Florida Power & Light, 9760 S.W. 344 St. Homestead, FL 33035

FPL.

SEP 16 2011
L-2011-369
10 CFR 50.90

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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Request for Additional Information Regarding
Extended Power Uprate License Amendment Request No. 205

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.

(2) Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information – Round 2," Accession No. ML111010080, April 11, 2011.

(3) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-141), "Response to NRC Requests for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11137A080, May 11, 2011.

(4) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 1.3 (Part 3)," Accession No. ML11202A174, July 21, 2011.

(5) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-233), "Response to NRC Requests for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11221A227, August 5, 2011.

(6) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Request for Additional Information - Round 3," September 8, 2011.

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

On March 31, 2011, a public meeting was held with the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM), NRC staff technical reviewers, and FPL representatives to discuss proposed NRC Requests for Additional Information (RAIs) related to the EPU License Amendment Request (LAR) [Reference 1]. During the meeting, the NRC Reactor Systems Branch (SRXB) presented fourteen draft RAI questions for discussion. By email dated April 11, 2011 [Reference 2], the NRC PM issued the final RAI with ten of the original fourteen questions. On May 11, 2011, FPL provided its response to the ten RAI questions via letter L-2011-141 [Reference 3].

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By email from the NRC PM dated July 21, 2011 [Reference 4], additional information was requested by the NRC SRXB staff. The RAI consisted of thirty-nine questions pertaining to loss-of-coolant accident (LOCA) and non-LOCA analyses. On August 5, 2011, FPL provided its response to RAI questions SRXB-1.3.1-1.3.6 and 1.3.16-1.3.39 via FPL letter L-2011-233 [Reference 5].

By email from the NRC PM dated September 8, 2011, FPL received seven additional RAI questions [Reference 6]. Of these, RAI questions SRXB-3.1-3.4 are follow-up questions to the FPL response dated May 11, 2011 [Reference 3], and RAI questions SRXB-3.5-3.7 are follow-up question from the FPL response dated August 5, 2011 [Reference 5]. FPL's responses to these seven RAI questions are provided in the Attachment to this letter.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

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

Executed on September 16, 2011.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachment

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4

RESPONSE TO NRC SRXB REACTOR SYSTEMS RAI
REGARDING EPU LAR NO. 205

ATTACHMENT

Response to Request for Additional Information

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

On March 31, 2011, a public meeting was held with the NRC Project Manager (PM), NRC staff technical reviewers, and FPL representatives to discuss proposed NRC RAIs related to the EPU LAR [Reference 1]. During the meeting, the NRC Reactor Systems Branch (SRXB) presented fourteen draft RAI questions for discussion. By email dated April 11, 2011 [Reference 2], the NRC PM issued the final RAI with ten of the original fourteen questions. The ten RAI questions involved specific piping design configurations, reactor fluence calculations, RCS temperature distributions and cooldown times, heat exchanger fouling factors, RHR system capability, natural circulation characteristics, and core outlet temperature monitoring. On May 11, 2011, FPL provided its response to the ten RAI questions via FPL letter L-2011-141 [Reference 3].

By email from the NRC PM dated July 21, 2011 [Reference 4], additional information was requested by the NRC SRXB staff. The RAI consisted of thirty-nine questions regarding loss-of-coolant accident (LOCA) and non-LOCA analyses. On August 5, 2011, FPL provided its response to RAI questions SRXB-1.3.1-1.3.6 and 1.3.16-1.3.39 via FPL letter L-2011-233 [Reference 5].

By email from the NRC PM dated September 8, 2011, FPL received seven additional RAI questions [Reference 6]. Of these, RAI questions SRXB-3.1–3.4 are follow-up questions to the FPL response dated May 11, 2011 [Reference 3], and RAI question SRXB-3.5-3.7 are follow-up questions from the FPL response dated August 5, 2011 [Reference 5]. FPL's responses to these seven RAI questions are provided below.

SRXB-3.1 **SRXB-2.6 stated that there appears to be no information that addresses the effect of the EPU on heat exchanger fouling factors. It asked that FPL address the behavior of heat exchanger fouling factors due to the higher heat load, longer cooldown times, and greater differential temperatures. In its response, FPL established that the excess cooling capacity that is illustrated by the need for throttling early in cooldown establishes that fouling is not a concern during early cooldown. Further, the maintenance program and technical specification action requirement verifies that design basis heat loads will be removed at the time of the verification but does not ensure potentially increased fouling will not occur that affects cooldown rate later in the cooldown. FPL has not established that fouling will not extend the Turkey Point predicted cooldown times beyond acceptable limits. Provide information to substantiate that fouling is not a concern.**

An example of an acceptable response would include an analysis that introduces fouling into the heat exchangers and establishes the amount of fouling that could be permitted while still achieving acceptable cooldown times. A comparison of this prediction to the existing allowance for fouling and operational experience could provide a quantitative assessment of allowance for fouling to support EPU operation; information that would support a judgment that residual heat removal (RHR) operability is

reasonably ensured. A commitment for follow-up assessment based on the first cooldown from EPU conditions would provide substantiating data and margin between prediction and experience.

The cooldown analysis for the PTN EPU project determined a UA value of the component cooling water (CCW) heat exchanger (HX) that includes an appropriate fouling level based on the following methodology. HX U and UA values were re-calculated for the CCW HXs to maintain the validity of the CCW HX operability curves used to determine HX effectiveness and maintenance intervals. The appropriate values of the heat transfer coefficients were calculated at the intake cooling water (ICW) temperatures to be used in the cooldown analysis, using CCW system and component design information. In order to maintain the validity of the operability curves, all calculations of UA and U performed for this analysis, at any temperature, used a constant heat transfer rate of 47.6 MBTU/hr per CCW HX.

The re-calculated values for HX UA and U, as well as the shell and tube side design flow rates, were used to automatically calculate a corrected UA value that describes actual HX performance at plant transient conditions. This corrected UA value, which includes the fouling factor, is then used in the cooldown analysis.

PTN UFSAR Table 9.3-1 also identifies the CCW HX heat removal capability (nominal design) as 47.6 MBTU/hr (per HX), representing the heat removal rate at design basis accident conditions. Actual heat removal will vary as a function of tube side and shell side flow rates, ICW temperature, accident conditions, and HX fouling.

Technical Specification surveillance requirements provide minimum requirements for determining CCW HX operability. To comply with these requirements, the CCW HX performance monitoring program was implemented. This program requires recording actual ICW and CCW conditions frequently and comparing these conditions against the calculated maximum ICW inlet temperature, with suitable design margin applied. Turkey Point assumes one half degree per day decrease of the calculated maximum allowable ICW temperature. When the margin between the decreasing allowable ICW temperature and the actual ICW temperature encroaches on three degrees, the HXs are scheduled for cleaning. However, Turkey Point built into the maintenance program a requirement to proactively clean the CCW HXs even before reaching the maximum allowed fouling factor. Review of the historical data has indicated that there is adequate margin between the observed fouling factor (calculated using the operability curves) and maximum allowable fouling factor used in the design basis CCW HX performance analysis at any given time. This monitoring and trending program thus ensures that the worst pair of CCW HXs will continue to retain the ability to remove the design basis heat load at the actual average canal temperature.

As described in the response to SRXB-2.6, the HX fouling factors are a function of water chemistry and cleanliness, which will not change due to the EPU conditions. The heat load, cooldown times, and differential temperatures have little effect on the overall fouling factors.

Plant operating data supports the judgment that there will be no significant increase in fouling of the CCW HX over the course of a normal plant cooldown. As described in the response to SRXB-2.4, normal plant cooldown time to cold shutdown (Mode 5 - 200°F) with both trains of CCW and RHR available decreased from 30 hours for the current power rating to 28 hours for the EPU. Hence, since the calculated normal cooldown times are not significantly different between pre- and post-EPU conditions, there is no concern over increased fouling of the CCW HXs during the course of a normal plant cooldown at EPU conditions. Because the assessments that are already performed for the HX monitoring program are shown to prevent the CCW HXs from exceeding the maximum allowable tube fouling, a separate assessment following the first cooldown from the EPU power level will not be required.

SRXB-3.2 **SRXB-2.7 requested justification for the conclusion that the EPU has no effect on the ability of the RHR system to remove residual heat at reduced reactor coolant system (RCS) inventory, and, therefore, Turkey Point will continue to meet the current licensing basis requirements with respect to NRC Generic Letter 88-17. FPL's conclusion is essentially that a 15 percent increase in decay heat generation rate will have no effect on the RHR system operation during reduced inventory operation. FPL has not justified this conclusion.**

As stated in SRXB-2.7, justify this conclusion in light of the increased decay heat generation rate that must be removed after shutdown. Include the effect on temperature, RHR flow rate including any limitations on flow rate as a function of RCS water level, and potential hot leg vortexing in your justification.

Plant operating procedures currently require a minimum shutdown time of 60 hours before entry into reduced inventory is permitted. Under EPU conditions, the plant will be required to remain shutdown for 126 hours before entering reduced inventory. This increase in waiting time ensures that decay heat generated during reduced inventory at EPU will not exceed the levels currently generated at reduced inventory. Therefore, the existing limitations on residual heat removal (RHR) system flow, RCS water level, and RCS temperature during reduced inventory will remain in effect for EPU. With RCS and RHR system fluid conditions at reduced inventory unaffected by EPU, the potential for hot leg vortexing is also unaffected, and PTN will continue to meet the current licensing basis requirements with respect to GL 88-17 [Reference 7].

SRXB-3.3 **SRXB-2.8 asked for a comparison of upper head temperatures predicted to exist during natural circulation cooldown for the existing power level and the proposed power level and requested that saturation temperature at the uppermost upper head elevation be included in the comparison. FPL did not provide this information for Turkey Point. Provide a table or graph that shows maximum upper head temperature at the head / water interface, RCS pressure, and saturation temperature as a function of time starting at the time of reactor trip and ending at the time of RHR initiation.**

pressure, and saturation temperature as a function of time starting at the time of reactor trip and ending at the time of RHR initiation.

As provided in the response to SRXB-2.8, voiding in the reactor vessel head is prevented by limiting the RCS cooldown rate to 25°F/hr as cited in the Emergency Response Guidelines (ERGs) on which Turkey Point specific Emergency Operating Procedures (EOPs) are based. The 25°F/hr cooldown rate provided in the ERGs is based on a generic bounding analysis performed by the Westinghouse Owners Groups (WOG) in response to NRC Generic Letter (GL) 81-21 [Reference 8]. In that generic analysis the upper head temperature at time of reactor trip was conservatively assumed to be equal to the vessel outlet temperature of 618.3°F. For Turkey Point, at the proposed power level, this temperature would be 616.8°F at time of reactor trip; therefore, the generic analysis is bounding for upper head temperature at transient start, and remains bounding through the entire cooldown by following the 25°F/hour maximum cooldown rate in the EOPs.

Additionally, as previously stated in the response to SRXB-2.8 the primary means of heat removal from the upper head and associated fluid is through the use of Control Rod Drive Mechanism (CRDM) cooling fans. For the proposed increased power level, the heat removal ability of the CRDM fans is unaffected, and only one CRDM cooling fan is needed to provide the heat removal capacity used in the generic analysis. Using the bounding values provided in the generic analysis, the heat removal rate provided only by CRDM cooling fans is 21°F/hr at transient start and decreases to 11°F/hr when the upper head is cooled to 350°F. An additional upper head cooldown rate of 10°F/hr is provided by the 25°F/hr RCS natural circulation cooldown rate. Therefore, the minimum total upper head cooldown rate is approximately 21°F/hr which ensures that the upper head temperatures do not significantly deviate from that of the RCS.

Since upper head temperature during natural circulation cooldown will closely follow that of the RCS, the potential for voiding in the upper head is eliminated, and the current licensing basis requirements with respect to GL 81-21 will continue to be met for EPU. Upper head temperature values were not directly calculated in the natural circulation cooldown analysis and, therefore, are not available to present in chart or tabular format.

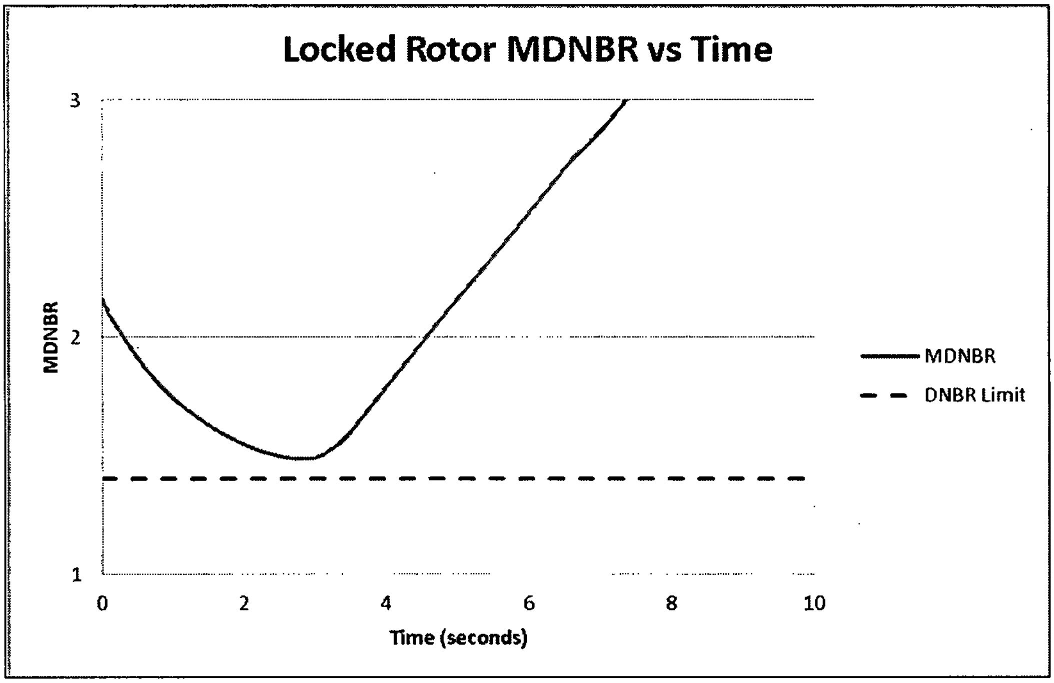
SRXB-3.4 **FPL states that "Table 2.8.7.2-4 ... shows ... Tavg decreases at approximately 25°F/hr as prescribed in the EOPs to insure head voiding doesn't occur." Table 2.8.7.2-4 illustrates cooldown at 1 hour with Tave = 585 °F. This corresponds to Tave = 610 °F at reactor trip at a cooldown rate of 25 °F/hr; somewhat less than the vessel outlet temperature of 616.8 °F in Table 2.8.3-1. Describe the reason for the difference in temperatures at the time of reactor trip.**

As discussed in the EPU Licensing Report (LR), Section 2.8.7.2, the natural circulation cooldown analysis assumes an initial 4-hour hold at hot standby conditions before commencing the 25 °F/hr cooldown. Thus, 610 °F should not be taken as the initial T_{avg} based on a reported T_{avg} of 585 °F at 1 hour. (For the proposed power level, T_{avg} at 100% power is 587 °F.)

Values in LR Table 2.8.7.2-4 for hour 1 are values equal to 1 hour after reactor trip. At time of reactor trip the analysis conservatively assumed T_{hot} equal to the maximum full power core exit temperature of 620.8 °F (versus the vessel outlet temperature of 616.8 °F); this is displayed in LR Table 2.8.7.2-1. Per LR Table 2.8.7.2-4, at 1 hour after reactor trip, ΔT through the core will decrease to a value of 24.08 °F, and T_{hot} would correspond to a value of 597.04 °F. This equates to about a 24 °F drop in hot leg temperature in the first hour while T_{avg} remains relatively stable as described above.

SRXB-3.5 For Licensing Report Section 2.8.5.3.2, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break, provide a plot of departure from nucleate boiling ratio versus time.

Below is a plot of the Minimum Departure from Nucleate Boiling Ratio (MDNBR) versus Time for the Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (referred to as Locked Rotor) for the Turkey Point Extended Power Uprate. The plot below represents the results for the Upgrade fuel, which bound the results of the Debris-Resistant Fuel Assembly (DRFA) fuel. The applicable DNBR limit is also shown on the plot.



SRXB-3.6 The current Rotor Seizure analysis, as described in the Turkey Point FSAR, concludes that fewer than 10% of the fuel rods undergo departure from nucleate boiling, while the analysis also conservatively assumes that 10% of the fuel rods fail. The EPU submittal assumes no failed fuel and indicates that a fuel upgrade, an increased minimum measured flow (MMF), and a reduced FΔH are responsible for this reduction. Provide more detail on the changes that lead the EPU analyses to conclude that there is no fuel damage.

As stated in LR Section 2.8.5.3.2.2.2 and in LR Table 2.8.5.0-1, the radiological dose analysis assumes 15% fuel rod failure for the Extended Power Uprate (EPU) Locked Rotor event. The DNB analysis of the EPU Locked Rotor event predicts no rods-in-DNB. The EPU analysis predicts fewer rods-in-DNB than the existing analysis of record, primarily due to the improved DNB performance of the 15x15 Upgrade fuel. This improved performance is accomplished by adding Intermediate Flow Mixer (IFM) grids, thereby decreasing the distance between mixing grids and increasing the turbulence in the limiting DNB region. The benefit from this improved mixing is the primary factor which results in higher predicted DNBRs, leading to the conclusion of 0% rods-in-DNB.

In addition to the effect of the IFMs, there are parameter changes associated with the EPU which provide DNBR benefits in the EPU analyses. For the EPU, the $F_{\Delta H}$ peaking factor was reduced approximately 3% for the Upgrade fuel and 20% for the remaining DRFA fuel, which does not contain IFMs. The minimum measured flow was increased by approximately 2%. Also, the analyses for the EPU were performed using transient VIPRE modeling described in WCAP-14565-P-A (Reference 9), where both thermal / hydraulic and conduction / convection calculations are performed simultaneously in VIPRE. As discussed in Reference 9 (RAI Question 13), the transient VIPRE modeling results in a small increase in predicted DNBR over the previous THINC/FACTRAN modeling.

The results of the analysis for Locked Rotor show that the minimum DNBR remains above the limit value, and thus there are no rods-in-DNB predicted. This is less than the 15% value assumed in the radiological dose analysis. The results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload process, consistent with the NRC-approved Westinghouse reload methodology (Reference 10).

SRXB-3.7 Provide Reference 5 from the response to RAI 1.3.22. NS-NRC-89-3466 that contains the detailed summary of the technical analysis and licensing bases for the use of the 2700°F peak clad temperature limit as an acceptable criterion for coolability in non-LOCA events.

NS-NRC-89-3466 was a letter previously issued from Westinghouse to Mr. Robert C. Jones, Reactor Systems Branch Chief, in 1989, but it is not currently available in the NRC records. Since it is a Westinghouse Proprietary letter which should not be placed on the docket along with these responses, it is being provided under separate cover by Westinghouse.

References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.
2. Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information – Round 2," Accession No. ML111010080, April 11, 2011.

3. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-141), "Response to NRC Requests for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11137A080, May 11, 2011.

4. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 1.3 (Part 3)," Accession No. ML11202A174, July 21, 2011.

5. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-233), "Response to NRC Requests for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11221A227, August 5, 2011.

6. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Request for Additional Information - Round 3," September 8, 2011.

7. NRC Generic Letter (GL) 88-17, "Loss of Decay Heat Removal," Accession No. ML031200496, October 17, 1988.

8. NRC Generic Letter (GL) 81-21, "Natural Circulation Cooldown," Accession No. ML031080586, May 5, 1981.

9. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Transmittal Accession No. ML993160073, October 1999.

10. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.