CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9606180097 DOC.DATE: 96/06/11 NOTARIZED: NO DOCKET #
FACIL:50-250 Turkey Point Plant, Unit 3, Florida Power and Light C 05000250
50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
AUTH.NAME AUTHOR AFFILIATION
HOVEY,R.J. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Provides responses to NRC 960424 RAI supporting technical review of proposed license amends re thermal power uprating.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 12
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-3 LA	1	1	PD2-3 PD	1	1
	CROTEAU,R	1	1			
INTERNAL:	FILE CENTER 01	1	1	NRR/DE/EMCB	1	1
	NRR/DRCH/HICB	1	1	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	NUDOCS-ABSTRACT	1	1
	OGC/HDS3	1	0			
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 12 ENCL 11

FPL

JUN 11 1996

L-96-137
10 CFR §50.36
10 CFR §50.90

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

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Request for Additional Information (RAI)
Proposed License Amendments
Thermal Power Uprate

By letter L-95-245, dated December 18, 1995, Florida Power and Light Company (FPL) submitted a request to amend Turkey Point Units 3 and 4 Operating Licenses and Technical Specifications. In a letter to T. F. Plunkett from R. P. Croteau dated April 24, 1996, the staff requested additional information to support the technical review of the Proposed License Amendments (PLA). The responses to sections A and B were provided in letter L-96-117, dated May 03, 1996. This letter contains the responses to section C and supplemental questions.

Should there be any questions, please contact us.

Very truly yours,

RJH

R. J. Hovey
Vice President
Turkey Point Plant

Attachment

JAH

cc: S. D. Ebneter, Regional Administrator, Region II, USNRC
T. P. Johnson, Senior Resident Inspector, USNRC, Turkey Point
W. A. Passetti, Florida Department of Health and Rehabilitative Services

9606180097 960611
PDR ADOCK 05000250
P PDR

A001 1/1

Response to NRC Request
for Additional Information
Related to Thermal Power Uprating
at Turkey Point Units 3 and 4

MECHANICAL ENGINEERING BRANCH

1. **"In Section 4.4.3, it is stated that stresses and fatigue usage factors for the limiting components of the upper and lower internals were evaluated for changes in RCS conditions due to the uprating program and are within acceptable limits. Provide the limiting internal components which were evaluated for the power uprate conditions. State the acceptable limits with regard to allowable stresses, acceptable criteria, operating conditions, loading combinations, code of record and code edition."**

FPL Response

As part of the uprating scope, all reactor internal components were reviewed, and the limiting components were identified for further evaluation.

The limiting reactor internal components evaluated for the Turkey Point Units 3 and 4 Power Uprating program were:

a) Lower Core Plate
b) Core Barrel
c) Baffle Plates and
d) Baffle/Barrel Region Bolts

Since the Turkey Point reactor internals were designed prior to the introduction of Subsection NG of the ASME Boiler and Pressure Vessel Code Section III, an ASME stress or design report on the reactor internals was not required. Nevertheless, the criteria used is similar to that found in Subsection NG of Section III of the ASME Code (1989 Edition/1990 Addenda). For example, for the 304 stainless steel lower core plate the following allowable stresses (psi) were utilized:

Normal and Upset Conditions

Primary membrane	16100
Primary membrane + bending	24150
Primary membrane + bending + secondary	48300

Faulted Conditions

Primary membrane	38640
Primary membrane + bending	57960

In addition, the cumulative fatigue usage factor was limited to less than 1.0.

The loading combinations, in addition to the loads developed as a result of the thermal performance of the reactor pressure vessel system and components, included loadings due to:

(a) pressure differentials due to coolant flow,
(b) weight of the structure,
(c) superimposed loads from other components,
(d) earthquake (or seismic) loads,
(e) loss of coolant accident (LOCA) loads,
(f) vibratory loads and
(g) preloads.

2. **"In Section 4.5.2, it is stated that the "50% step load decrease" transient was found to increase the ΔP above 2250 psia from the E-Spec (Westinghouse Equipment Specification) value of 120 psi to 128.7 psi (max). The resultant pressure is less than the design pressure and the increase is considered insignificant. Provide the design pressure for the reactor coolant pump."**

<u>FPL Response</u>

For the uprating program, the Reactor Coolant Pumps (RCP) were evaluated for any temperature increases or pressure increases that exceeded the RCP Equipment Specification (E-Spec.). The "50% step load decrease" transient was found to increase the ΔP above 2250 psia from the RCP E-Spec. value of 120 psi to 128.7 psi (max). The resultant pressure of 2378.7 psi is less than the RCP E-Spec. design pressure of 2500 psia, therefore, the ΔP increase is considered insignificant.

3. **"In Section 4.6.2, it is stated that the Uprating Transients are bounded by the original transients except for a) the large step load decrease which now has a higher maximum pressure of 2379 psia, and b) feedwater cycling. Provide the basis for the structural integrity of the control rod drive mechanisms regarding the increase of pressure and temperature transients at the uprated conditions."**

<u>FPL Response</u>

The transients for the Turkey Point Uprating were compared to the original Turkey Point Equipment Specification (E-Spec) values. The Uprating Transients are bounded by the original transients except for the following transients:

- the large step load decrease which now has a higher maximum pressure of 2379 psia, and
- feedwater cycling.

For the two transient cases listed above which were not bounded by the original analysis, the fatigue waiver criteria of the ASME Code, Section NB-3222-4(d) (1989 Edition through 1990 Addenda) has been used. From the Code NB-3222-4(d) fatigue waiver, a significant pressure fluctuation is one which exceeds a pressure difference of 1282 psi. A significant temperature difference is a change of 51.6°F. The new large step load decrease transient increases the pressure difference above 2250 psia from the original CRDM E-Spec value of 120 psi to 128.7 psi (max). The feedwater cycling transient only has a temperature change of 32°F. The transient pressure/temperature changes associated with these uprated condition transients do not qualify as significant fluctuations and hence, any fatigue usage increase is insignificant.

4. **"In Section 4.7.4, it is stated that the applicable load combinations of deadweight, pressure, seismic and thermal loads were checked against the appropriate allowable for the loop piping material. State why the LOCA loads are not considered in the load combinations for calculation of the piping stresses."**

<u>FPL Response</u>

LOCA loads were not included in the load combinations for calculation of pipe stresses for the following reason:

Simple statically calculated P*A forces were originally used in the pipe rupture analysis of the reactor coolant system. Utilizing primary loop leak-before-break (LBB) methodology, there are no postulated breaks in the main loop piping and the P*A force to be used in the pipe rupture analysis is calculated using a branch pipe area rather than a primary loop pipe area.

The ratio of pipe cross sectional flow areas is as follows:

(HOT LEG AREA)/(SURGE LINE AREA) > 9

(COLD LEG AREA)/(ACCUMULATOR LINE AREA) > 7

Based on the above, the conclusion was drawn that pipe rupture loadings would be significantly smaller than the original evaluation, since the pipe rupture force is at least 7 times smaller. The existing analyses were therefore determined to be bounding for the uprating conditions.

5. **"In Section 4.7.3, discuss how the acceptability of the piping and primary components supports was determined while the design basis calculation was not available. State the acceptance criteria of the support loads for each loading condition for the power uprate."**

FPL Response

The primary equipment support analysis/evaluation was accomplished by calculating allowable capacities for the various support elements, and then comparing the loads on the support elements with the calculated capacities. The support capacities were calculated using the detail design drawings of the support structures, allowable stress criteria per Section 5.1.9.1 of the Turkey Point UFSAR, and material minimum yield strengths per ASTM. (The Section 5.1.9.1 stresses are basically the 1963 AISC Specification "working stresses".) The loading combinations considered for the support evaluation are also taken from UFSAR Section 5.1.9.1. The loading and stress criteria for the primary equipment supports are summarized as follows:

Dead + Thermal Loads ≤ Working Stress Limits

Dead + Thermal + Design Earthquake Loads ≤ 1.33 x Working Stress Limits

Dead + Thermal + Maximum Earthquake Loads ≤ 1.5 x Working Stress Limits

Dead + Thermal + Pipe Rupture Loads ≤ Yield Stress

6. **"In Section 4.7.3, provide an evaluation of system components such as valves, RPV nozzles, guides, penetrations and piping suspension devices regarding analysis methods, assumptions and compliance with their Code of record for normal, upset and faulted conditions. The discussion should include the code and edition used for evaluating the stresses, displacements and fatigue usage for power uprate."**

FPL Response

The power uprate did not have a significant impact on any of the system components evaluated for the reactor coolant loop (RCL) piping and supports. The following components were evaluated for potential impact: RCL piping, primary equipment nozzles, primary equipment supports, reactor vessel head vent system piping, and the pressurizer surge line piping. The power uprate program generated changes in two of the basic inputs to the evaluation of the stated system components. Namely, small temperature changes in the primary piping and some minor changes in the system design thermal transients. Because of the early vintage of this plant (early piping codes did not require a fatigue evaluation), the design thermal transients did not impact the design basis analysis of anything but the pressurizer surge line. The surge line had an existing piping fatigue analysis in place to address NRC IE Bulletin 88-11 (Pressurizer Surge Line Thermal Stratification). This surge line fatigue analysis used the design thermal transients as input so that any changes in these transients due to the uprating program needed to be reconciled. The reconciliation was performed and showed only minor change (.002) in the usage factor for the surge line.

The second basic area examined due to the uprating program was the normal operating temperature in some of the components. These small temperature changes (< 2%) resulted in small changes in the thermal expansion analysis of the RCL piping, primary equipment supports, and primary equipment nozzles. Branch line piping attached to the primary loop and the various components that may be associated with the branch piping (valves, nozzles, guides, penetrations, hangers, etc.) have been reviewed. The existing thermal loadings experience little or no change depending on location and have been determined to be acceptable.

The RCL piping evaluation was performed in conformance with the code of record for the plant, ASA B31.1, 1955 edition. The 1973 edition, through 1976 addenda were used for the evaluation because these were the addenda that first placed the criteria into equation form. These two versions of B31.1 will give the same results for this evaluation.

The following load combinations were checked against the appropriate allowable for the loop piping material:

EQN 1	$P + DW \leq S_h$
EQN 2	$P + DW + SSE \leq 1.2S_h$
EQN 3	$THERMAL \leq S_A$

The primary equipment nozzles were compared to the umbrella loadings contained in the appropriate equipment specifications. Those equipment specifications, in turn, meet the appropriate code requirements. The original equipment was analyzed to ASME Section III, 1965 edition. The replacement steam generators met ASME Section III, 1974 edition, through Summer 1976 addenda.

The evaluation of the primary equipment supports was discussed in the response to Question 5.

7. **"In Section 4.9.2, there is no evaluation of fatigue cumulative usage factor (CUF) for the steam generators. Provide such an evaluation including the methodology, assumptions and the calculated CUFs at the critical locations for the power uprate."**

FPL Response

Fatigue usage factors were calculated at the critical pressure boundary component locations as summarized below. In the fatigue evaluation, the stress ranges, which have one of the two load states affected by the uprating condition, were multiplied by the enveloping factors representing the effect of the uprating conditions. These are then used in the calculations for fatigue usage. In all cases, the cumulative fatigue usage factors are within the allowable limit of 1.0.

Summary of Cumulative Fatigue Usage Factors

Component	Fatigue Usage Original Condition	Fatigue Usage Uprated Condition
Tube-to-Tubesheet Weld	0.312	0.324
Tube	0.398	0.398
Feedwater Nozzle	0.753	0.753

8. **"In Section 4.11, discuss the potential for the flow induced vibrations due to the increased flow at the uprated power conditions in the NSSS equipment such as heat exchanger, valves and pumps."**

FPL Response

No fluid velocities increased in the NSSS auxiliary systems, therefore, there was no need to investigate any type of vibration issues associated with the auxiliary equipment. The uprating considered the ability of the auxiliary equipment to operate at higher temperatures instead of having to accommodate increased fluid velocities. Since the auxiliary systems transients are unchanged or bounding, temperature effects were inconsequential and the auxiliary equipment original design basis applies with respect to qualification of equipment.

9. **"In Section 6.3.2, discusses the effects of power uprate on the environmental and dynamic qualification of safety-related equipment with respect to LOCA events, annulus pressurization and jet loads in the context of power uprate."**

FPL Response

The effects of uprating on environmental qualification were assessed and found to be bounded by the previous EQ limits established for the Turkey Point Units. This topic is addressed in Section 3.6.1 of WCAP 14276, Revision 1.

The large break LOCA is the design basis pipe break for inside containment. Regarding dynamic effects as a result of uprating, a plant specific Leak-Before-Break (LBB) analysis was performed, submitted, and approved by the NRC during the course of the uprating design analyses, reference letter dated June 23, 1995. With the use of LBB licensing, it is not necessary to consider the local dynamic effects of a main loop LOCA. Instead, equipment was analyzed for LOCA integrity considering the next most limiting auxiliary line breaks. For Turkey Point, the next most limiting auxiliary line breaks would be the pressurizer surge line on the hot leg and the accumulator line on the cold leg. The LOCA forces analysis was performed incorporating the uprating conditions and the auxiliary line breaks.

The previous analysis of LOCA forces for the steam generator and loop accounted for the effects of RCS loop breaks. The results of the analysis bound LOCA effects due to the smaller, auxiliary line breaks and more than compensate for the impact of uprating parameters.

The reactor vessel/internals forces were analyzed, which had previously credited LBB for those forces, and the resulting structural analyses of the vessel and internals were determined to be acceptable. It is noteworthy that for the reactor vessel and internals that

jet loads are considered for the dynamic analysis only for the reactor vessel inlet and reactor vessel outlet breaks. The Turkey Point Uprating analysis did not have to consider these particular vessel inlet and outlet breaks due to utilization of LBB methodology.

With respect to annulus pressurization, a subcompartment pressure evaluation was performed for the uprating, taking credit for LBB methodology. It was determined that the variations in RCS temperature associated with the uprating, which would impact the subcompartment pressure evaluation, are offset by the benefits (lower predicted releases) obtained from LBB methodology. Therefore, the original design basis subcompartment analysis, which considered releases resulting from double-ended breaks of the primary loop piping, continues to be bounding for the Turkey Point Units.

10. **"In Section 6.2.1, provide an evaluation of the increased MSIV closure dynamic loads on the main steam line piping. State the effects of the increased fluid dynamic loads on the closure capability of the various safety related valves in the plant."**

FPL Response

Main Steam piping inside containment was analyzed to conditions that equal or exceed Thermal Power Uprate conditions. The existing thermal hydraulic analysis utilized a conservative valve closure time. This approach resulted in conservative piping support loads and stresses. An analysis was performed which considered a more realistic main steam isolation valve (MSIV) closing time and the increased steam flow due to Thermal Power Uprate. The results of this analysis concluded that the original thermal hydraulic analysis enveloped the uprated conditions for the Inside Containment piping. A detailed evaluation of the Main Steam piping outside containment was also performed, including the uprated thermal hydraulic effects and the results demonstrate the piping system to be acceptable.

Valves within the Emergency Core Cooling Systems and NSSS Auxiliary Systems are not subjected to increased flows as a result of thermal power uprate, and component flows are within allowable limits. Valves within the Steam and Power Conversions systems (Main Steam, Feedwater, Condensate) have been evaluated for thermal power uprate conditions as required and found acceptable.

11. **"In Sections 6.4.2, specify the code and edition used for the power uprate evaluation of balance-of-plant (BOP) piping and pipe supports including anchorages. List the limiting BOP piping stresses and components with respect to the maximum stresses and safety margin as a result of the power uprate."**

FPL Response

The Construction Code for BOP piping at Turkey Point is ASA B31.1-1955 as modified by the UFSAR. The modifications in the UFSAR address load combinations and allowable stresses with low probability events not explicitly covered in ASA B31.1-1955. For analysis purposes, the 1973 through Winter 1976 Addenda of ANSI B31.1 is used for piping analysis because that is the first Edition that prescribes in equation form the calculation of stresses resulting from occasional loads. For pipe supports, the following Code requirements form the basis for design evaluation: ANSI B31.1-1973 (through Winter 1976 Addenda), AISC Manual 8th Edition and Vendors Catalogs. Anchorages in turn are evaluated in accordance with the criteria of USNRC I&E Bulletin Number 79-02.

For systems with a potential thermal increase of 1% through 5% in temperature over that used in the original analyses, a review of the analyses was conducted and acceptability documented in individual system calculations.

For those systems with a thermal increase exceeding 5 %, evaluations/analyses were performed to document design basis compliance. The evaluations included reviews of pipe stress levels, pipe supports and equipment nozzle loads, as required.

The Component Cooling Water (CCW) system was the most extensively evaluated system to document design basis compliance and resulted in the highest interaction ratio. These evaluations determined a maximum pipe stress interaction ranging from 0.06 to 0.99 for uprate conditions. This interaction for the thermal expansion case in ANSI B31.1 Code, (Equation 13), could be further reduced by considering the combined sustained plus thermal expansion stresses of Equation 14. For the affect on nozzle loads, the range of maximum interaction was determined to be 0.09 to 0.99, the basis of which is a conservative pipe stress criteria. These could be further reduced with detailed review by the equipment vendors. These systems are therefore concluded to be acceptable for uprate conditions.

12. **"It appears the submittal did not address the testing for the power uprate. Discuss how will the licensee ensure an adequate plant operation under the proposed uprated conditions with the increased thermal power, and the changes in temperature, pressure and flow induced dynamic loads."**

FPL Response

Individual modifications required to support plant uprating will receive post-modification testing as required, to ensure proper component operation and system integration at both the current and uprated power level.

As described in the proposed license amendments and the associated licensing report, components, systems and structures have been evaluated and found to be acceptable at

uprated conditions. The evaluations concluded that affected systems and components would not be subjected to any unanticipated loading conditions at the uprated power level.

The initial escalation beyond 2200 MWt will be controlled under a specially prepared procedure. Power level will be raised in small increments. Secondary calorimetric and monitoring of the plant systems and components will provide confirmation of satisfactory performance prior to escalation to the next power level.

I&C BRANCH

1. **On page 5-17 of WCAP-14276, Revision 1, WCAP-12745, Revision 1 is referenced. WCAP-12745, Revision 0 was approved by the NRC in 8/91. Explain the changes to WCAP-12745 between Revisions 0 and 1.**

FPL Response

WCAP-12745, Revision 0, defines the "square root sum of the squares" methodology and calculates the methodology terms utilized for the protection system setpoints for Turkey Point. The methodology utilized in WCAP-12745, Revision 1 is the same as the methodology defined in WCAP-12745, Revision 0. WCAP-12745, Revision 1 has been prepared only to address those RPS and ESFAS functions which require re-evaluation (recalculation of methodology terms) to address the uprating.

The following are the functions re-evaluated:

Overtemperature Delta-T Reactor Trip
Overpower Delta-T Reactor Trip
Reactor Coolant Flow - Low
Steam Generator Water Level - Low-Low, Low
Steam Generator Water Level - High-High
Containment Pressure - High-High
High Steam Line Flow - SI, Steam Line Isolation
Tavg - Low-Low

REACTOR SYSTEMS BRANCH

1. **Provide the steam line break analysis DNBR limit acceptance criteria and the minimum DNBR calculated for the conditions applicable both before and after uprating.**

FPL Response

The pre-uprating DNBR values for the steam line break event are as follows:

Safety Analysis Limit	1.49
Minimum DNBR Typical Cell	1.57
Minimum DNBR Thimble Cell	1.64

The new values which reflect the uprated conditions are as follows:

Safety Analysis Limit	1.45
Minimum DNBR Typical Cell	1.48
Minimum DNBR Thimble Cell	1.57

RCS PIPING AND SUPPORTS:

1. **Page 4-14 of WCAP-14276, Revision 1 refers to an upgraded seismic response spectra that was used for the uprating analysis. What is this upgraded seismic response spectra?**

FPL Response

The RCS piping and supports were analyzed in accordance with IE Bulletin 79-07 using a response spectra circa the original plant design (around 1968). For IE Bulletin 79-02 and IE Bulletin 79-14, floor response spectra were generated for all levels using the original ground response spectrum, and were used for Turkey Point IE Bulletin 79-02 and 79-14 analyses and responses. Final bulletin responses were submitted via FPL letters L-87-383, dated October 22, 1987 and L-90-358, dated October 25, 1990 and are considered complete by the NRC. For RCS piping analyses performed for the uprating, the floor response spectra was reconciled against the original response spectra. The same floor response spectra that was used in Turkey Point's IE Bulletin 79-02 and 79-14 analyses was utilized for the uprating analyses with acceptable results.