UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 1, 2019

Mr. Mano Nazaar
President, Nuclear Division and Chief Nuclear Officer
700 Universe Blvd
EX/JB
Juno Beach, FL 33408

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4 - REPORT FOR THE IRRADIATED CONCRETE AUDIT REGARDING THE SUBSEQUENT LICENSE RENEWAL APPLICATION REVIEW (EPID NO. L-2018-RNW-0002)

Dear Mr. Nazar:

By letters dated January 30, 2018, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18037A812), February 9, 2018 (ADAMS Accession No. ML18044A653), February 16, 2018 (ADAMS Accession No. ML18053A123), March 1, 2018 (ADAMS Accession No. ML18072A224), and April 10, 2018 (ADAMS Accession Nos. ML18102A521 and ML18113A132), Florida Power & Light Company (FPL) submitted an application for subsequent license renewal of Renewed Facility Operating License Nos. DPR-31 and DPR-41 for the Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point) to the U.S. Nuclear Regulatory Commission (NRC) pursuant to Section 103 of the Atomic Energy Act of 1954, as amended, and part 54 of title 10 of the *Code of Federal Regulations*, "Requirements for renewal of operating licenses for nuclear power plants."

The staff of the U.S. Nuclear Regulatory Commission (NRC) completed its Irradiated Concrete regulatory audit from July 16, 2018 through October 17, 2018, in accordance with the Irradiated Concrete regulatory audit plan (ADAMS Accession No. ML18173A087). The audit report is enclosed.

If you have any questions, please contact me by e-mail at Bill.Rogers@nrc.gov.

Sincerely,

/RA/

Bill Rogers, Project Manager
License Renewal Project Branch
Division of Materials and License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosure:
Audit Report

cc w/encl: Listserv

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4 - REPORT FOR THE IRRADIATED CONCRETE REGULATORY AUDIT REGARDING THE SUBSEQUENT LICENSE RENEWAL APPLICATION REVIEW (EPID NO. L-2018-RNW-0002) DATED: FEBRUARY 01, 2019

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ADAMS Accession Nos. ML19032A536

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DATE	02/01/2019	12/03/18	01/25/2019	12/01/2019

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Audit Report

Irradiated Concrete Regulatory Audit Regarding the Turkey Point Nuclear Generating Units 3 and 4, Subsequent License Renewal Application

July 16, 2018 – October 17, 2018

Division of Materials and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Enclosure

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION, DIVISION OF LICENSE RENEWAL

Docket Nos:	50-250 and 50-251
License No:	DPR-31 and DPR-41
Licensee:	Florida Power & Light Company
Facility:	Turkey Point Nuclear Generating Units 3 and 4
Locations:	Homestead, Florida and Rockville, Maryland
Dates:	June 16 – October 17, 2018
Reviewers:	B. Rogers, Project Manager, Division of Materials and License Renewal (DMLR) A. Buford, Structural Engineer, Division of Engineering (DE) S. Cuadrado de Jesus, Structural Engineer, DE C. Fairbanks, Senior Materials Engineer, DMLR A. Hiser, Senior Level Advisor, DMLR A. Prinaris, Structural Engineer, DE A. Patel, Nuclear Engineer, Division of Safety Systems (DSS)
Approved By:	Eric Oesterle, Chief License Renewal Projects Branch Division of Materials and License Renewal Brian Wittick, Chief Structural Engineering Branch Division of Engineering Robert Lukes, Chief Nuclear Performance and Code Review Branch Division of Safety Systems

Report for the Irradiated Concrete Regulatory Audit Regarding the Turkey Point Nuclear Generating Units 3 and 4, Subsequent License Renewal Application

1. Introduction

The U.S. Nuclear Regulatory Commission (NRC or the staff) conducted an audit of Florida Power & Light Company's (FPL's or the applicant's) methodology and results for the evaluation of the effects of irradiation on concrete and steel structural elements located within containment and management of the effects of aging on the structures, in support of the staff's review of the Turkey Point Nuclear Plant Units 3 and 4 (PTN) Subsequent License Renewal Application (SLRA).

The regulatory basis license renewal requirements are specified in Title 10 of the Code of Federal Regulations (10 CFR), Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." 10 CFR 54.17, "Filing of Application," requires applicants for renewed licenses to send written correspondence to the NRC. The 10 CFR 54.37, "Additional Records and Record Keeping Requirements," requires that license renewal applicants maintain documents demonstrating compliance with the requirements of 10 CFR Part 54 in auditable and retrievable form. During review of an SLRA, there may be supporting information retained as records under 10 CFR 54.37 that, although may not necessarily be required to be submitted as part of the SLRA, provide additional information and technical bases for the submitted information that would facilitate staff's review, and therefore the staff may determine an audit is necessary. Staff guidance is provided in NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), dated July 2017, and in NUREG-2191, "Generic Aging Lessons Learned (GALL) Report for Subsequent License Renewal," dated July 2017.

The purpose of the audit was for the staff to verify in accordance with 10 CFR 54.21, that the applicant has demonstrated the effects of irradiation on concrete and structural steel, for containment internal structures, will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation (SPEO).

The scope of this audit was to examine the applicant's supporting documentation for its disposition of the "Further Evaluation" provided in SLRA Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation." Staff guidance in SRP-SLR Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation states":

> 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 E > 0.1 MeV) or gamma radiation exceeds the respective threshold level during the subsequent period of extended operation or if plant-specific OE 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

SLRA Section 3.5.2.2.2.6 discusses evaluations and references published papers in support of: (1) its determination of projected fluence to 80 years of operation, and (2) its conclusion that a plant-specific aging management program (AMP) is unnecessary to manage reduction in strength and mechanical properties due to irradiation of concrete. The applicant determined that containment internal concrete structures at Turkey Point would be capable of performing their intended functions through the SPEO without being managed for this aging effect.

2. Audit Activities

The initial audit activities occurred at the applicant's facility near Homestead, Florida during the week of July 16-20, 2018 and at the applicant's facility in Rockville, Maryland, during September 18-20, 2018. Further staff review of applicant documents occurred at the applicant's facility in Rockville, Maryland from September 20 to October 17, 2018. The audit activities consisted of the following:

- Applicant presentations and presentation and discussions with applicant representatives to obtain additional clarification related to the disposition of the irradiation effects on containment internal concrete and steel structures.
- Examination of the applicant's program basis documents and related references for the AMPs.
- Review of the supporting calculations and evaluations.
- Assessment of additional information necessary to be submitted to the NRC (docketed) to support the completion of the staff's review.

2.1 **Areas of Review**

The staff reviewed the SLRA, supporting documentation, CLB information, and presentation material, and had extensive discussions with the applicant concerning: the methodologies used, the results obtained, and the conclusions drawn by the applicant. The staff's review centered around the following areas:

- Applicant methodology and process, including justification of the use of referenced publications applied to the evaluation and how the specifics of the references are used in calculations/analyses.

- Review of plant-specific calculations/analyses for irradiation aging effects, specifically: Determination of neutron fluence and gamma radiation, thermal heating and their combined effects at the face of concrete structures and their attenuation within the reactor pressure vessel (RPV) supporting concrete and bio-shield

- How the CLB design loading combinations and acceptance criteria are considered (design codes used), including design basis events such as loss of coolant accident (LOCA) and safe shutdown earthquake (SSE)

- Consideration or disposition of any potential effects of changes in loading scenarios associated with refueling outages when water and fuel loads may change

- Consideration or disposition of serviceability conditions such as deformations or deflections and the potential effect of resulting irradiation and thermal heat effects on containment building structures (CBS)/structural support functions

- Rationale for assumptions related to the effects of neutron and gamma radiation, and thermal effects on concrete strength and modulus of elasticity (including concrete damage and depth of damage, exposed and embedded steel damage and capacity to carry load, steel anchor bolt capacity, reduction in bond strength; or bounding case if applicable), considering plant-specific fluence, gamma, and thermal loading estimates

- How the structural configuration and detailing (dimensions, distances, anchorage of RPV and nozzles to supports, materials used, placement of reinforcement and embedments) of Turkey Point Units 3 and 4 internal concrete structures is considered for both fluence assessments and structural capacity (also local design and section checks where necessary)

2.1 Applicant Presentations

The applicant's presentations, made at the audit during the week of the July 16, 2018, indicated that upon the receipt of the NRC audit plan, dated July 5, 2018, the applicant had identified additional actions that were required and resulted in the following:

- The CLB structural load analysis for the primary shield wall and reactor vessel supports were reviewed and determined to have several loading omissions. The identified omissions resulted in corrective action to include additional loads from reactor vessel head replacement and re-performance of the CLB structural load analysis for use in the evaluation of effects of radiation on concrete and steel structural elements within containment. The CLB structural load analysis was re-performed and made available in a final draft form for the staff's review during the week of July 16, 2018.

- The original SLRA Section 3.5.2.2.2.6 indicated that the applicant had concluded that the first 17.5 inches of primary shield wall concrete (measuring outward from the face nearest the reactor vessel) was not required for the primary shield wall to perform its intended function of structural support and, therefore, any impacts from radiation limited to affecting the portion within the first 17.5 inches did not need to be evaluated. However, during an audit presentation the applicant indicated that it was in the process of developing a new position. The applicant indicated that Section 3.5.2.2.2.6 of the SLRA would be revised to address the new position, however, the revision was still being developed and was not yet available for the staff's review.

- As determined by the staff's review of SLRA Section 3.5.2.2.2.6, the applicant confirmed that it had not considered the effect of radiation on the exposed structural steel that, along with the concrete portions of the primary shield wall, provide structural support to the RPV nozzles (and the reactor vessel). The applicant indicated that Section 3.5.2.2.2.6 of the SLRA would be revised to also address an evaluation of the structural steel, however, the revision was still being developed and was not yet available for the staff's review.

- The basis for the applicant's conclusion on radiation fluence information at the primary shield wall was based on plant specific environmental qualification (EQ) information. The applicant indicated that it was revising its method for determining fluence, however, that re-evaluation was still being developed and was not yet available for the staff's review.

The applicant's presentations, made at the applicant's Rockville, Maryland, facility during the September 18-20, 2018, portion of the audit, provided additional information on the applicant's efforts to determine the effects of irradiation on concrete and structural steel within containment and included the information on revised draft documents. In addition, the applicant indicated that it had altered it's methodology for determining fluence levels from the initial use of the 40-year EQ document package to information describing estimated neutron and fluence values based on Turkey Point's PTN's 2009 extended power uprate (EPU).

2.2 Revised Applicant Documents

During the period from September 20, 2018, to October 17, 2018, the applicant provided the following documents for the staff's review at the applicant's facility located in Rockville, Maryland.

- Revised CLB calculation, which included the identified loading omissions.

- Revised Irradiated Concrete "130 Report" based on the final and approved CLB calculation and including:
  - Updated Appendix D – irradiated concrete evaluation
  - Added Appendix E – structural steel evaluation
  - Added Appendix F– Westinghouse fluence information

- SLRA supplement dated October 5, 2018 (ML 18283A308) modifying Section 3.5.2.2.2.6, which modified the applicant's methods of evaluating irradiated concrete and structural steel.

2.3 Staff's Review of the Revised Documents, Discussions with the Applicant, and Observations

2.3.1 Fluence

The staff reviewed the applicant's neutron and gamma fluence analysis consistent with the acceptance criteria provided in SRP-SLR, Section 3.5.2.2.2.6, which included a detailed review of the neutron and gamma fluence/dose analysis methodology.

The staff identified several areas where additional information would be required to support the development of the staff's finding.

- The staff reviewed the basis for SLRA Section 3.5.2.2.2.6 (Rev. 1), which 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 pgs. G-7 and G-10 of G-11, the peak fluence determined by the applicant is based on values reported by Westinghouse in Audit Document Westinghouse Letter FPL-09-41, "Turkey Point Units 3 and 4 - Extended Power Uprate (EPU)," Response to Shaw Request for Radiological Information, February 2009. These values are: (1) based on an azimuthally averaged value instead of the peak azimuthal value and (2) reported at a location 8 centimeters (cm) into the shield wall concrete instead of at the surface. The staff noted that additional information may be required for a justification for using the azimuthally averaged value 8 cm into the shield wall concrete instead of the peak surface fluence value given that the stated intent of SLRA Section 3.5.2.2.2.6 (Rev. 1) is to determine maximum fluence values incident on the shield.

- Concerning the use of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence," Section 1.4, "Methodology Qualification and Uncertainty Estimates" (ADAMS No. ML010890301), the staff noted that additional information may be required in order to establish the accuracy of the fluence estimates supporting SLRA Section 3.5.2.2.2.6 as follows:

  - For the validation of 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): the inclusion of comparisons with applicable measurement and calculational benchmarks and the inclusion of additional margin for uncertainty as appropriate if no applicable measurement or calculational benchmarks are available.

  - For the analytic uncertainty that has not been quantified for the peak 80 year fluence values provided: the inclusion of analytic uncertainty estimates for the reported fluence values, including all relevant sources of uncertainty, to demonstrate the accuracy of the methodology.

- 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^2s – 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." The staff determined that it was not clear that the model used to generate the data in Figure 4-2 is relevant to PTN given that 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." As a result, the staff noted that the following information may be required:

  - The results of the applicant's use of (1) a detailed 3-D spatial source specification and (2) a fission spectrum specific to the more important and highly burned peripheral fuel assemblies is 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. In addition, the results of the applicant's consideration of the publicly available Ref. 6 cited in Audit Document EPRI 3002002676, which simulates a more realistic reactor-shield wall configuration, and indicates that the attenuation profile used by the applicant may non-conservatively overestimate the actual attenuation.

  - Information to justify and qualify use of the simplified model to determine the radial neutron fluence profile throughout the PTN shield wall.

  - Consideration of how <u>not</u> using a concrete specific to PTN is justified 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).

- Regarding the reactor vessel support displacements per atom (dpa) calculation as reviewed in Audit Document FPLCORP020-REPT-130, Rev. 1, Appendix E, "Irradiated Reactor Vessel Supports Evaluation," pgs. E-5 and E-6 of E-9, supporting SLRA Section 3.5.2.2.2.6, the staff noted that the following information may be required:
  - Verification that the calculation of the dpa rate was performed consistently with the method chosen.
  - Validation of the model used to determine dpa 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)") with consideration of dpa uncertainty
  - 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. Considering this information, the staff noted that additional information may be required to 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 PTN for 80 years of operation.

2.3.2 Reduction of Strength and Loss of Mechanical Properties of Concrete and Steel Due to Irradiation

The staff reviewed SLRA AMR Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation," and supporting documentation using the guidance provided in SRP-SLR, Section 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation."

During its audit, the staff interviewed the applicant's staff and reviewed onsite documentation provided by the applicant. During the audit and after further review of Audit Documents (also made available in the PTN portal) the staff made the following observations:

- FPLCORP020-REPT-130, Revision 1, references a Maruyama, et al, 2017 paper titled "Development of Soundness Assessment Procedure for Concrete Members affected by Neutron and Gamma-Ray Irradiation," (Maruyama's 2017 paper) as a basis to determine the decrease in concrete compressive strength due to neutron fluence at the primary shield wall concrete (PSW). In its review, the staff noted that the applicant specifically references Figure 54, "Comparison of observed strength ratio (Fc/Fc_0) and total neutron fluence in preceding research and the present study," of Maruyama's 2017 paper and concludes that the reduction of concrete compressive strength at the PSW due to neutron fluence is 10 percent. The staff also noted that Figure 54 shows data from a variety of concretes with different aggregates, water/cement ratios (w/c), and test temperatures that were bounded by a "lower boundary curve." Considering the variability in the data of Figure 54 and absent a clarification by the applicant on how the concrete at PTN PSW compares to the data in Figure 54, it is not clear how the applicant selected a value for Fc/Fc_0 of 0.9 (i.e., a 10 percent reduction in concrete compressive strength as a measure of concrete degradation) that is above the "lower boundary curve" value of approximately 0.8 (i.e., a 20 percent reduction in concrete compressive strength) for a neutron fluence of 3.57×10^{19} n/cm^2 in Figure 54 and the lower bound curve value of 0.5 (50 percent) shown in Figure 3 of the docketed SLRA letter L-2018-187 dated October 5, 2018.

- FPLCORP020-REPT-130 Revision 1 references Maruyama's 2017 paper as the basis to conclude that there will be no degradation of the PSW concrete due to gamma radiation. The staff noted that in the studies by Maruyama, the concrete specimens were exposed to a gamma ray dose rate that may be 2-20 times greater than the dose rate that is expected at concrete components near a PWR reactor vessel. The staff also noted that the test temperature of the Maruyama study specimens is lower (10-30 degrees Celsius) than the operating temperature for the concrete at PTN's PSW (approximately 49 degrees Celsius). In addition, 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 FPLCORP020-REPT-130, Revision 1, while Maruyama's gamma radiation tested concrete specimens used high early strength Type I cements with a much lower w/c. Based on these factors it is not clear how the Maruyama's test results for gamma dose aging effects on concrete are relatable and applicable to Turkey Point's PSW concrete.

- Appendix D, "Irradiated Reactor Shield Wall Evaluation," of FPLCORP020-REPT-130, Revision 1, provides the structural analysis for the PSW concrete structure under the governing CLB load case while considering the effects of loss of compressive strength of concrete due to neutron and gamma radiation and radiation induced volumetric expansion (RIVE). In the analysis, the applicant calculates the maximum interaction ratio (IR) for the irradiated concrete under the CLB governing load case (D (dead load) + L (live load) + T (thermal load) + new LOCA (pipe rupture loads)). The maximum IRs provide a measure of the structural components' required capacity vs the available capacity under the CLB loads and as such are a measure of the margin available in the capacity of the structures.

  The staff may request that the applicant provide a description and justification of the assumed governing load case, the respective maximum horizontal and vertical loads on the PSW, and the resulting maximum IRs for the irradiated PSW concrete structure to assess the margin in capacity available under all stress conditions for the PSW concrete structure for the subsequent period of extended operation.

- Appendix E, "Irradiated Reactor Vessel Support Evaluation," of FPLCORP020-REPT-130, Revision 1, provides an evaluation of the RPV supports for the aging effect of reduction of fracture toughness due to irradiation embrittlement. The staff noted that the RPV supports are embedded into the concrete of the PSW and based on the applicant's conclusion in Appendix D this concrete is expected to have a loss of compressive strength due to the effects of neutron radiation and RIVE. However, the staff noted the evaluation of the RPV supports in Appendix E does not take into consideration the expected degradation of the PSW concrete due to irradiation.

The staff may need additional information regarding assessments of the degree of fixity of the RPV steel supports at the degraded PSW concrete to evaluate the CLB structural steel frame(s) capacities due to altered states associated with the induced cumulative irradiation effects during the SPEO. Specifically the staff may need information regarding the possible redistribution of maximum stresses (tension, compression, and shear), change in maximum IRs (tension, compression, and shear), pull-out/ slippage capacity, and any potential settlement of the RPV supports due to the expected degradation of the surrounding concrete caused by the combined effect of neutron fluence, gamma dose, and RIVE. The staff may need this information to assess the margin in available structural capacity under all stress conditions for the RPV support structure for the SPEO.

- Appendix E, "Irradiated Reactor Vessel Support Evaluation," of FPLCORP020-REPT-130, Revision 1, provides an evaluation of the RPV support steel for the aging effect of reduction of fracture toughness due to irradiation embrittlement. In this evaluation, the applicant opted to use the analysis methodology of NUREG-1509, which documented resolution of generic safety issue GSI-15, as its basis for a finding of reasonable assurance that the structures can perform their intended functions through the SPEO without employing aging management activities. The staff noted the following regarding the analysis described in Appendix E:

    a. The analysis used the fitted curve in Figure 3-1 of NUREG-1509, in contrast to the examples in the report that used the upper bound curve.

    b. The transition temperature analysis in Figure 4-4 of NUREG-1509, which, although uncited, is the analysis used in Appendix E, includes an action to "Evaluate TT_{EOL} + ***Margin*** ≤ LST" (emphasis added), where LST is the lowest service temperature; the analysis in Appendix E does not identify a margin term, which is intended to address uncertainty in the estimated NDT shift.

    c. 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 or cracks as cited in NUREG-1509.

    d. The report lacked discussion of the assessment of the reduction in fracture toughness for the bolting.

e. Identification of specific neutron fluence values, necessary for determination of neutron embrittlement, for RV steel support components at specific locations and elevations was not included.

The table below lists the documents that were reviewed by the staff and were found relevant to the review of these items. These documents were provided by the applicant.

Relevant Documents Reviewed

Document	Title	Revision / Date
EPRI Report No. 3002002676	Expected Condition of Reactor Cavity Concrete after 80-years of Radiation Exposure	March 2014
	I. Remec, ORNL, Radiation Environment in Concrete Biological Shields of Nuclear Power Plants	2013
FPLCORP020-REPT-130	Primary Shield Wall Evaluation	Revision 0
FPLCORP020-REPT-130	Primary Shield Wall Irradiation Evaluation	Revision 1
	I Remec, et al, ORNL, Characterization of Radiation Fields in Biological Shields of Nuclear Power Plants for Assessing Concrete Degradation	February 2016
	T.M. Rosseel, et al, ORNL, Radiation Damage in Reactor Cavity Concrete	September 2014
EPRI Report No. 3002011710	Irradiation Damage of the Concrete Biological Shield Wall for Aging Management	May 2018
EPRI Report No. 3002008129	Long-Term Operations: Impact of Radiation Heating on PWR Biological Shield Concrete	December 2016
	K.G. Field, et al, Perspective on Radiation Effects in Concrete for Nuclear Power Plants Part I: Qualification of Radiation Exposure and Radiation Effects	February 2015
ACI SP 55-10	H.K. Hilsdorf, et al, The Effects of Nuclear Radiation on the Mechanical Properties of Concrete	1978
Journal of Advanced Concrete Technology, Vol 15	Maruyama, et al, Development of Soundness Assessment Procedure for Concrete Members affected by Neutron and Gamma-Ray Irradiation	2017
Applicant's Audit Presentation	Turkey Point Units 3 and 4 Subsequent License Renewal Application Irradiation of Concrete NRC Audit July 17, 18, 19 and 20, 2018	
	Turkey Pont RV Support and Anchor Concrete Qualification for IHA	Revision 3 08/13/2004
NTPI-QI-17.0	Turkey Point Units 3 T-H Analysis of RV Support & Adjacent Concrete	08/17/10 Revision 3
	Turkey Point Units 3 and 4 Original Plant Calculations Civil (Containment-Reactor Shield Wall)	
Calc. No. CN-RVHP-08-38	Turkey Point Units 3 and 4 Reactor Vessel Support Stiffness Calculation	Revision 2
Calc. No. CN-RVHP-09-11	Turkey Point Units 3 and 4 EPU Project Reactor Vessel Support Load Reconciliation	Revision 2
PTN-BSHM-09-004	Impact of EPU on NOP Radiation Levels, Shielding Adequacy, and NOP Radiation Environments in EQ Zones	08/172010

PTN-BOHC-18-001	Evaluation of the Existing Reactor Shield Wall for CLB Loading for Units 3 and 4	Revision 0
NEECORP030	Response to NRC Requests for Information in Support of the Audit	07/16/2018
	Transmittal of Approved Calculation 2008-08528, Revision 1 Analysis of Postulated Reactor Vessel Head Drop Onto the Reactor Vessel Flange	09/24/2015
NEI 08-05	Industry Initiative on Control of Heavy Loads	Revision 0 July 2008
WCAP-14237	Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plant	December 1994
	Nuclear Regulatory Commission Docket #'s 50-250 and 50-251 Letter from Richard P. Croteau, Project Manager, Project Directorate II-I, Division of Reactor Projects-I/II, NRC, to Mr. J. H. Goldberg, President, Florida Power and Light Company, Subject: “Turkey Point Units 3 and 4, Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping (TAC Nos. M91494 and M91495),”	June 23, 1995
SPEC No. 5177-074-C-112	Performance Specification for Placing Reinforcing Steel	Rev 1
5177-074-C-103	Performance Specification for Forming, Placing, Finishing and Curing of Concrete	Rev 4
5177-074-C-121	Technical Specification for Purchase of Structural Steel	Rev 4
5177-074-C-122	Performance Specification for Erection of Structural Steel	Rev 4
5177-074-C-131	Technical Specification for Purchase of Miscellaneous Metal	Rev 8
5177-074-C-132	Performance Specification for Erecting Miscellaneous Metal	Rev 4
5177-074-C335	Technical Specification for Subcontract for Furnishing Concrete	Rev 4
5177-074-C351	Technical Specification for Purchase of Reinforcing Steel	Rev 3
5177-074-C441	Specification for Testing of Concrete-Related Materials	Rev 4
Book #13 Item #1	Turkey Point Units 3 and 4 Original Plant Calculations Civil Containment Nuclear Vessels Supports	
Spec. No. 5177-M-53	Performance Specification for the Control of Special Processes: Welding, Brazing and Heat-Treating for Plant Modification of Turkey Point Plant Units 3 & 4 Florida Power & Light Co.	Rev 23
Drawing No. 5610-C-192	Containment Structure Reactor Shield Wall Reinf. Sheet #1	3
Drawing No. 5610-C-561	Containment Structure Reactor Support Details	2
Drawing No. 5610-C-558	Containment Structure Reactor Primary Wall Liner Plate	0
Drawing No. 5610-C-193	Containment Structure Reactor Shield Wall Reinf Sheet #2	4
Drawing No. 5610-C186	Containment Structure Reactor Pit Foundation SHT 1	7
Drawing No. 5610-C-1313	Units 3 & 4 Containment General Arrangement for Removal and Replacement of Concrete, Rebar and Steel	0
Drawing No. 5610-C191	Containment Structure Reactor Shield Wall	2
Drawing No. 5610-M-56	Ground Floor Plan Elevation 18’-0”	66
Drawing No. 5610-M-400-4	Arrangement of Reactor Vessel Longitudinal Section	13
Drawing No. 5610-M-400-20	Turkey Point Units 3 & 4 Reactor Vessel Assembly & Final Machining	8
Drawing No. 5610-C-559	Containment Structure Reactor Primary Wall Penetrations Sheet #1	1

3. Applicant Personnel Contacted During Audit

Name	Affiliation
Bill Maher	Florida Power & Light (FPL)
Paul Jacobs	FPL
Steve Franzone	FPL
Brian Messitt	FPL
Chuck Ramdeem	FPL
Stephen Hale	ENERCON
Hoan-Kee Kim	ENERCON
Jeffrey Head	ENERCON
Andy Cianek	ENERCON
James Wicks	ENERCON
Mitch McFarland	ENERCON
John Ahearn	Westinghouse Electric Company (WEC)
Amy Freed	WEC
Arzu Alpan	WEC

4. Exit Meeting

An exit meeting was held with the applicant on October 17, 2018, to discuss the results of the operating experience audit. The staff is considering the issuance of requests for additional information to support completion of the staff's SLRA review.