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

SEP 30 2011
L-2011-400
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 Reactor Systems Branch 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 S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 1.3 (Part 3)," Accession No. ML11202A174, July 21, 2011.

(3) 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.

(4) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 2.3 (Part 3)," September 20, 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 dated July 21, 2011 [Reference 2], the Nuclear Regulatory Commission (NRC) Project Manager (PM) issued a Request for Additional Information (RAI) from the NRC Reactor Systems Branch (SRXB). 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 3].

By email from the NRC PM dated September 20, 2011 [Reference 4], FPL received two follow-up RAI questions requiring additional analyses to be performed for two events that had not been previously analyzed for Turkey Point: Inadvertent Opening of a Power-Operated Relief Valve (PORV) and Feedwater Line Break events. The analysis results for the Inadvertent Opening of a PORV (RAI question SRXB-2.3.1, a follow-up to SRXB-1.3.29) are presented in the attachment to this letter.

A001
NRR

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 30, 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 REACTOR SYSTEMS BRANCH (SRXB)
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].

By email dated July 21, 2011 [Reference 2], the Nuclear Regulatory Commission (NRC) Project Manager (PM) issued a Request for Additional Information (RAI) from the NRC Reactor Systems Branch (SRXB). 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 3].

By email from the NRC PM dated September 20, 2011 [Reference 4], FPL received two follow-up RAI questions requiring additional analyses to be performed for two events that had not been previously analyzed for Turkey Point: Inadvertent Opening of a Power-Operated Relief Valve (PORV) and Feedwater Line Break events. The analysis results for the Inadvertent Opening of a PORV (RAI question SRXB-2.3.1, a follow-up to SRXB-1.3.29) are presented below.

SRXB-2.3.1 In the Turkey Point Nuclear (PTN) Final Safety Analysis Report (FSAR) Chapter 3, "Reactor," Section 3.1, "Design Basis," it is stated, "the Reactor Control and Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio greater than or equal to the DNBR limit of the applicable DNB correlation."

The equation for the overtemperature delta-T (OTΔT) trip setpoint includes a differential pressure compensation term, which adjusts the trip setpoint based on deviations from the nominal reactor pressure. The setpoint is lowered when the reactor pressure is less than nominal, because a lower pressure decreases the subcooling margin. It is necessary to validate the DNB-protective function against both transients that increase the reactor heat flux, such as rod cluster control assembly withdrawal errors, and transients that reduce reactor pressure, such as a spurious power-operated relief valve or safety valve opening.

While in your response to RAI SRXB-1.3.29 asserts that the OTΔT trip provides the requisite protection, it is without a valid safety analysis on which to base this assertion. Demonstrate that this claim is true. A supporting analysis is necessary in order to conclude that the trip setpoint proposed for TS 2.2, Functional Unit 5, has been "so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded," consistent with the requirement promulgated by 10 CFR 50.36(c)(1)(ii)(A).

Event Description:

An accidental depressurization of the Reactor Coolant System (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening,

the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback (positive MTC), but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor may be tripped by the following reactor protection system signals:

a. Overtemperature ΔT

b. Pressurizer low pressure

Method of Analysis:

The accidental depressurization transient is analyzed by employing the detailed digital computer code RETRAN. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Normal reactor control systems are not required to function, with the exception of the rod control system which is conservatively assumed to be in automatic mode because it functions to maintain the power and average temperature for as long as possible during the transient, which delays reactor trip and results in a limiting case. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

Inputs were modeled to maximize the decrease in minimum DNBR consistent with the Revised Thermal Design Procedure (RTDP) and current Westinghouse guidance as provided in Table SRXB-2.3.1-1 below.

Results:

The calculated percent margin to minimum DNBR is 13.6%. A time sequence of events table is presented below in Table SRXB-2.3.1-2. Additionally, the transient plots for this analysis are provided in Figures SRXB-2.3.1-1 through SRXB-2.3.1-5 below. Note that a steady state run of 100 seconds preceding the transient has been removed for clarity. The final 100 seconds of transient time is plotted below.

Conclusion:

The results of the RCS Depressurization analysis confirm that the OTΔT trip provides adequate protection during an RCS Depressurization transient by demonstrating that the minimum DNBR safety analysis limit is not violated.

Table SRXB-2.3.1-1
Critical Parameters for the RCS Depressurization Analysis

Parameter	Value	Conservative Direction
Nuclear Steam Supply System (NSSS) Power Rating (MWt)	2652	High
Total RCS Flow Rate (gpm)	270000	Low
Moderator Temperature Coefficient (pcm/°F)	0.0	High (1)
Doppler Power Coefficient (pcm/%power vs. power)	-9.55+0.037Q(2)	High (1)
Nominal Pressurizer Pressure (psia)	2250	---
Nominal Pressurizer Water Level (% span)	60.0	---
Nominal RCS Average Temperature (°F)	583.0	High
Initial Steam Generator Water Level (% NRS)	50.0	---
Steam Generator Tube Plugging (%)	10.0	High

(1) Least negative
(2) Q = Percent of Nominal Power

Table SRXB-2.3.1-2
Time Sequence of Events for the RCS Depressurization Analysis

Event	Time (sec)
Transient Initiation	100.0
OTΔT Reactor Trip Signal	136.90
Rod Motion Begins	138.90
Minimum DNBR	139.50
Turbine Trip	141.40
Safety Injection Signal	148.17

Figure SRXB-2.3.1-1
Nuclear Power vs. Time

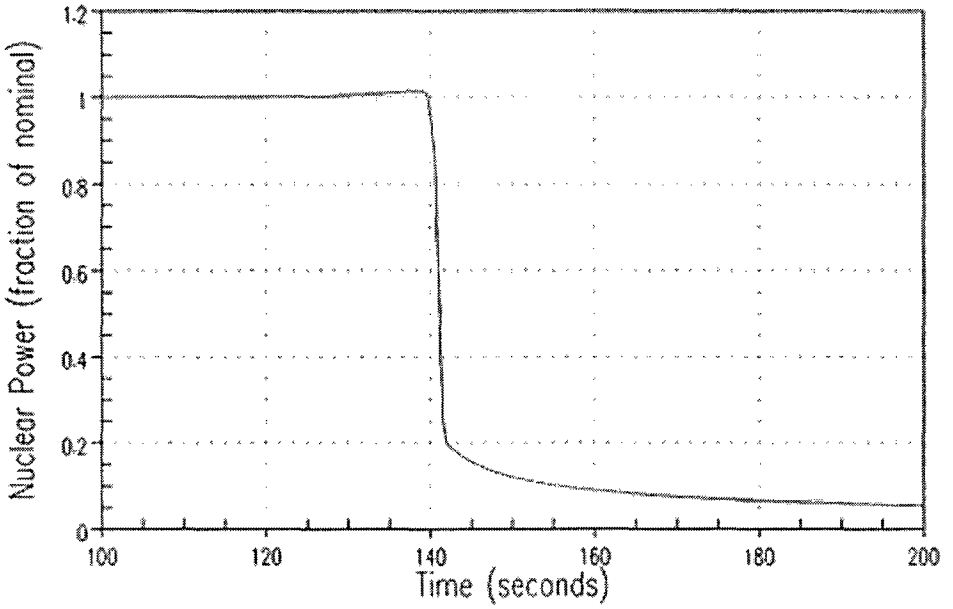


Figure SRXB-2.3.1-2
Pressurizer Pressure vs. Time

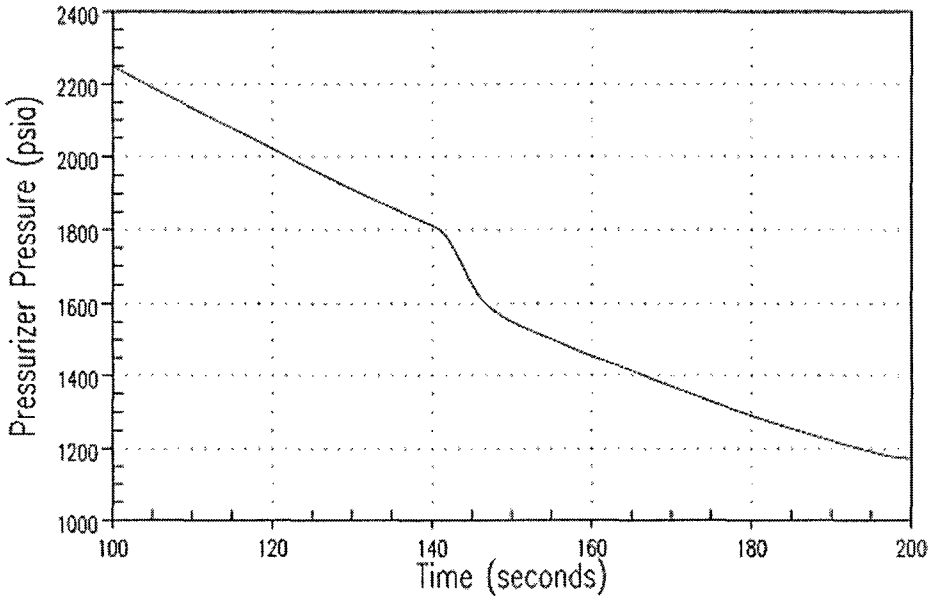


Figure SRXB-2.3.1-3
Reactor Coolant System Average Temperature vs. Time

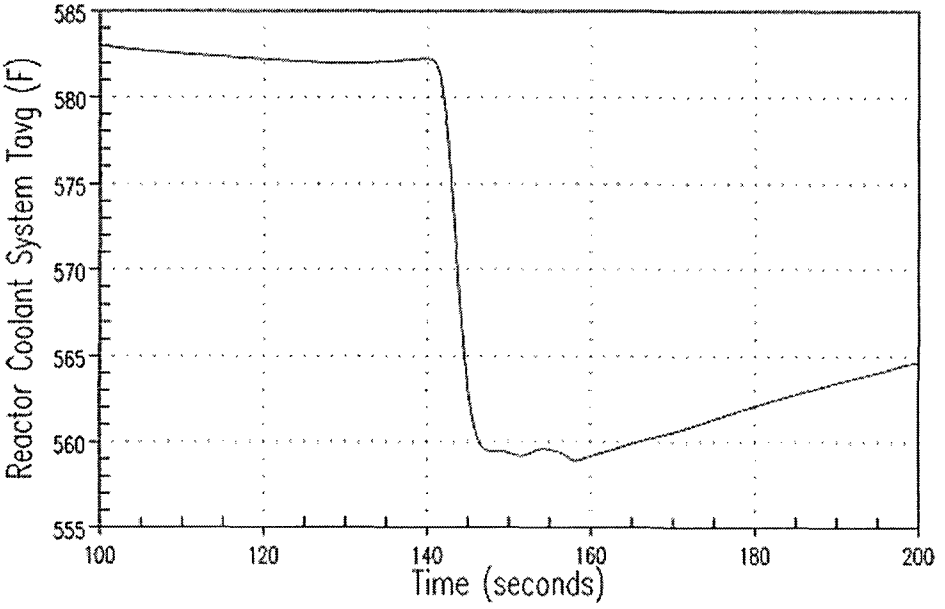


Figure SRXB-2.3.1-4
DNBR vs. Time

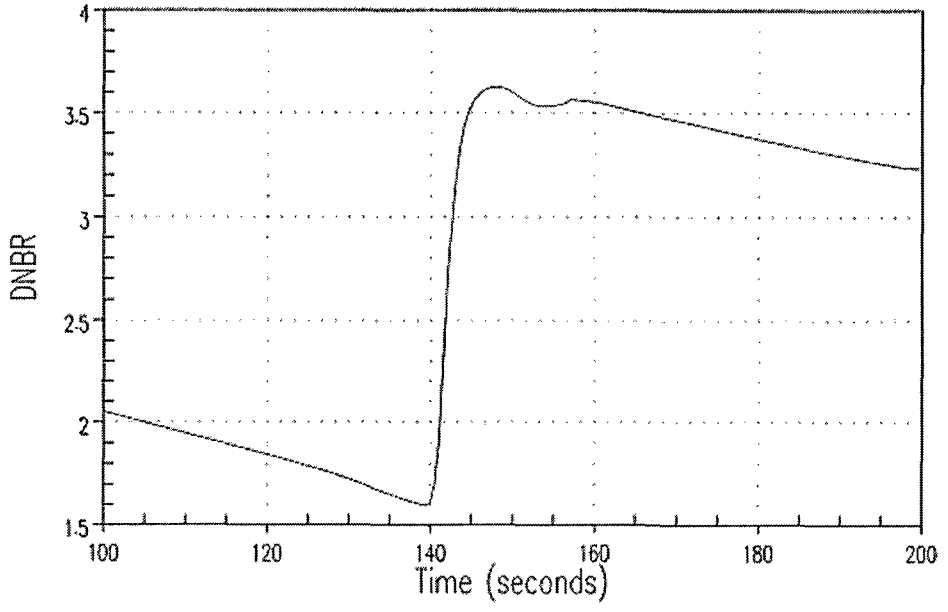
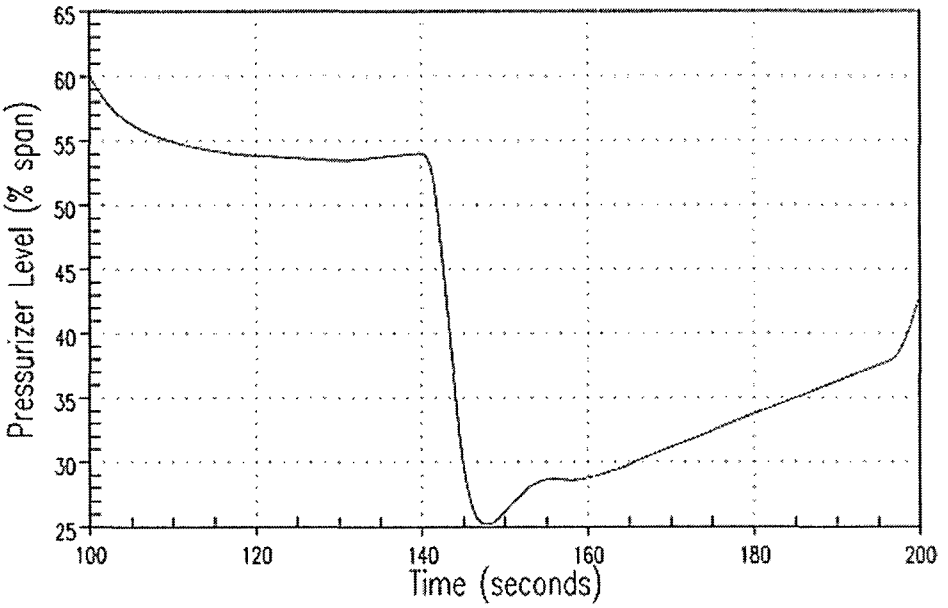


Figure SRXB-2.3.1-5
Pressurizer Level vs. Time



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 S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 1.3 (Part 3)," Accession No. ML11202A174, July 21, 2011.
3. 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.
4. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 2.3 (Part 3)," September 20, 2011.