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L-2011-070
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 (LAR) No. 205 and Piping and Non-Destructive Examination Issues

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 - Piping and NDE Branch (CPNB) Request for Additional Information - Round 1," Accession No. ML110590491, February 28, 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).

By email from the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM) dated February 28, 2011 [Reference 2], additional information regarding piping design and inspection issues was requested by the NRC staff in the Piping and NDE (Non-Destructive Examination) Branch (CPNB) to support their review of the EPU License Amendment Request (LAR) [Reference 1]. The Request for Additional Information (RAI) consisted of six (6) questions regarding selected plant specific general design criteria, inspection programs and results. These six RAI questions and the applicable FPL responses are documented 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.

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NRR

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

Executed on March 23, 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 RAI REGARDING EPU LAR NO. 205
AND CPNB PIPING AND NDE ISSUES

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) No. 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL letter L-2010-113 on October 21, 2010 [Reference 1].

In an email dated February 28, 2011 [Reference 2], the NRC staff requested additional information regarding FPL's request to implement the Extended Power Uprate. The RAI consisted of six (6) questions from the NRC Piping and NDE Branch (CPNB) regarding selected plant specific general design criteria, inspection programs and results. These six RAI questions and the applicable FPL responses are documented below.

CPNB-1.1 By letter dated October 21, 2010, in the section titled "Current Licensing Basis" you state that "GDC-31 is analogous to the 1967 Atomic Energy Commission Proposed GDC-35 on Reactor Coolant Pressure Boundary Brittle Fracture Prevention which was subsequently deleted in favor of GDC-34. PTN has no commitment to GDC-35." Clarify the above statement and describe how you meet the criteria of GDC-31, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.

On July 10, 1967, the Atomic Energy Commission (AEC) issued a proposed set of General Design Criteria (GDC) for comment. On October 2, 1967 the Atomic Industrial Forum (AIF) issued its comments on those GDCs. PTN's licensing basis was and is based on the proposed AEC GDCs as amended by the AIF. 10CFR50 Appendix A GDC-31 is analogous to the 1967 AEC GDCs 34 and 35. However, the AIF amended version of GDC-34 effectively combined the two AEC criteria.

10CFR50 Appendix A, Criterion 31--Fracture Prevention of Reactor Coolant Pressure Boundary: The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

AEC GDC-34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention: The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

AEC GDC-35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention: Under conditions where the reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation of 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

PTN GDC-34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention: "The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes."

As described in UFSAR Section 4.1.3, the reactor coolant pressure boundary has been shown to be capable of accommodating without rupture the static and dynamic loads imposed as a result of a sudden reactivity insertion and is designed to reduce to an acceptable level the probability of a rapidly propagating type failure. As described in Technical Specification (TS) 3/4.4.9, "Pressure / Temperature Limits", and its bases documentation, all components in the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are induced by normal load transients, reactor trips and startup and shutdown operations. During RCS heatup and cooldown, the temperature and pressure changes are limited to be consistent with design assumptions and to satisfy stress limits for brittle fracture. These limits are consistent with the requirements given in ASME Boiler and Pressure Vessel Code, Section III, Appendix G. Thus, brittle fracture during normal operation is not considered to be credible. As described in UFSAR Section 4.3.3, Charpy V-notch toughness test curves are run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator, and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times as part of the design control of materials. All pressure containing components of the RCS are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in UFSAR Table 4.1-9.

CPNB-1.2 Code Case N-729-1 requires: an initial bare metal visual examination to be performed before or during the third refueling outage after installations of the replacement head, or within 5 calendar years of replacement, whichever occurs first; and repeat bare metal visual examinations shall be performed at least every third refueling outage or every 5 calendar years, whichever occurs first. The licensee performed bare metal visual inspections of the Unit 3 replacement reactor vessel closure head (RVCH) during the spring 2009 outage and performed bare metal visual inspections on the Unit 4 RVCH

during the fall 2009 outage. What were the results of the bare metal visual examinations of the reactor vessel closure heads for both Units 3 and 4?

Bare metal visual inspections of the reactor vessel closure heads for Units 3 and 4 found no evidence of accumulated boric acid on any of the reactor vessel closure head penetrations or on the heads themselves. 100% coverage was obtained for both units.

CPNB-1.3 Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55(a)(g)(6)(ii)(D)(3) states that, "Instead of the specified 'examination method' requirements for volumetric and surface examinations in Note 6 of Table 1 of Code Case N-729-1, the licensee shall perform volumetric and/or surface examination of essentially 100 percent of the required volume or equivalent surfaces of the nozzle tube, as identified by Figure 2 of ASME Code Case N-729-1. A demonstrated volumetric or surface leak path assessment through all J-groove welds shall be performed. If a surface examination is being substituted for a volumetric examination on a portion of a penetration nozzle that is below the toe of the J-groove weld [Point E on Figure 2 of ASME Code Case N-729-1], the surface examination shall be of the inside and outside wetted surface of the penetration nozzle not examined volumetrically." Turkey Point's inspection plan does not appear to be consistent with 10 CFR 50.55(a)(g)(6)(ii)(D)(3). Provide explanation of your inspection plan and any deviations from the regulation. If there are any deviations, provide justification for deviating from the regulation.

The Turkey Point Inservice Inspection Program Plan quotes the following paragraph from 10CFR50.55(a)(g)(6)(ii)(D)(1) for the required examination of the reactor vessel head penetration:

> "All licensees of pressurized water reactors shall augment their inservice inspection program with ASME Code Case N–729–1 subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section. Licensees of existing operating reactors as of September 10, 2008 shall implement their augmented inservice inspection program by December 31, 2008. Once a licensee implements this requirement, the First Revised NRC Order EA–03–009 no longer applies to that licensee and shall be deemed to be withdrawn."

Turkey Point understands that essentially 100% of the required volume or equivalent surfaces of the nozzle tube shall be examined by volumetric and or surface examination as identified by Figure 2 of ASME Code Case N-729-1 and, if for any reason a deviation is needed, Appendix 1 of ASME Code Case N-729-1 shall not be used without prior approval from the NRC.

CPNB-1.4 The licensee stated that they have recently adopted a comprehensive Alloy 600 management program that identifies Alloy 600/82/182 locations, evaluates and prioritizes the locations based 2 on primary water stress corrosion cracking (PWSCC) susceptibility and develops mitigation and repair options. Summarize the results of volumetric examinations performed during the past inservice inspection of all Alloy 82/182 welds in the reactor coolant system (RCS).

Turkey Point does not have any butt welds containing Alloy 600/82/182 in the

Inservice Inspection Program. The only Alloy 600 material is located on the lower reactor head nozzle welds which are subject to an augmented examination performed per ASME Code Case N-722.

CPNB-1.5 The licensee stated that at Turkey Point a small increase in the hot leg temperature was assessed due to the EPU and that the effect of this change in the service temperature on the thermal aging is considered. The Topical Report WCAP-14575-A, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, indicates that thermal aging causes reduction in fracture toughness of the cast austenitic stainless steels (CASS) component material and hence reduction in the critical flaw size that could lead to component failure. The impacted RCPB CASS components include RCS piping elbows, valve bodies, reactor coolant pump (RCP) casing and closure flanges. The evaluation documented in WCAP-15354, Technical Justification for Eliminating Large Primary Loop Pipe rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants for the 60 Year Plant Life, demonstrated that a significant margin exists between detected flaw size and flaw instability. The increase in the hot leg is within the evaluation of WCAP-15354 and accordingly an aging management program to manage the effect for the RCS piping components is not required beyond the examinations required by ASME Section XI.

Westinghouse performed an evaluation of the Code Case N-481 integrity analysis to identify if it is acceptable for the extended operating period. The results of the evaluation concluded that the previous integrity analysis conclusions documented in WCAP-13045, Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems, and WCAP-15355, A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of the Turkey Point Units 3 and 4, for the Turkey Point Units 3 and 4 RCP casings remain valid for the 60-year licensed operating period. The increase in the hot leg is within the evaluations of WCAP-13045 and WCAP-15355 and an aging management program beyond the examinations required in Section XI is not required to manage the thermal embrittlement effect for the RCP casings.

Topical Report, WCAP-14575-A proposed programs to manage the effects of thermal aging of CASS components during the period of extended operation. The NRC assessed these programs and the safety evaluation (Section 3.3.3) states that Valve bodies are adequately covered by existing inspection requirements in Section XI of the ASME Code and that screening for susceptibility to thermal aging is not required during the period of extended operation because the potential reduction in fracture toughness of these components should not have a significant impact on critical flaw size. The licensee believes that thermal aging as a result of the EPU is not expected to significantly affect cast components, including pumps, piping and valves at Turkey Point. Justify using WCAP reports, WCAP-15354, WCAP-13045, and WCAP-15355 as the basis for considering thermal aging due to the increase of the hot leg temperature.

The evaluation documented in WCAP-15354, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants for the 60 Year Plant Life," used the fully aged (saturated) fracture toughness values for the hot leg. Fully aged fracture toughness values used in the evaluation are very conservative (low) as calculated by the Westinghouse methodology. If NUREG/CR-4513 Revision 1 methodology [Reference 3] is used then the saturated fracture toughness values will be 2 times higher than the values used in WCAP-15354. Also, the change in fracture toughness values for a 9°F hot leg temperature change will be about 0.5% (which is negligible) if NUREG/CR-4513 Revision 1 methodology is used. Considering that very conservative fracture toughness values were used in the WCAP-15354 evaluation and the change in fracture toughness values will be negligible for a change in 9°F hot leg temperature, the evaluation performed in WCAP-15354 remains acceptable for the EPU conditions.

WCAP-15355, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of the Turkey Point Units 3 and 4," used the normal operating temperature of 550°F which is about 1°F higher than the maximum EPU temperature for the cold leg and cross-over leg. This 1°F temperature change will have insignificant impact on the fracture temperature values used in the WCAP-15355 evaluation, and the WCAP-15355 temperature is also bounding and conservative compared to EPU. The WCAP-15355 conclusions are therefore shown to be acceptable under EPU conditions. WCAP-13045 report is for the WOG generic report and the information from this WCAP report was used in the WCAP-15355 evaluation.

Based on this comparison of the key parameters associated with the EPU to those assumed in the WCAP reports, it can be concluded that the EPU will not have a significant effect on the thermal aging of the cast components at Turkey Point, including pumps, piping, and valves.

CPNB-1.6 The licensee states that environmentally assisted fatigue cumulative usage factor (CUF) for RCS components that were evaluated during License Renewal were shown to be less than 1.0 for EPU conditions, with the exception of the RCS hot leg pressurizer surge line nozzle which assumed the presence of conservative stratification loads. The licensee states that this result is consistent with the CUF evaluation performed for License Renewal for which the licensee has committed to inspect all Turkey Point welds in the surge lines of Units 3 and 4. All welds have been inspected except one, which will be completed during the 2010 fall outage. Results of all inspections for the surge line welds show no effects of the stratification loads. What was the result of the above mentioned remaining inspection during the 2010 fall outage?

An ultrasonic test (UT) examination of the outstanding Unit 3 weld on the RCS pressurizer surge line was completed on October 9, 2010 and found acceptable.

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 (NRC), "Turkey Point EPU – Piping and NDE (CPNB) Request for Additional Information - Round 1," Accession No. ML110590491, February 28, 2011
3. O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 1, U. S. Nuclear Regulatory Commission, Washington, DC, August 1994