad). Regarding the gamma dose rate, Maruyama provides the related explanation/consideration of their experiments in Section 3.1 of the paper (Reference 4) as shown below.

  "In principle, it is necessary to examine gamma-ray of various energy levels, but it was difficult to perform irradiation tests using monochromatic gamma radiation at different energy levels; thus, it was decided to perform gamma-ray irradiation tests using cobalt-60 (60Co, two gamma rays with energies of 1.17 and 1.33 MeV are emitted). In addition, it is impossible, in principle, to determine whether the obtained data can be applied to commercial reactors without assessing the effects of dose-rate (It was unfeasible to do so in the neutron irradiation tests). Therefore, ***it was decided to run gamma-ray irradiation tests for different dose-rates in this investigation****. The quantities of energy deposited by radiation when it hydrolyzed the water in cement paste was evaluated using alanine dosimeters."*

Regarding the compressive strength with different gamma dose rates (from 1.25 to 10 kGy/h), Figures 68(a) and 69(a) of Reference 4 provide test results, and Section 3.3.2 states that:

"In terms of compressive strength, the sealed Con-A specimens had nearly identical values throughout the 32-mo. testing period, whereas a slight increase in strength was observed for the sealed Con-B specimens (Fig. 30). A tendency for the strength of gamma-ray-irradiated specimens to increase the longer they were irradiated was observed for both Con-A and Con-B specimens."

Per Figures 68(a) and 69(a), concrete strength is consistently maintained or slightly increased for each gamma dose rate in the test. Overall, a similar tendency (slope) of the compressive strength response is observed for each gamma dose rate in the test. Section 3.3.1 states that "***total water production showed little dependency on irradiation dose-rate***", which means that the quantities of energy deposited by radiation (measured by the hydrolyzation of the water) is barely dependent on irradiation dose-rate.

Therefore, the effect of the gamma dose rate was considered insignificant in the PTN PSW evaluation based on the tendencies described in the Maruyama paper relative to minimal dependency on irradiation dose rate and the compressive strength to remain approximately the same with longer exposure, with the much lower dose rate and integrated dose for the PTN PSW.

Even though it is recommended to increase the reference value for gamma rays or abandon entirely with further verification in the Maruyama paper (2017), a caveat is also provided as stated below:

"Theoretically, gamma radiation causes metamictization in rock-forming minerals, but the required doses for metamictization differ greatly from the dose released during the lifetimes of nuclear power plants. Therefore, ***the reference value for gamma rays [i.e., 2.0 x 10^{10} rad] can be raised to a very large value, or even abandoned entirely. Before implementing this recommendation, however, it is necessary to verify the reproducibility of our experimental results.*** *"*

- **Carbonation of concrete**

Section 3.4 of the Maruyama paper (Reference 4) states that gamma radiation preserved the layered structure of the calcium silicate hydrate (C-S-H) and enhanced strength in cement paste under conditions where they caused carbonation. As a result, concrete strength is maintained or increased due to the carbonation effect. The carbonation effect on the properties of concrete due to the gamma-ray radiation is equivalent to that of heating and drying effects such that they offset each other.

Section 6.2 of the paper states that:

*"**Based on our experimental research, gamma rays do not cause concrete degradation**. In fact, concrete exposed to gamma radiation increases in strength. In addition, the physical properties of hcp [i.e., hardened cement paste] was increased, and those of aggregates did not change dramatically; rather**,** the observed changes were small enough that the properties of concrete showed no degradation."* and that

*"In this study**, it was confirmed that neutron fluence causes strength degradation in concrete. This is mainly due to aggregate expansion caused by the metamictization of rock-forming minerals**."*

The gamma radiation effect on concrete strength degradation is relatively small compared to the neutron radiation effect. Also, the calculated PTN gamma dose is less than the reference dose (critical level for gamma exposure when assessing the reduction of concrete strength under nuclear irradiation) of 2.0 x 10^{10} rad. Therefore, the carbonation effect was not considered in the PTN PSW evaluation since the carbonation effect on concrete strength due to the calculated PTN gamma radiation was found to be insignificant.

- **Temperature**

The temperature range of 10 to 30 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. In the Maruyama paper (2017), hydrogen generation tests were performed for different temperatures and gamma dose rates. Concrete specimens were pre-dried and tested with different temperatures (25, 40, and 60°C). The hydrogen generation rates were observed to peak immediately after irradiation onset but tapered off thereafter. Section 3.3.1 of the paper states that *"H_2 generation rates were almost completely unaffected by specimen temperatures."* which means that the temperature range used in Reference 4 didn't affect the concrete.

In a reinforced concrete design, the typical temperature limit is considered 150°F (If the normal operation temperature is less than that, a further analysis on temperature effect is not needed). Section E.4.1 of ACI 349 (Reference 5) states that a temperature of 150°F (or less) is allowed without considering loss of significant compressive strength of the concrete. For local areas, the allowable temperature is increased up to 200°F. The PTN operating temperatures are 49 to 65.6°C (i.e., 120 to 150°F in the reactor cavity and at the RPV supports, respectively). Considering the allowable temperature of 200°F (for local areas) and the Maruyama test results (2017), the expected PTN operating temperature is unlikely to affect the compressive strength of the concrete.

- **Cement type**
  As provided in PTN SLRA 3.5.2.2.2.6 and 3.5.2.2.2.6 Rev. 1 (Attachment to Reference 1), PTN uses Type II cement, which is for general purpose with moderate sulfate resistance. In the Maruyama paper (2017), mainly two types of experiments were performed with respect to the gamma irradiated concrete: (1) interaction tests, which focused on radiolysis within cement paste & (2) physical property tests, which focused on examining the properties of exposed concrete. Among these, the physical property tests are to examine the strength reduction and material property changes of the concrete exposed to gamma radiation. The cement used in the physical property tests of the Maruyama (2017) paper (Reference 4) is Type III (high-early-strength cement). Note that high-early-strength cement is known as Type III, not Type I as indicated in the issue above. Per ASTM C-150 (Reference 6), 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 6), 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 (Reference 7), Table 6.3.4(a), typical concrete compressive strength corresponding to a 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 1, 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 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 (2017) paper represents typical concrete compressive strength and was considered as comparable to the w/c ratio estimated for PTN.

As provided in the above, the gamma radiation and considerations (i.e., gamma dose rate, carbonation of concrete, temperature, cement type, and w/c ratio) for the PTN PSW in relation to the Maruyama paper were examined. It was determined that the considerations in the Maruyama paper are either compatible to or envelope those used in the PTN PSW. However, given the disclaimers in the Maruyama paper and lack of definitive applicability of the paper to PTN PSW concrete, SLRA Section 3.5.2.2.2.6, Rev. 1 (Attachment to Reference 1) concluded that *"Based on the above, a plant-specific program to manage the effects of concrete irradiation on its strength and mechanical properties is **not expected** to be necessary to ensure the components perform their intended function consistent with the CLB through the subsequent period of extended operation. However, as the potential for irradiation-related degradation cannot be fully eliminated, FPL will continue to follow the on-going industry efforts, such as through EPRI, that are clarifying the effects of irradiation on concrete and corresponding aging management recommendations as noted in Commitment Number 53 in Table 17-3, and will:*

a) ensure their applicability to the PTN Unit 3 and Unit 4 primary shield wall and associated reactor vessel supports;
b) update design calculations, as appropriate; and
c) *develop an informed plant-specific program, if needed."*

SLRA Section 3.5.2.2.2.6, Rev. 1, is further amended below to clarify the pertinent statements (on pages 12, 17, and 18 of 19 in the Attachment to Reference 1) that although the degradation of the concrete due to gamma irradiation is unlikely but is not screened out it will be managed consistent with on-going industry efforts on the topic.

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)
2. 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.
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. ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349-13) and Commentary", Printed June 2014.
6. ASTM C-150-07, "Standard Specification for Portland Cement".
7. ACI 211.1-91, "Standard Practice for Selecting Proportions for Normal, Heavyweight, and Mass Concrete", Reapproved 2002.

Associated SLRA Revisions:

SLRA Section 3.5.2.2.2.6, Rev. 1 (Attachment to Reference 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 12 of 19 (Revise the 1st and 2nd paragraph and add new paragraph between the 1st and 2nd paragraphs) of Reference 1 as follows:

The Maruyama paper suggested that either the threshold reference value for gamma exposure be raised to a high level or abandoned entirely **with further verification**. With consideration of the prior Hilsdorf data and the available test data presented by Maruyama **and as described in NUREG/CR-7171**, the test data indicates gamma irradiation up to ~~and beyond a threshold of~~ 2.3 x 10^{10} Rads **likely** has no effect on material properties.

The considerations, such as gamma dose rate, carbonation of concrete, temperature, cement type, and w/c ratio in the Maruyama paper were examined in relation to the corresponding considerations for the PTN PSW. It was determined that the considerations in the Maruyama paper are either compatible to or envelope those used in the PTN PSW evaluation.

Based on the above discussion, and considering the 80 year gamma dose incident on the primary shield wall at PTN is 1.9 x 10^{10} Rads, there will **likely** be no **or minimal** degradation of the primary shield concrete at PTN due gamma radiation. However, FPL will continue to follow EPRI and industry efforts to better define the effects of gamma radiation on concrete, and will update this evaluation and implement a**n informed** plant specific AMP**, consistent with industry finding relative to gamma irradiation,** if necessary as noted in Commitment Number 53 in Table 17-3 **to provide reasonable assurance that the PSW will perform its intended function through the subsequent period of extended operation**.

Revise SLRA Section 3.5.2.2.2.6, Rev. 1, Pages 17 and 18 of 19 of the Attachment to Reference 1 as follows:

Based on the above, a plant-specific program to manage the effects of concrete irradiation **on its strength and mechanical properties** is not expected to be necessary to ensure the components perform their intended function consistent with the CLB through the subsequent period of extended operation. However**, as the potential for irradiation-related degradation cannot be fully eliminated,** FPL will continue to follow the on-going industry efforts**, such as through EPRI,** that are clarifying the effects of irradiation ~~of~~ **on** concrete and corresponding aging management recommendations as noted in Commitment Number 53 in Table 17-3, and will:

a) ensure their applicability to the PTN Unit 3 and Unit 4 primary shield wall and associated reactor vessel supports;

b) update design calculations, as appropriate; and

c) develop an informed plant-specific program, if needed.

Associated Enclosures:

None