L-2015-046
10 CFR 50.90
February 20, 2015

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Turkey Point Nuclear Plant, Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to Request for Additional Information Regarding License Amendment Request No. 229, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program"

References:

1. Florida Power & Light Company letter L-2014-033, License Amendment Request No. 229, Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," April 9, 2014 [ML 14105A042]

2. NRC letter "Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Request for Additional Information on License Amendment Request to Revise Technical Specifications to Implement TSTF-425, Revision 3, 'Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5B' (TAC Nos. MF3931 and MF3932)," August 7, 2014 [ML 14212A713]

3. Florida Power & Light Company letter L-2014-266 "Response to NRC Technical Specifications Branch Request for Additional Information Regarding License Amendment Request No. LAR-229, 'Application for Technical Specification Change Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program'," August 29, 2014 [ML 14252A228]

4. NRC letter "Request for Additional Information Re. LAR 229 for Turkey Point 3 & 4 (TACs MF3931 & MF3932)," January 22, 2015 [ML 15023A080]

In Reference 1 and supplemented by Reference 3, Florida Power & Light Company (FPL) submitted a request for an amendment to the Technical Specifications (TS) for Turkey Point Units 3 and 4. The proposed amendment would modify the TS by relocating specific surveillance frequencies to a licensee-controlled program with implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk-Informed

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Method for Control of Surveillance Frequencies." The changes are consistent with U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force Standard TS change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed TS Task Force] Initiative 5b," Revision 3.

In Reference 3, the NRC staff requested additional information in order to complete its review of the requested amendment. The enclosure to this letter provides FPL's response to the request for additional information (RAI).

This response to the RAI does not alter the conclusion in Reference 1 that the proposed changes do not involve a significant hazards consideration.

This RAI response contains no new regulatory commitments and does not modify any existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Mitch Guth, Licensing Manager, at 305-246-6698.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on February 20, 2015

Sincerely,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Enclosure

cc: NRC Regional Administrator, Region II
NRC Senior Resident Inspector
NRC Project Manager
Ms. Cindy Becker, Florida Department of Health

ENCLOSURE

Response to Request for Additional Information Regarding License Amendment Request No. 229, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program"

APLA RAI-1

APLA RAI-2

APLA RAI-3

APLA RAI-4

APLA RAI-1

Nuclear Energy Institute (NEI) 04-10, Revision 1 (ADAMS Accession No. ML071360456), Section 4.0, Step 8, states:

> The risk impact of a proposed [Surveillance Test Interval (STI)] adjustment shall be calculated as a change of the test-limited risk (see Regulatory Guide 1.177, Section 2.3.3). Since the test-limited risk is associated with failures occurring between tests, the failure rate that shall be used in calculating the risk impact of a proposed STI adjustment is the time-related failure rate associated with failures occurring while the component is in standby between tests (i.e., risk associated with the longer time to detect standby-stress failures).

Describe how the Turkey Point Surveillance Frequency Control Program will address the standby time-related contribution for extended surveillances.

Response

The standby time-related contribution evaluation will be performed in accordance with NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies," Revision 1. Any changes to the frequencies listed in the Surveillance Frequency Control Program (SFCP) will comply with the following guidance from NEI 04-10, Revision 1:

> *In general, the failure probability values of components used in probabilistic risk assessments (PRAs) consist of a time-related contribution (i.e. the standby time-related failure rate) and a cyclic demand-related contribution (i.e. the demand stress failure probability). The risk impact of a proposed STI adjustment shall be calculated as a change of the test limited risk (see Regulatory Guide 1.177, Section 2.3.3). Since the test-limited risk is associated with failures occurring between tests, the failure rate that shall be used in calculating the risk impact of a proposed STI adjustment is the time-related failure rate associated with failures occurring while the component is in standby between tests (i.e., risk associated with the longer time to detect standby-stress failures). Therefore, caution should be taken in dividing the failure probability into time-related and cyclic demand-related contributions because the test-limited risk can be underestimated when only part of the failure rate is considered as being time-related while this may not be the case. Thus, if a breakdown of the failure probability is considered, it shall be justified through data and/or engineering analyses. When the breakdown between time-related and demand-related contributions is unknown, all failures shall be assumed to be time-related to obtain the maximum test-limited risk contribution.*

APLA RAI-2

NEI 04-10, Revision 1, Section 4.0, Step 10, provides guidance on the initial assessment of Internal Events, External Events, and Shutdown Events. Describe how shutdown events will be assessed as part of the Turkey Point Surveillance Frequency Control Program.

Response

The shutdown risk evaluation will be performed in accordance with NEI 04-10, Revision 1, which permits quantitative or qualitative assessment of shutdown risk impacts. Fleet procedures will be written consistent with Nuclear Energy Institute (NEI) industry guidance document, NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies," Revision1, for performing the shutdown risk assessment. Documentation of the assessment will include the following:

- *Identification of applicable MODES of Operation that were used.*
- *If shutdown risk can be quantified, then core damage frequency (CDF) and large early release frequency (LERF) will be calculated for shutdown risk and included in the cumulative risk of all changes assessed. Turkey Point (PTN) does not currently have a RG 1.200 shutdown model. As such the shutdown risk assessments will be based on the PTN shutdown safety program developed in support of NUMARC 91-06, an application-specific shutdown analysis, a bounding sensitivity analysis, or other acceptable method described in NEI 04-10, Revision 1.*
- *Justification for a qualitative analysis (if quantitative was not used).*
- *Shutdown risk will be included in the comparison to applicable thresholds.*

APLA RAI-3

The enclosure to Attachment 2 of the LAR provides the peer review Facts and Observations (F&Os). Address the impact of the following F&Os on this application, clarifying the disposition of the F&O, as necessary:

DA-D5-01, DA-D6-01, DA-D6-02, IE-C14-01, IE-C14-02, IE-C14-03, IE-C14-04, IE-C14-05, IE-C14-06, and IE-C14-07.

Response

See table below.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
DA-D5-01	FPR 2013	For several CCF groups, a "global common cause event" (as described at the end of Section 4.2 of PTN-BFJR-2008-012, Rev. 0) is used. While this is a reasonable simplification, the global common cause event needs to account for the common cause combinations that are not included explicitly. However, for several 6-component groups (AFW AOVs FTO, AFW CVs FTO, AFW MOVs FTO), the 5-of-6 term was not included and the 6-of-6 term was not adjusted. A similar issue appears to be present for SG SVs FTO (4-component group), where only the 4-of-4 term is included (the 2-of-4 and 3-of-4 terms are missing and the 4-of-4 term was not adjusted).	Two alternatives. The missing CCF terms could be added to the CAFTA fault trees and CCF basic events calculated for the new terms. A simpler alternative is to revise the calculation of the α6 term to include the missing α5 value. Thus, α6' = α5 + α6. This overestimates the a5 contribution, since it is applied to the case where all 6 components fail, but this should be a small and conservative approximation. (Similar correction for the 4-component group, α4' = α2 + α3 + α4).	A complete CCF modeling update addressing this issue is planned for the next internal events model update. In fact, there is a working model with the CCF update already implemented. If a 5b application is started prior to the internal events model update, the working model with the CCF update will be used to perform a sensitivity analysis for the application.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
DA-D6-01	FPR 2013	The CCF notebook did not include a review of plant failure data for common cause events.	Review plant-specific component failure events from the most recent data update to identify any common cause failures. If CCFs are identified, verify that the CCF is modeled for the specific component and failure mode. If this data indicates a significantly larger fraction of failures are CCFs than the generic CCF parameters would predict, plant-specific CCF parameters should be calculated. If the data is limited (one or two failures in a specific component group), this would not be sufficient evidence to justify plant-specific CCF parameters.	During the review of plant records for the data analysis used in the current model-of–record, no common cause failure events were found. This fact will be added to the CCF notebook.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
DA-D6-02	FPR 2013	Section 3.0 of the CCF Notebook includes the assumption that CCFs are not included in fault tree initiating events with year-long mission times due to excessive conservatism in applying CCF factors that are developed for 24-hr mission time. However, this is not sufficient basis for excluding CCFs for fault tree IE models.	Provide a basis for excluding CCFs from system initiating events and include CCFs where a basis for exclusion cannot be established. For example, include CCF in system initiating event models only for active components that are in the same configuration (i.e., between normally operating pumps in the same system but not between operating and standby pumps in the same system).	CCFs are included for the components in the initiating event fault trees. For example, in the CCW system where 2/3 pumps are normally running, there are AND gates with a single FTR event of one of the normally running pumps with an 8760-hour mission time and CCF events for the other 2 running pumps with mission times equal to the MTR of the pumps. There is not a CCF for all 3 pumps with a mission time of 8760 hours, nor should there be; all 3 pumps are not normally running at the same time, and certainly not for 8760 hours. This F&O will therefore not be a factor in 5b applications.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
IE-C14-01	FPR 2013	RCP TBHX rupture probability - The IE frequency for tube rupture is based on a Reference 5 value of 3.48E-08/hr (peer review did not verify this reference) for "HX Tube External Leak Large >50 gpm". This hourly frequency is multiplied by 8760hr/yr for an annual IE frequency of 3.05E-04/yr. Depending on the application of the data, this IE frequency could be applied at each RCP, thus event tree top event "RCP TBHX Tubes Intact?" would be multiplied by a factor of 3. Applicability of the TBHX data to one or all RCPs should be examined/documented for impact on the total %ZZISLTBCCW initiator/results.	Assess the tube rupture original source data and whether it is applicable to each thermal barrier cooler/RCP. Revise initiator %ZZISLTBCCW and document any changes or basis accordingly.	Resolution of this F&O is planned for the next internal events model update. In fact, there is a working model with the RCP TBHX rupture probability change already implemented. If a 5b application is started prior to the internal events model update, the working model with this change will be used to perform a sensitivity analysis for the application.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
IE-C14-02	FPR 2013	Manual operator action is credited for local manual closure of MOV-*-626 (should it fail to close) and/or to local closure of manual valve *-736. Operator success ensures that the CCW piping remains intact. Although the HEP for the local action is 0.5, the time window basis should document to ensure that the operator has sufficient time to perform these actions before the CCW piping boundary fails.	Evaluate and document whether the operator action should be credited and remove credit for the action if it cannot be justified	The fact that the pressure increase in the CCW system due to the TBHX tube rupture would be mitigated by the CCW surge tank expansion volume and the relief valve RV-3/4-707 opening at 50 psig are obviously the reason some credit is given to closing a valve to isolate the leak. The time available for performing the isolation will depend upon the size of the rupture as well as other factors. This will be addressed in the next internal events model update. If a 5b application is started prior to the internal events model update, a sensitivity analysis will be performed by setting the HEP to 1.0 for the application.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
IE-C14-03	FPR 2013	Thermal Barrier ISLOCA IE Frequency – RCP Thermal Barrier CCW Supply Penetration #3 - This penetration is not evaluated for potential ISLOCA contribution. This penetration is protected by two normally open, active check valves (717 and 721A/B/C) inside containment and two normally open MOVs (716A/B) outside containment. The associated piping inside containment appears to be designed for full RCS pressure. However, given a thermal barrier tube breach, the active check valves could fail to close (w/CCF). The active failure of the outboard MOVs (also w/CCF) may be highly unreliable due to low differential pressure design capability and lack of relevant closure signals, and there might not be sufficient time for manual action. Failure of this penetration should be assessed for possible contribution to the TBCCW ISLOCA event frequency and sequences.	Evaluate and document the TBCCW supply penetration for possible ISLOCA initiating events. Should also assess the impact on CCW return line from RCP motor cooling and lifting of RV-729 if V-712A fails open. Ensure that these penetrations are also identified in Table 1, list of penetrations.	These penetrations for ISLOCA potential will be examined in the next internal events model update; however, the risk impact if these penetrations are included in the ISLOCA model will likely be minimal. The CCF of the two check valves to close is 5.2E-06. The frequency of a thermal barrier tube breach is likely less than 1E-03 per year, bringing us to a frequency of (1E-03 per year)*(5.2E-06) = 5.2E-09 per year. If no credit is assumed for the closing of the MOVs 716A/B, the ISLOCA will, at worst, fail the unit's CCW pumps. For a LOCA at PTN, all four HHSI pumps start on the SI signal and inject to the stricken unit, and the opposite-unit HHSI pumps will not be affected by the loss of CCW. Even if a CCDP of 1.0 is assumed, a delta CDF of 5.2E-09 per year will not be significant for any 5b application.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
IE-C14-04	FPR 2013	ISLOCA assessment of Penetration 1 (RHR SDC suction line) did not consider that the common suction piping beyond the RHR pumps could be affected by the over-pressurization event. This would impact the function of the high head SI pumps and the RWST (and Containment Spray pumps, which are not important in ISLOCA scenarios). As a result, the current RHR small ISLOCA event sequences apply too much credit for the associated Unit's RWST and HHSI pumps.	Evaluate and document the RHR small ISLOCA sequences taking no credit for associated Unit HHSI pumps and RWST.	Resolution of this F&O is planned for the next internal events model update. In fact, there is a working model with this modeling change already implemented. If a 5b application is started prior to the internal events model update, the working model with this change will be used to perform a sensitivity analysis for the application.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
IE-C14-05	FPR 2013	Penetrations 58/59/60: (HHSI cold leg injection) - These penetrations are qualitatively screened from further detailed evaluation on the basis that …."the combination of three check valves is equivalent to three locked/closed isolation valves", for meeting NUREG/CR-5928 criterion (c), systems isolated by redundant normally closed and locked manual valves that are independently verified to be closed and locked before plant startup". This comment is also applicable to Penetration 18. Additional basis is needed to support this equivalency assertion for screening these penetrations.	Review these penetrations and provide further basis for screening.	The relevant failure mode here is a check valve transferring open against the pressure that is holding it closed – difficult to conceive of the motive force causing such a failure. While perhaps not quite as secure of an isolation as three, locked-closed isolation valves, the series of 3 closed check valves is considered to be adequate for screening out these penetrations.
IE-C14-06	FPR 2013	Suggestion. The PTN ISLOCA analysis is based on early NUREG information and industry practice, which continue to provide a reasonable source of inputs/practice for consideration in ISLOCA modeling. In general however, the evaluation might benefit from aspects of the latest industry ISLOCA best practice/methodology presented in WCAP-17154, Rev.1.	Consider updating the ISLOCA evaluation to current industry practice and reference material. It is noted that there are limitations in the WCAP-17154, Revision 1 methodology and its complete adoption is not recommended.	This is a suggestion only. At the next opportunity, the ISLOCA analysis will be updated to the latest guidance, but a sensitivity analysis for 5b is not necessary.

F&O #	Peer Review	Description	Possible Resolution	Plans for Addressing in 5b Applications
IE-C14-07	FPR 2013	Suggestion. Table 1 "Potential ISLOCA Flow Paths" - Consider adding more detail in the ISL Screening Results column. For example, Penetrations 13 and 14 (Letdown and Charging) may not cleanly screen. Both systems interface with low pressure systems (letdown-purification piping and charging-pump suction). Typically there are redundant isolation means to isolate - thus IE frequency should be low. However, this cannot be concluded from the table details. Also, Penetration 3, "RCP CCW Supply" indicates that this penetration was screened based on "not connected to the RCS". However, this penetration provides the CCW supply to RCP thermal barrier cooling and should be assessed (refer to F&O IE-C14-2).	Consider updating the ISLOCA report to improve the details in Table 1, primarily the column information under "ISL Screening Results"	This is a suggestion only, dealing with documentation only. A sensitivity analysis for 5b is not necessary.

APLA RAI-4

The NRC staff notes that the response to PRA Request for Additional Information (RAI) 22.01 (ADAMS Accession No. ML14113A176) associated with the licensee's request to adopt National Fire Protection Association (NFPA) Standard 805 (NFPA 805) provides the results of the Turkey Point Gap Analysis to the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) Standard (ASME/ANS RA-Sa-2009) as endorsed by Regulatory Guide (RG) 1.200, Revision 2 (ADAMS Accession No. ML090410014).

Confirm whether the results of the assessment for F&Os LE-F1-01 and LE-G5-01, as discussed in PRA RAI 22.01, are applicable to the LAR to relocate specific surveillance frequency requirements to a licensee controlled program. If the results are not applicable, address the impact of the F&Os for this application.

Response

See table below.

F&O	Issue	Basis for Significance	Possible Resolution	Plans for Addressing in 5b Applications
LE-F1-01	Endstate frequency totals are given in Table 5 of the Level 2 notebook, PTN-BJFR-99-010, Rev. 1, and results by release category are given in Table 6. However, results using the Plant Damage State definitions of Section 4.2 are not provided. CC II is not met because relative contribution to LERF by PDS is not shown, although information is available to provide such data.	PDS relative contribution to LERF is not provided as specified in the SR.	Perform summary calculation to quantify PDS relative contribution to LERF.	This finding only addresses the categorization of LERF results. This will be done in the next model update, but will have no effect on 5b applications.

F&O	Issue	Basis for Significance	Possible Resolution	Plans for Addressing in 5b Applications
LE-G5-01	There is no discussion of limitations of severe accident understanding and modeling. This includes such matters as the impact of uncertainty regarding thermally induced SGTR on quantification, the uncertainty of ISLOCA break size and location on timing and source term, and the assignment of CET to endstates. Conservative treatment of some phenomena can affect LERF quantification, which in turn impacts LERF and delta LERF results when applying RG 1.174 guidelines in risk-informed changes to the licensing basis, for example.	Does not meet the intent of providing a discussion regarding limitations on the understanding of severe accident phenomenology, and how the Level 2 modeling uncertainties could impact LERF quantification and potential risk-informed applications.	Provide a discussion of possible limitations of the LERF analysis based on, for example, limitations on the state of severe accident understanding and level 2 PRA analysis. Briefly describe how key uncertainties in the LERF quantification could impact risk-informed changes to the licensing basis under RG 1.174, for example.	This finding only addresses the uncertainty discussion of LERF results. This will be done in the next model update, but will have no effect on 5b applications.