PRIORITY 1
(ACCELERATED RIDS PROCESSING)

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9512270307 DOC.DATE: 95/12/18 NOTARIZED: YES 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. Florida Power & Light Co. See Proposed Change
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk) Tech Specs.

SUBJECT: Application for amends to licenses DPR-31 & DPR-41,revising definition of RTP from 2,200 MWt to 2,300 MWt.Proprietary TR WCAP-13719,Rev 2 & nonproprietary TRs WCAP-13718,Rev 2 & WCAP-14276,Rev 1 encl.Proprietary TR WCAP-13719 withheld.

DISTRIBUTION CODE: AP01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 47+594
TITLE: Proprietary Review Distribution - Pre Operating License & Operating R

NOTES: See WCAP-13718P

| | RECIPIENT ID CODE/NAME | COPIES LTTR | COPIES ENCL | RECIPIENT ID CODE/NAME | COPIES LTTR | COPIES ENCL |
|---|---|---|---|---|---|---|
| | PD2-1 LA | 1 | 1 | PD2-1 PD | 1 | 1 |
| | CROTEAU,R | 1 | 1 | | | |
| INTERNAL: | FILE CENTER 01 | 1 | 1 | OGC/HDS3 | 1 | 0 |
| EXTERNAL: | NRC PDR | 1 | 1 NP | | | |

P R I O R I T 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 P1-37 (EXT. 504-2083 ) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 6 ENCL ~~4~~ 5

AA2

FPL

DEC 18 1995

L-95-245
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
Proposed License Amendments -
Thermal Power Uprate

In accordance with 10 CFR 50.90, Florida Power and Light Company (FPL) requests that Appendix A of Facility Operating Licenses DPR-31 and DPR-41 be amended to revise the Turkey Point Units 3 and 4 definition of RATED THERMAL POWER from 2200 Megawatt-thermal (MWt) to 2300 MWt. In order to operate at the higher power level, detailed evaluations of the Nuclear Steam Supply System (NSSS) (including Loss of Coolant Accident (LOCA), non-LOCA, Containment Responses and Dose Consequences), engineered safety features, power conversion, emergency power, support systems and environmental issues have been performed. The results of these evaluations and analyses, where appropriate, as documented in the enclosed report WCAP-14276, confirm that Turkey Point Units 3 and 4 can safely operate at the increased power level.

In addition, FPL proposes changing Technical Specifications (TS) in the following Sections: TS 1.24, Definitions - Rated Thermal Power; TS 2.1.1, Safety Limit - Reactor Core; TS 2.2, Limiting Safety System Settings - Reactor Trip System Instrumentation Setpoints; TS 3/4.2 Power Distribution Limits; TS 3/4.3.2 Engineered Safety Features Actuation System Instrumentation; TS 3/4.4 Reactor Coolant System; TS 3/4.5 Emergency Core Cooling Systems; TS 4.6.2.2 Emergency Containment Cooling System; TS 3/4.7 Plant Systems; TS 6.9.1.7 Core Operating Limits Report and the associated BASES. The proposed revisions to the Technical Specifications are described in detail in the attachments.

The Revised Thermal Design Procedure (RTDP) methodology used in the Thermal Power Uprate Analyses has been previously approved by the NRC in Westinghouse topical report WCAP-11397-P-A. This methodology provides an increase in the Departure from Nucleate Boiling (DNB) margin by convoluting statistically the uncertainties on power, pressure, temperature, flow and the DNB correlation to define the Departure from Nucleate Boiling Ratio (DNBR) limit and core thermal limits. These uncertainties have been calculated for Turkey Point in Westinghouse topical report WCAP-13719, Revision 2, which is provided as an enclosure to this submittal.

FPL has determined that the proposed license amendments do not involve a significant hazards consideration pursuant to 10 CFR 50.92. A description of the amendments request is provided in Attachment 1. The no significant hazards determination in support of the proposed Technical Specification changes are provided in Attachment 2.



A001 1/1
Ltr. Encl. 1 1NP
Change: NRC PDR

[illegible]

Attachment 3 provides the proposed revised Technical Specifications.

Enclosure 1 contains a single copy of Westinghouse Report, WCAP-14276, Revision 1, entitled "Florida Power and Light Company Turkey Point Units 3 and 4 Uprating Licensing Report." WCAP-14276 is a non-proprietary topical report. Enclosure 2 contains a single copy of Westinghouse proprietary report, WCAP-13719, Revision 2, entitled "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Florida Power & Light Company Turkey Point Units 3 and 4" and a single copy of the Westinghouse non-proprietary report, WCAP-13718, Revision 2. Enclosure 3 includes a Westinghouse authorization letter, CAW-95-890, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

Since Enclosure 2 contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-95-890 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety & Regulatory Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

In accordance with 10 CFR 50.91(b)(1), a copy of these proposed license amendments is being forwarded to the State Designee for the State of Florida.

The proposed amendments have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board.

Should there be any questions on this request, please contact us.

Very truly yours,

Robert J. Hovey
Vice President
Turkey Point Plant

RJT

Attachments
Enclosures (3)

[illegible]

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
R. P. Croteau, Project Manager, USNRC, Washington, D.C.

STATE OF FLORIDA )
) ss.
COUNTY OF DADE )

Robert J. Hovey being first duly sworn, deposes and says:

That he is Vice President, Turkey Point Nuclear Plant, of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert J. Hovey

Subscribed and sworn to before me this

18th day of December, 1995.

James E. Knorr
Name of Notary Public (Type or Print)

JAMES E. KNORR
MY COMMISSION # CC 434300
EXPIRES: January 22, 1999
Bonded Thru Notary Public Underwriters

NOTARY PUBLIC, in and for the County of Dade, State of Florida

My Commission expires Jan. 22, 1999
Commission No. CC 434300

Robert J. Hovey is personally known to me.

# ATTACHMENT 1

## DESCRIPTION OF AMENDMENTS REQUEST

9512270307

# DESCRIPTION OF AMENDMENTS REQUEST

## 1.0 INTRODUCTION

Florida Power and Light Company has undertaken a comprehensive program to increase the allowed rated thermal power for Turkey Point Units 3 and 4 from 2200 MWt to 2300 MWt. This program is documented in WCAP-14276, "Florida Power and Light Company Turkey Point Units 3 and 4 Uprating Licensing Report." This topical report provides the following criteria which formed the basis for the Turkey Point thermal power uprate:

1. The review encompassed the aspects of Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) design and operation which are impacted by the power uprating. The scope of this review included the NSSS and BOP safety analyses, the functional capability of the systems for normal and abnormal plant operations and design basis accidents, and the mechanical design of NSSS and BOP components and structures.

2. Safety analyses were performed to Updated Final Safety Analysis Report (UFSAR) quality standards, and evaluated in accordance with criteria and standards that apply to the current Turkey Point Units 3 and 4 operating licenses.

3. Equipment structural designs were evaluated in accordance with the regulatory requirement, codes and standards to which the equipment was originally built.

The proposed plant changes to Turkey Point for the thermal power uprate program involve minor hardware changes.

The proposed changes are addressed and grouped as follows, and considered as such in the Attachment 2, "Determination of No Significant Hazards Consideration":

- License Condition, Rated Thermal Power, Core Safety Limits, Reactor Trip System Instrumentation Trip Setpoints, Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoints, Departure from Nucleate Boiling (DNB) Parameters and Reactor Coolant Pump (RCP) Breaker Position Trip

The comprehensive thermal power uprate program undertaken by FPL resulted in revisions to the definition of RATED THERMAL POWER, core safety limits, DNB parameters, RCP breaker position trip function, reactor trip setpoints and ESFAS trip setpoints. The use of the Revised Thermal Design Procedure (RTDP) methodology, the change in peaking factors, and the other changes associated with the uprating (including increased rated thermal power) have resulted in changes to core safety limits and reactor trip and ESFAS trip setpoints. Analyses that are affected by these proposed changes have been reanalyzed or evaluated and the acceptance criteria as indicated in WCAP-14276 are met. The revised setpoints reflect the analysis input assumptions, and consequently the acceptable analysis results justify their revision.

The RTDP methodology has been previously approved by the NRC in Westinghouse topical report WCAP-11397-P-A (reference 1). By letters L-95-131 and L-95-250 (references 2 and 8), FPL submitted a proposed revision to the Turkey Point Units 3 and 4 Technical Specifications to implement the Revised Thermal Design Procedure. The proposed changes to the Reactor Protection System (RPS) and ESFAS trip setpoints are based on the use of the RTDP methodology as documented in WCAP-13719, Revision 1 (reference 3). In support of the thermal power uprate, Westinghouse generated a Revision 2 to WCAP-13719 (reference 4) to address changes to the RPS and ESFAS setpoints as a result of the increase in power level.

- <u>Available Volume Change for Condensate Storage Tank and the Demineralized Water Storage Tank and Reduced Safety Injection (SI) Pump Discharge Head Requirement</u>

FPL evaluated and modified these parameters to more accurately reflect plant conditions needed to operate at the uprated conditions and to include additional operating margin where appropriate. The analyses and evaluations support the revised TS values.

- <u>Pressurizer and Main Steam Safety Valve Setpoint Tolerances</u>

The pressurizer and main steam safety valve setpoint tolerances were increased. The reanalysis supported the revised values. These values provide flexibility and margin in valve testing.

- **Operation at Reduced Power with Inoperable Main Steam Safety Valves (MSSVs)**

  Since the maximum allowable power range neutron flux high setpoint is based on the nominal Nuclear Steam Supply System (NSSS) power rating of the plant (including reactor coolant pump heat), a reanalysis was performed to establish the revised values.

- **Service Period for Heatup and Cooldown Pressure-Temperature Limit Curves**

  The increased vessel fluence resulting from higher core power required a revision to the service period for the heatup/cooldown curves. The existing curves were verified to be acceptable for the revised service period.

- **Modification to Surveillance Requirement for Emergency Containment Cooling (ECC) System**

  To ensure that the acceptance criteria for containment integrity, component cooling water system operation and post-LOCA long-term containment analyses are met for the uprated power conditions, the TS Surveillance Requirement was modified.

- **Control Room Emergency Ventilation System**

  The Technical Specifications issued with the operating license for Turkey Point Units 3 and 4 did not include any Limiting Condition for Operation associated with the Control Room Emergency Ventilation System. The Atomic Energy Commission (AEC) requested inclusion of such Technical Specifications in 1974 (reference 5) and provided model Technical Specifications for inclusion in the Turkey Point plant licenses. These model Technical Specifications were based on the removal of greater than or equal to 90% radioactive methyl iodide. The Technical Specifications approved by the NRC in April 1982 (reference 6) included a methyl iodide removal efficiency of 90%.

  To assure consistency between testing efficiency and analysis assumptions for post-accident control room doses, the required methyl iodide removal efficiency is being increased to 99%. This increase is consistent with the recommendations of Regulatory Guide 1.52 (reference 7), and supports the analysis for post-accident control room doses. Since this change is clearly conservative, personnel safety will not be adversely impacted.

- **Relocation of $F_Q$ and $F_{\Delta H}$ Limits from Technical Specifications to the Core Operating Limits Report (COLR) and Editorial Corrections**

Generic Letter (GL) 88-16, dated October 4, 1988, encouraged licensees to amend the Technical Specifications related to cycle specific parameters. The GL provided guidance for relocation of certain cycle-dependent core operating limits from a licensee's Technical Specifications to the COLR. This would allow changes to the values of the core operating limits without prior NRC approval (i.e., license amendment), as long as an NRC approved methodology for the parameter limit calculation is followed. The Large Break Loss of Coolant Accident (LBLOCA) was analyzed at the uprated core power level with the increased $F_Q$ and $F_{\Delta H}$ peaking factors. The proposed Technical Specification changes will relocate these cycle specific peaking factors from the Technical Specifications to the COLR. In addition, editorial changes are made to ensure consistency and accuracy within the Technical Specifications.

## 2.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

FPL proposes to change the following Technical Specifications in support of the proposed amendments:

A. License Condition, Rated Thermal Power, Core Safety Limits, Reactor Trip System Instrumentation Trip Setpoints, ESFAS Instrumentation Trip Setpoints, DNB Parameters and RCP Breaker Position Trip

1. License Condition 3.A for Operating License DPR-31 and DPR-41, "Maximum Power Level": Change the License Condition to read as follows:

> "... applicant is authorized to operate the facility at reactor core power levels not in excess of 2300 megawatts (thermal)."

Justification: FPL has evaluated the increase in rated thermal power from 2200 MWt to 2300 MWt. WCAP-14276 documents the results. Reanalysis or evaluation including LBLOCA, Small Break LOCA (SBLOCA), containment response, radiological consequences and non-LOCA analyses, have concluded that Turkey Point Units 3 and 4 are acceptable for operation at the uprated condition. Analysis (when appropriate) and evaluation of NSSS and Balance of Plant (BOP) systems and components have concluded that the systems and components are acceptable for operation at the uprated power level. The pertinent plant aspects for Turkey Point Units 3 and 4 have been reviewed and it has been concluded that both units can safely operate at the increased power level.

2. TS 1.24 Definition of "RATED THERMAL POWER": Change 2200 MWt to 2300 MWt to reflect the new uprated power level.

Justification: FPL has evaluated the increase in rated thermal power from 2200 MWt to 2300 MWt. WCAP-14276 documents the results. Reanalysis or evaluation including LBLOCA, SBLOCA, containment response, radiological consequences and non-LOCA analyses, have concluded that Turkey Point Units 3 and 4 are acceptable for operation at the uprated condition. Analysis (when appropriate) and evaluation of NSSS and BOP systems and components have concluded that the systems and components are acceptable for operation at the uprated power level. The pertinent plant aspects for Turkey Point Units 3 and 4 have been reviewed and it has been concluded that both units can safely operate at the increased power level.

3. <u>TS Figure 2.1-1, "Reactor Core Safety Limits - Three Loops in Operation":</u> Revise Figure 2.1-1 to reflect changes associated with the new operating conditions at the uprated power level.

<u>Justification:</u> New core safety limits have been generated to include the effects of the proposed uprating. Factors such as increased power level, revised flowrate, and increased peaking factors were included in the revised core safety limits. In addition, new overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) protection lines were developed to ensure the core limits are not violated. Reanalysis of those transients, such as rod withdrawal at power, which use OTΔT as the primary reactor trip, has shown that the acceptance criteria as established in WCAP-14276 have been met. The revised core limits include the use of the RTDP methodology. By letters L-95-131 and L-95-250 (references 2 and 8), FPL submitted a proposed revision to the Technical Specifications to include the implementation of the RTDP methodology for Turkey Point.

4. <u>TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints", Functional Unit 5, Overtemperature ΔT,</u> Revise the following values:

(a) Changes to NOTE 1
   (1) $K_1$ from "1.25" to "1.24",
   (2) $K_2$ from "0.016" to "0.017",
   (3) $T'$ from "$\leq$ 574.2 $^{o}$F" to "$\leq$ 577.2 $^{o}$F",
   (4) $K_3$ from "0.0011" to "0.001",
   (5) Limits of percent rated thermal power
      - change "46%" to "50%" (negative),
   (6) ΔT trip setpoint reduction from "1.5%" to "0.0%" (negative), and ΔT trip setpoint reduction from "2.3%" to "2.19%" (positive).

(b) Change to NOTE 2
   (1) change % of instrument span from "0.73%" to "0.84%."

<u>Justification:</u> The revised core safety limits of TS Figure 2.1-1 required changes to the overtemperature ΔT (OTΔT) setpoints. The use of the RTDP methodology and the inclusion of Turkey Point specific instrument uncertainties have resulted in revisions to the other values (such as allowance, etc.) associated with the OTΔT trip function. These revised setpoints were used in the accident analysis to verify their acceptability. The accident analysis criteria established in WCAP-14276 have been met. Modification to inputs for the f(ΔI) function are consistent with the use of the current Relaxed Axial

Offset Control (RAOC) strategy employed at Turkey Point.

The RTDP methodology has been previously approved by the NRC in Westinghouse topical report WCAP-11397-P-A (reference 1). By letters L-95-131 and L-95-250 (references 2 and 8), FPL submitted a proposed revision to the Turkey Point Units 3 and 4 Technical Specifications to implement RTDP. The proposed changes to the RPS and ESFAS trip setpoints are based on the use of the RTDP methodology as documented in WCAP-13719, Revision 1 (reference 3). In support of the thermal power uprate, Westinghouse generated a Revision 2 to WCAP-13719 to address changes to the RPS and ESFAS setpoints as a result of the increase in the power level.

5. <u>TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints", Functional Unit 6 - Overpower ΔT,</u> Change the following:

(a) Changes to NOTE 3
   (1) $K_6$ from "0.00232" to "0.0016", and
   (2) $T''$ from "= Indicated $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 574.2 $^{o}$F)" to "$\leq$ 577.2 $^{o}$F (Nominal $T_{avg}$ at RATED THERMAL POWER)".

(b) Change to NOTE 4
   (1) the % of instrument span from "0.4%" to "0.96%".

<u>Justification:</u> The revised core safety limits of TS Figure 2.1-1 required changes to the overpower ΔT (OPΔT) setpoints. The revised setpoints have been evaluated and found to be appropriate. The use of the RTDP methodology and the inclusion of Turkey Point specific instrument uncertainties have resulted in revisions to the other values (such as allowance, etc.) associated with the OPΔT trip function. These revised setpoints (with uncertainties) were used in accident analyses to verify their acceptability. All accident analysis acceptance criteria continue to be met.

The RTDP methodology has been previously approved by the NRC in Westinghouse topical report WCAP-11397-P-A (reference 1). By letters L-95-131 and L-95-250 (references 2 and 8), FPL submitted a proposed revision to the Turkey Point Units 3 and 4 Technical Specifications to implement RTDP. The proposed changes to the RPS and ESFAS trip setpoints are based on the use of the RTDP methodology as documented in WCAP-13719, Revision 1 (reference 3). In support of the thermal power uprate, Westinghouse generated a Revision 2 to WCAP-13719 (reference 4) to address changes to the RPS and

ESFAS setpoints as a result of the increase in the power level.

6. <u>TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints" Functional Unit 10 - Reactor Coolant Flow-Low,</u> Change the following:

(a) Loop design flowrate from "89,500 gpm" to "85,000 gpm" in ★ footnote.

<u>Justification:</u> The reduced loop flowrate accounts for an analyzed increase in the percentage of steam generator tubes plugged. The effects of the reduced flow have been accounted for in the revised core safety limits and in the accident analysis. Since the criteria indicated in WCAP-14276 have been met, the use of the reduced flowrate is acceptable.

7. <u>TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints" Functional Unit 11 - Steam Generator Water Level - Low-Low and Functional Unit 12 - Steam Generator Water Level - Low,</u> Change the following:

(a) Allowable Value from "8.9%" to "8.15%".

<u>Justification:</u> Turkey Point specific uncertainties were used to modify these values in accordance with Westinghouse NRC-approved setpoint methodology of WCAP-12745 (reference 9). Accident analyses, such as the loss of normal feedwater event, have verified their acceptability since all acceptance criteria indicated in WCAP-14276 have been met.

8. <u>TS Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints"</u>
<u>Functional Unit 1, Safety Injection,</u>
<u>Functional Unit 4, Steamline Isolation, and</u>
<u>Functional Unit 6, Auxiliary Feedwater.</u>
Change the following values:

(a) 1.f (Steam Line Flow--High) Trip Setpoint from "120% to "114%",
1.f (Steam Line Flow--High) Allowable Value from "42.6%" to "44%", and
1.f (Steam Line Flow--High) Allowable Value from "122.6%" to "116.5%".

(b) 4.d. (Steam Line Flow--High) Trip Setpoint from "120%" to "114%",
4.d. (Steam Line Flow--High) Allowable Value from "42.6%" to "44%", and
4.d. (Steam Line Flow--High) Allowable Value from "122.6%" to "116.5%".

(c) 6.b (Steam Generator Water Level--Low-Low) Allowable Value from "8.9%" to "8.15%".

Justification: Turkey Point specific calculations have resulted in changes to various engineered safety features (ESF) functions. Analyses have been performed in accordance with the methodology of WCAP-12745 (reference 9). All affected changes have been found acceptable with respect to accident analysis. Adequate margin is maintained for all trip functions.

9. **TS 3.2.5 "DNB Parameters" and Associated BASES**
Revise the following:

(a) TS 3.2.5a. from "576.6" to "581.2",
(b) TS 3.2.5b. from "2209" to "2200",
(c) TS 3.2.5c. from "277,900" to "264,000", and
(d) Revise the BASES to address the above proposed change and the 12 hour surveillance criteria for RCS flow.

Justification: By applying Turkey Point specific instrument uncertainties and values for $T_{avg}$, Pressure and Flow associated with the uprating, appropriate indicated values of $T_{avg}$, Pressure and Flow have been calculated. Also included in the flow value is a loop design flow reduction of 4500 gpm. The revised DNB parameters have been analytically verified and their effects have been included in the revised core thermal limits of TS Figure 2.1-1.

The RTDP methodology has been approved by the NRC in Westinghouse topical report WCAP-11397-P-A (reference 1). By letters L-95-131 and L-95-250 (references 2 and 8), FPL submitted a proposed revision to the Turkey Point Units 3 and 4 Technical Specifications to implement RTDP. The proposed changes to the DNB parameters are based on the use of the RTDP methodology as documented in WCAP-13719, Revision 1 (reference 3). In support of the thermal power uprate, Westinghouse generated a Revision 2 to WCAP-13719 (reference 4) to address changes to the RPS and ESFAS setpoints as a result of the increase in the core power level.

The Reactor Coolant System (RCS) Flow limit has been reduced to 264,000 gpm. This limit is based on a Thermal Design Flow (TDF) limit of 255,000 gpm (85,000 gpm per loop) which assumes a Steam Generator tube plugging level of 20%. The above limit also includes a 3.5% RCS flow measurement uncertainty.

The Surveillance Requirement associated with the RCS flow rate has been clarified to more closely resemble the original intent of the surveillance. The indicated RCS flow rate must be greater than the TDF plus instrument channel uncertainties (including parallax errors). The proposed surveillance also takes advantage of the additional margin gained.

**10.** <u>TS BASES Page B 2-7, Reactor Coolant Pump Breaker Position Trip</u> - Delete the following sentence:

> "No credit was taken in the accident analyses for operation of these trips."

<u>Justification:</u> Credit is taken for the RCP Breaker Position Trip. The Turkey Point plant underfrequency signal does not directly result in a reactor trip, but rather it trips the RCP breakers which in turn trip the reactor. Therefore, RCP Breaker Position Trip is credited and assumed for reactor trip.

B. Available Volume Change for Condensate Storage Tank and the Demineralized Water Storage Tank and Reduced Safety Injection (SI) Pump Discharge Head Requirement

1. TS 3.7.1.3 "Condensate Storage Tank" and Associated BASES, TS 3.7.1.6 "Standby Steam Generator Feedwater System" and Associated BASES.

Revise the following:

(a) TS 3.7.1.3 - change "185,000" to "210,000",
(b) TS 3.7.1.3 - change "370,000" to "420,000",
(c) TS 3/4.7.1.3 ACTION and SURVEILLANCE REQUIREMENTS statements - substitute the word "indicating" for the word "containing", and add the word "indicated,"
(d) TS 3.7.1.6 - change "60,000" to "135,000", and
(e) TS 3.7.1.6, ACTION c. - add the word "indicated."

Justification: Operation at the uprated power conditions required that tanks such as the condensate storage tank (CST) and the demineralized water storage tank (DWST) have minimum volumes available so that their design safe shutdown functions can be met. Analyses have verified that the proposed minimum volumes meet the requirements for their design safe shutdown functions.

Thermal power uprate analysis results show increased emergency feedwater flow and volume delivery requirements. This requires increased CST and DWST reserve volumes and a corresponding change in the Technical Specifications. This also presents an opportunity to clarify existing text by using consistent terminology and distinguishing between the minimum usable volume required for delivery and the minimum indicated volume, which includes allowance for instrument uncertainties and a portion deemed unusable.

The proposed changes to the TS BASES section for DWST and CST include a discussion of the minimum indicated volume and the basis for which the Technical Specification value is developed.

2. <u>TS 4.5.2 - "Emergency Core Cooling System," and Associated BASES.</u>
Make the following changes:

(a) TS 4.5.2c.1) change "1126 psid" to "1083 psid" (for normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), and change "1156 psid" to "1113 psid" (for Unit 3 SI pumps aligned to Unit 4 RWST).

<u>Justification:</u> Analysis has supported a reduction in the pump discharge head. This reduction in required head will provide margin for meeting SI pump test acceptance criteria and also provide available margin should pump degradation occur.

## C. <u>Pressurizer and Main Steam Safety Valve Setpoint Tolerances</u>

1. <u>TS 3.4.2.1 - "Safety Valves",</u>
<u>TS 3.4.2.2 - "Safety Valves",</u>
<u>TS Table 3.7-2 - "Steam Line Safety Valves Per Loop", and Associated BASES for TS 3/4.4.2 and 3/4.7.1.1.</u>
Change the following:

(a) TS 3.4.2.1 and 3.4.2.2: Increase the pressurizer safety valve tolerances from "±1%" to "+2%, - 3%",
(b) TS 3.4.2.1, 3.4.2.2: Add the footnote "All valves tested must have "as-left" lift setpoints that are within 1% of the lift setting value."
(c) TS Table 3.7-2: Change MSSVs lift setting tolerances from "± 1%" to "± 3%",
(d) TS Table 3.7.2: Add the footnote "All valves tested must have "as-left" lift setpoints that are within ± 1% of the lift setting value listed in Table 3.7-2."

<u>Justification:</u> The use of increased tolerances has been accommodated in the accident analysis in WCAP-14276. Valve operability is not affected by these proposed changes. The valves will continue to perform their intended safety functions.

FPL proposes the use of the "± 3%" tolerance for the "as-found" acceptance criteria for the main steam safety valves and "+2%, -3%" tolerance for the "as-found" acceptance criteria for the pressurizer safety valves. The proposed changes require that the pressurizer and main steam safety valve setpoints be restored to within ± 1% of their nominal setpoints following testing.

D. Operation at Reduced Power with Inoperable Main Steam Safety Valves (MSSVs)

1. TS Table 3.7-1 - "Steam Line Safety Valves Per Loop". Change the following maximum allowable power level with inoperable main steam line safety valves (MSSV):

(a) For one inoperable MSSV (per steam line), reduce the maximum allowable power level from "56%" to "53%", and
(b) For two inoperable MSSV (per steam line), reduce the maximum allowable power level from "35%" to "33%".

Justification: Turkey Point's Technical Specification 3/4.7.1.1 is based on an algorithm which defines the maximum power level at which the plant is allowed to operate as a function of the available MSSV relief capacity. This algorithm as identified in the BASES Section for TS 3/4.7.1.1 is used to calculate the maximum allowable power level specified in TS Table 3.7-1. Included in this algorithm is the nominal NSSS power rating of the plant, which is currently assumed as 2208 MWt (including reactor coolant pump heat). As a result of the thermal power uprate, the nominal Nuclear Steam Supply System (NSSS) power rating of the plant (including reactor coolant pump heat) will be changed from 2208 MWt to 2308 MWt.

E. Service Period for Heatup and Cooldown Pressure-Temperature Limit Curves

1. TS Figure 3.4-2 - "RCS Heatup Limitations (60 $^{O}$F/Hr) - Applicable Up To 19 EFPY",
TS Figure 3.4-3 - "RCS Heatup Limitations (100 $^{O}$F/Hr) - Applicable Up To 19 EFPY", and
TS Figure 3.4-4 - "RCS Cooldown Limitations (100 $^{O}$F/Hr) - Applicable Up To 19 EFPY", and Associated BASES. Change the following:

(a) Service period from "20 EFPY" to "19 EFPY".

Justification: The changes in neutron fluence resulting from the proposed Turkey Point Units 3 and 4 thermal power uprating have been evaluated and their impact on reactor vessel integrity has been determined. Fluence projections on the vessel were calculated for the uprated power level and calculations were performed for the revised fluences on vessel integrity. It was determined that the current heatup and cooldown curves contained in the Technical Specifications are applicable for the uprated conditions for a service period to 19 EFPY.

F. Modification to Surveillance Requirement for Emergency Containment Cooling System

1. TS 4.6.2.2 - "Emergency Containment Cooling System" and Associated BASES. Revise TS 4.6.2.2b.1) to read:

"Verifying that two emergency containment cooling units start automatically on a safety injection (SI) test signal, and ..."

Justification: The proposed configuration will ensure that the acceptance criteria for containment integrity, component cooling water system operation and post-LOCA long term containment response are met. The auto-start of all three ECC units on an SI signal is no longer required. For containment integrity safety analyses and component cooling water system thermal analyses, a maximum of two ECCs can receive an automatic start signal following generation of an SI signal without exceeding component cooling water system temperature limits during injection and/or recirculation phases of a LOCA. To support post-LOCA long-term containment pressure/temperature analyses, a minimum of one ECC is required to start immediately with a second ECC unit starting within 24 hours following the event. The third (swing) ECC is required to be operable for manual starting following a postulated LOCA event for containment pressure/temperature suppression. This has been confirmed by evaluations.

The proposed changes to the TS BASES address the proposed plant configuration.

G. Control Room Emergency Ventilation System

1. TS 4.7.5c.2) "Control Room Emergency Ventilation System." Revise the methyl iodide removal efficiency from "90%" to "99%."

Justification: The Technical Specifications issued with the operating license for Turkey Point Units 3 and 4 did not include any Limiting Condition for Operation associated with the Control Room Emergency Ventilation System. The Atomic Energy Commission (AEC) requested inclusion of such Technical Specifications in 1974 (reference 5) and provided model Technical Specifications for inclusion in the Turkey Point plant licenses. These model Technical Specifications were based on the removal of greater than or equal to 90% radioactive methyl iodide. The Technical Specifications approved by the NRC in April 1982 (reference 6) included a methyl iodide removal efficiency of 90%.

To assure consistency between testing efficiency and analysis assumptions for control room doses post-accident, the required methyl iodide removal efficiency is being increased to 99%. This increase is consistent with the recommendations of Regulatory Guide 1.52 (reference 7), and supports the analysis for control room doses post-accident. Since this change is clearly conservative, personnel safety will not be adversely impacted.

**H. Relocation of $F_Q$ and $F_{\Delta H}$ Limits from Technical Specifications to the Core Operating Limits Report (COLR) and Editorial Corrections**

1. **TS 3.2.2 - "Heat Flux Hot Channel Factor",
TS 3.2.3 - "Nuclear Enthalpy Rise Hot Channel Factor",
TS 6.9.1.7 - "Core Operating Limits Report" (COLR),
And Associated BASES for TS 2.1.1, 3/4.2.1, 3/4.2.2 and 3/4.2.3.**
Revise the following:

(a) TS 3.2.2 - Relocate the specific $F_Q$ value to the COLR,

(b) TS 3.2.3 - Relocate the specific values for $F_{\Delta H}$ and the Power Factor multiplier to the COLR,

(c) TS 6.9.1.7 -

(1) Add the appropriate wording to reflect the inclusion of $F_Q(Z)$ and $F_{\Delta H}$ in the COLR,

(2) Add the following statement - "4. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3", and

(3) Update references to be consistent with the current analyses.

Justification: Generic Letter 88-16, dated October 4, 1988, encouraged licensees to amend the TS related to cycle specific parameters. The GL provided guidance for relocation of certain cycle-dependent core operating limits from a licensee's Technical Specifications to the COLR. This would allow changes to the values of the core operating limits without prior NRC approval (i.e., license amendment), as long as an NRC approved methodology for the parameter limit calculation is followed. The proposed TS changes will relocate cycle specific parameter limits from the TS to the COLR. In accordance with the recommendations of GL 88-16 , FPL proposes the addition of the Heat Flux Hot Channel Factor [$F_Q(Z)$], and Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) to the COLR.

These parameters are added to the Axial Flux Difference, Rod Bank Insertion Limits and the K(Z) curve currently included in the COLR in accordance with TS 6.9.1.7.

The Core Operating Limit Report (COLR) is an appropriate document for compiling $F_Q$ and $F_{\Delta H}$. The increased peaking factors ($F_Q$ = 2.35 and $F_{\Delta H}$ = 1.64) will be maintained in the COLR. These values have been justified based on the thermal power uprate analysis.

The proposed changes to the TS BASES address the relocation of the peaking factors from the Technical Specifications to the COLR.

2. TS BASES 3/4.7.1.4 - Correct the abbreviation for "Dose Conversion Factor" to read "DCF".

   Justification: The proposed editorial change is made to ensure consistency within the Technical Specifications.

3. TS BASES Page B 3/4 2-4 - Delete the following sentence: "The current limit is valid for tube plugging levels up to 5%".

   Justification: The analysis in WCAP-14276 assumed up to 20% steam generator tube plugging level for the Small Break LOCA and non-LOCA Analyses, while the Large Break LOCA analysis performed using the BASH Evaluation Model assumed a 5% steam generator tube plugging level. Upon NRC approval of the Westinghouse Best Estimate Large Break LOCA (BELOCA) methodology, FPL intends to reanalyze the Large Break LOCA event using the BELOCA methodology and assuming a 20% tube plugging level. This sentence is deleted since the statement is unnecessary in the context of the Technical Specifications BASES.

## 3.0 REFERENCES

1. WCAP-11397-P-A, "Revised Thermal Design Procedure", dated April 1989.

2. Letter, T. F. Plunkett (FPL) to USNRC, "Proposed License Amendments - Implementation of the Revised Thermal Design Procedure and Steam Generator Water Level Low-Low Setpoint", L-95-131, dated May 5, 1995.

3. WCAP-13719, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Florida Power & Light Company Turkey Point Units 3 and 4", dated January 1995.

4. WCAP-13719, Revision 2, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Florida Power & Light Company Turkey Point Units 3 and 4", dated September 1995.

5. Letter, G. Lear (AEC) to R. E. Uhrig (FPL), "Installed Filter Systems", dated December 10, 1974.

6. Letter, M. Grotenhius (USNRC) to R. E. Uhrig (FPL), "Issuance of License Amendments 83 and 77 for Operating Licenses DPR-31 and DPR-41", dated April 9, 1982.

7. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", dated March 1978.

8. Letter, T. F. Plunkett (FPL) to USNRC, "Proposed License Amendments - Implementation of the Revised Thermal Design Procedure and Steam Generator Water Level Low-Low Setpoint", L-95-250, dated September 28, 1995.

9. WCAP-12745, "Westinghouse Setpoint Methodology for Protection Systems for Florida Power & Light Company Turkey Point Units 3 and 4 Thermal Uprate Project", dated December 1995.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

## DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

## 1.0 BACKGROUND

Florida Power and Light Company (FPL) proposes to uprate the core thermal output of Turkey Point Units 3 and 4 from 2200 MWt to 2300 MWt.

Detailed evaluations of the Nuclear Steam Supply System (NSSS) (including Loss of Coolant Accident (LOCA), non-LOCA, Containment Responses and Dose Consequences), engineered safety features (ESF), power conversion, emergency power, support systems and environmental issues have been performed. A thorough review and assessment of applicable criteria for NSSS systems and components (i.e., steam generator, reactor vessel, pressurizer, etc.) have concluded that the applicable criteria are met for these system components. The results of these evaluations and analyses (when appropriate), show that Turkey Point Units 3 and 4 can safely operate at the increased power level and the conditions associated with the increased power level.

Certain Technical Specifications require revision to accommodate the thermal power uprate. All proposed design and Technical Specification changes have been verified to be acceptable at the proposed uprated conditions.

The proposed Technical Specification changes are divided into eight groups. Explanations of the rationale for the group categorization will follow. This submittal includes a "No Significant Hazards" evaluation for each of the eight groups.

The groupings are as follows:

*1)* ***Technical Specification changes associated with the uprated power level, the revised core safety limits, revised DNB parameters, Engineered Safety Features Actuation System (ESFAS) and reactor trip setpoint changes, and Reactor Coolant Pump (RCP) Breaker Position Trip, will be evaluated together. The safety of these proposed changes are verified by the accident analyses that were completed in support of the uprated power.***

Parameters associated with the uprated conditions provide direct input into accident analyses. The revised core safety limits and revised DNB parameters are key accident analysis parameters necessary to ensure analysis criteria are met. The flow reduction included in the DNB parameters is also direct analysis input. The acceptability of the revised ESFAS and reactor trip setpoints (including OTΔT and OPΔT) are verified by these same accident analyses. A single "No Significant Hazards Evaluation" will be provided for these proposed changes.

2) ***Technical Specification changes associated with reducing the SI pump discharge head requirement and increasing usable volume requirements for the Demineralized Water Storage Tank (DWST) and the Condensate Storage Tank (CST) will be addressed together.***

Operation at the uprated power level required FPL to ensure that systems necessary for the safe operation of Turkey Point Units 3 and 4 meet their intended function. During the comprehensive review, it was determined that the lower limits for several storage tank volumes should be increased. The systems in which these changes have been proposed have been verified to be acceptable for operation at the increased power level.

The reanalysis effort afforded FPL the opportunity to provide margin in the required head for delivered SI flow. This decrease in head has been found acceptable from a systems performance perspective and provides SI pump testing margin as well as margin to account for any possible future pump degradation.

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

3) ***Technical Specification changes associated with pressurizer and main steam safety valve setpoint tolerance increases will be assessed together.***

The reanalysis effort associated with the proposed power uprate provided an opportunity for FPL to expand the tolerances on the pressurizer and main steam safety valves. These valves have been evaluated at the new tolerances and it has been confirmed that they will be capable of performing their intended functions. The expanded tolerances will facilitate in-situ testing.

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

4) ***Technical Specification changes associated with operation at reduced power with inoperable main steam safety valves will be assessed separately.***

Turkey Point's Technical Specification 3/4.7.1.1 is based on an algorithm which defines the maximum power level at which the plant is allowed to operate as a function of the available MSSV relief capacity. This algorithm as identified in the BASES Section for TS 3/4.7.1.1 is used to calculate the maximum allowable power level specified in TS Table 3.7-1. Included in this algorithm is the nominal NSSS power rating of the plant, which is currently assumed as 2208 MWt (which includes reactor coolant pump heat). As a result of the thermal power uprate, the nominal

Nuclear Steam Supply System (NSSS) power rating of the plant (including reactor coolant pump heat) will be changed from 2208 MWt to 2308 MWt.

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

5) ***Technical Specification changes associated with the service period for heatup and cooldown pressure-temperature limit curves will be assessed together.***

The changes in neutron fluence resulting from the proposed Turkey Point Units 3 and 4 uprating have been evaluated and their impact on reactor vessel integrity has been determined. Fluence projections on the vessel were calculated for the uprated power level and calculations were performed for the revised fluences on vessel integrity. It was determined that the current heatup and cooldown curves contained in the Technical Specifications are applicable for the uprated conditions to 19 Effective Full Power Years (EFPY).

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

6) ***The Surveillance Requirement change for the emergency containment cooling unit operability will be handled separately since this is a design change that required extensive evaluations.***

During the systems review for the proposed thermal power uprating for Turkey Point Units 3 and 4, FPL has determined that modifications to the actuation logic for the Emergency Containment Cooling (ECC) units are appropriate. Previously, all 3 ECC units were automatically started on a safety injection (SI) signal. The revised design requires only two ECCs to automatically start on an SI signal. All analyses and evaluations to support this change have been completed and the acceptance criteria have been met.

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

7) ***Technical Specification change associated with the methyl iodide removal efficiency in the Control Room Emergency Ventilation System will be assessed separately.***

The Technical Specifications issued for the operating licenses for Turkey Point Units 3 and 4 did not include any Limiting Condition for Operation associated with the Control Room Emergency Ventilation System. The Atomic Energy Commission (AEC) requested inclusion of such Technical Specifications in 1974 and provided model Technical Specifications for inclusion in the Turkey Point plant licenses. These model Technical Specifications were based on the removal of greater than or equal to 90% radioactive methyl iodide. The Technical Specifications approved by the NRC in April 1982 included a methyl iodide removal efficiency of 90%.

To assure consistency between testing efficiency and analysis assumptions for control room doses post-accident, the required methyl iodide removal efficiency is being increased to 99%. This increase is consistent with the recommendations of Regulatory Guide 1.52, and supports the analysis for control room doses post-accident. Since this change is clearly in the conservative direction, personnel safety will not be adversely impacted.

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

8) ***All LOCA related changes dealing with the peaking factor increase, COLR changes, Evaluation Model references, and relocation of peaking factors from the Technical Specifications and subsequent inclusion in the COLR will be included in one "No Significant Hazards" evaluation. All of the items are closely related since the LOCA analysis is performed to ensure peaking factor acceptability.***

A revised Large Break LOCA analysis has been performed for the thermal power uprating for Turkey Point Units 3 and 4 to 2300 MWt (core power). This analysis was based on the NRC approved BASH Evaluation Model, with modifications to improve the code stream. The Large Break LOCA Analysis was performed to ensure that the acceptance criteria of 10 CFR 50.46 are met for the uprated conditions. Technical Specification 6.9.1.7 will be modified to support the relocation of cycle specific limits from the Technical Specifications to the COLR and to include the latest references to the methodology used in the analysis.

These proposed Technical Specification changes will be evaluated in a single "No Significant Hazards" consideration.

## 2.0 DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

## 2.1 LICENSE CONDITION, RATED THERMAL POWER, CORE SAFETY LIMITS, REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS, ESFAS INSTRUMENTATION TRIP SETPOINTS, DNB PARAMETERS AND RCP BREAKER POSITION TRIP

### Description of Proposed License Amendments

Revised core thermal limits reflected in TS Figure 2.1-1 were generated employing the Revised Thermal Design Procedure (RTDP) methodology and including the effects of the uprated power conditions and reduced RCS flow. Overtemperature $\Delta T$ and Overpower $\Delta T$ reactor trip setpoints and associated uncertainties were calculated based on the new core safety limits. A review of the Turkey Point Updated Final Safety Analysis Report (UFSAR) was performed to determine those events sensitive to changes in the Overtemperature $\Delta T$ and Overpower $\Delta T$ reactor trip setpoints. Each of the events [i.e., Rod Withdrawal at Power, Boron Dilution, and Loss of Load] have been analyzed, to determine if the various acceptance criteria were met. In all cases, the acceptance criteria were met, and therefore the margin of safety is maintained. The revised ESF and reactor trip setpoints were verified to be acceptable.

The DNB parameters were modified to reflect Turkey Point's specific uncertainties, and to be consistent with values used in accident analyses. The flow value includes a flow reduction. This flow reduction is supported by accident analyses.

### INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

**Discussion**

**(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of all plant systems are unaffected. The overtemperature ΔT and overpower ΔT reactor trip functions as well as ESFAS functions are part of the accident mitigation response and are not accident initiators. All proposed changes have been assessed and no design and analysis acceptance criteria have been exceeded. Therefore the probability of occurrence previously evaluated is not affected.

The proposed changes do not affect the integrity of the fission product barriers utilized for mitigation of dose consequences as a result of an accident. Dose consequences were reviewed and reanalyzed (as needed) and found acceptable. Therefore, the probability or consequences of an accident previously evaluated are not significantly increased.

**(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because their effects do not affect accident initiation sequences. All new operating configurations have been evaluated and no new limiting single failures have been identified. In addition, no new failure modes have been identified. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated has been created as a result of these revisions.

**(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The proposed changes do not involve a reduction in a margin of safety because the margin of safety associated with these parameters as verified by the results of the accident analyses, are within acceptable limits. All transients impacted have been analyzed and have met the applicable accident analyses acceptance criteria (e.g., DNBR, RCS pressure, secondary side pressure, etc.). The margin of safety required for each affected safety analysis is maintained. The adequacy of the revised Technical

Specifications values has been confirmed such that there is no reduction in the margin of safety. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above discussion, it has been determined that the proposed changes to the Technical Specifications are acceptable. These revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated; they neither create the possibility of a new or different kind of accident from any accident previously evaluated, nor involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.2 AVAILABLE VOLUME CHANGE FOR CONDENSATE STORAGE TANK (CST) AND DEMINERALIZED WATER STORAGE TANK (DWST), AND REDUCED SAFETY INJECTION (SI) PUMP DISCHARGE HEAD REQUIREMENT.

**Description of Proposed License Amendments**

The required available volumes for the CST and the DWST required an increase to support the uprated conditions. Evaluations have concluded that the revised values will ensure that the tanks continue to perform their intended safe shutdown functions.

FPL has reevaluated the SI pump performance. To provide further margin for pump testing, the discharge head at each present flow point was decreased from its present analysis value by 100 feet of water. This will provide margin in testing to the SI pump acceptance criteria as well as provide margin for any possible future pump degradation.

All acceptance and performance criteria continue to be met for the systems and components involved with these changes.

## INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

**Discussion**

**(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The revised tank volumes and SI head requirements have been evaluated with respect to system performance and analysis impacts. All accident analysis acceptance criteria continue to be met. The design function of all affected systems have been reviewed and all system design criteria continue to be met. The structural and functional integrity of the affected systems are unaffected. These changes are not initiators for any accident and therefore the probability of occurrence of an accident previously

evaluated has not increased.

The proposed changes do not affect the integrity of the fission product barriers for mitigation of dose consequences. All dose consequences remain well within the 10 CFR 100 limits. Therefore there is no increase in the probability or consequences of an accident previously evaluated.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The revised tank volumes and SI head requirements do not create the possibility of a new or different kind of accident from any accident previously evaluated because these modifications do not affect accident initiation sequences. No new operating configuration is being imposed by the adjustments that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated have been created as a result of these revisions.

(3) **Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The proposed changes do not involve a reduction in a margin of safety because the margin of safety associated with these parameters, as verified by the results of the accident analyses and system evaluations, are within acceptance limits. The margin of safety required for each affected safety analysis is maintained. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above discussion, it has been determined that the proposed changes to the Technical Specifications are acceptable. These revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated; they neither create the possibility of a new or different kind of accident from any accident previously evaluated, nor involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.3 PRESSURIZER AND MAIN STEAM SAFETY VALVE SETPOINT TOLERANCES

### Description of the Proposed License Amendment

Initial pressurizer and main steam safety valve tolerances have been evaluated as part of the thermal power uprating program for Turkey Point Units 3 and 4. It was concluded that expanding the range of tolerances was acceptable from both valve design and function effects as well as safety analysis effects. The valves will continue to function as designed and afford overpressure protection to limit pressure transients to equal to or less than 110% of design pressure. All accident analysis criteria continue to be met.

### INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

### Discussion

**(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The revised tolerances for main steam safety valves and pressurizer safety valves do not involve an increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analytical acceptance criteria, such as those applicable to primary and secondary side pressures to be exceeded. The structural and functional integrity of the valves are unaffected by this proposed change. The tolerance changes do not initiate or cause initiation of any transient. Therefore, the probability of occurrence previously evaluated is not affected.

The changes do not affect the integrity of the fission product barriers utilized for dose consequence mitigation. Therefore, the probability or consequences of an accident previously evaluated is not increased.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The revised valve tolerances do not create the possibility of a new or different kind of accident from any accident previously evaluated because the tolerances do not affect accident initiation sequences. No new operating configuration is being imposed by the tolerances that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated have been created as a result of these revisions.

(3) **Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The changes to valve tolerances do not involve a reduction in a margin of safety because the margin of safety associated with the MSSVs and the pressurizer safety valves, as verified by the results of the accident analyses and valve evaluations, are within acceptable limits. Transients impacted by this change have been analyzed and have met the applicable accident analyses acceptance criteria, such as those applicable to primary and secondary side pressure. The margin of safety required for each affected safety analysis is maintained. This conclusion is not changed by the valve tolerances for the main steam safety valves and the pressurizer safety valves. Therefore, the changes do not involve a significant reduction in the margin of safety.

Based on the above discussion, it has been determined that the proposed changes to the tolerances for MSSVs and pressurizer safety valves are acceptable. These revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated; they neither create the possibility of a new or different kind of accident from any accident previously evaluated, nor involve a significant reduction in the margin of safety. Therefore, it is concluded that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.4 OPERATION AT REDUCED POWER WITH INOPERABLE MAIN STEAM SAFETY VALVES (MSSVs)

### Description of Proposed License Amendments

Turkey Point's Technical Specification 3/4.7.1.1 is based on an algorithm which defines the maximum power level at which the plant is allowed to operate as a function of the available MSSV relief capacity. This algorithm as identified in the BASES Section for TS 3/4.7.1.1 is used to calculate the maximum allowable power level specified in TS Table 3.7-1. Included in this algorithm is the nominal Nuclear Steam Supply System (NSSS) power rating of the plant, which is currently assumed as 2208 MWt (which includes reactor coolant pump heat). As a result of the thermal power uprate, the nominal Nuclear Steam Supply System (NSSS) power rating of the plant (including reactor coolant pump heat) will be changed from 2208 MWt to 2308 MWt.

### INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendment.

### Discussion

(1) **Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed maximum allowable power level values will ensure that the secondary side steam pressure will not exceed 110 percent of the design pressure following a Loss of Load/Turbine Trip event, when one or more main steam safety valves (MSSVs) are declared inoperable. The proposed change will not impact the classification of the Loss of Load/Turbine Trip event as a Condition II probability event (faults of moderate frequency) per ANSI - N18.2, 1973. Accordingly, since the proposed maximum allowable power level will maintain the capability of the MSSVs to perform their pressure relief function associated with a Loss of Load/Turbine Trip event, there will be no effect on the probability or consequences of an accident previously evaluated.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the configuration of any plant equipment, and no new failure modes have been defined for any plant system or component. The proposed maximum allowable power level as specified in TS Table 3.7-1 will improve the capability of the MSSVs to perform their pressure relief function to ensure the secondary side steam pressure does not exceed 110 percent of design pressure following a Loss of Load/Turbine Trip event. Therefore, since the function of the MSSVs is improved by the proposed changes, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

(3) **Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The proposed changes to the Technical Specifications do not involve a significant reduction in a margin of safety. The algorithm methodology used to calculate the maximum allowable power level is conservative and bounding since it is based on a number of inoperable MSSVs per loop; i.e., if only one MSSV in one loop is out of service, the required action to reduce power to the maximum allowable power level would be the same as if one MSSV in each loop were out of service. Another conservatism with the algorithm methodology is with the assumed minimum total steam flow rate capability of the operable MSSVs. The assumption is that if one or more MSSVs are inoperable per loop, the inoperable MSSVs are the largest capacity MSSVs, regardless of which capacity MSSVs are actually inoperable. Therefore, since the maximum allowable power level calculated for the proposed changes using the algorithm methodology are more conservative and ensure that 110 percent of secondary side steam pressure is not exceeded following a Loss of Load/Turbine Trip event, this proposed license amendment will not involve a significant reduction in a margin of safety.

Based on the above discussion, it has been determined that the proposed Technical Specification changes do not involve a significant increase in the probability or consequences of an accident previously evaluated; create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety; therefore, the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.5 SERVICE PERIOD FOR HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

**Description of Proposed License Amendments**

The changes in neutron fluence resulting from the proposed Turkey Point Units 3 and 4 uprating have been evaluated and their impact on reactor vessel integrity has been determined. Fluence projections on the vessel were calculated for the uprated power level and calculations were performed for the revised fluences on vessel integrity. It was determined that the current heatup and cooldown curves contained in the Technical Specifications are applicable for the uprated conditions to 19 Effective Full Power Years (EFPY).

## INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

Discussion

(1) **Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

Calculation of the service period for the heatup and cooldown curves does not involve an increase in the probability or consequences of an accident previously evaluated because the calculations were completed to verify the adequacy of the existing curves and to determine an appropriate service period. The use of approved methods and the acceptable results have shown that no design or analysis criteria are changed. The structural and functional integrity of the reactor vessel has been verified.

No fission product barriers or inputs to dose analyses are adversely affected by these calculations and reverification of the existing heatup/cooldown curves. Therefore, the probability or consequences of an accident previously evaluated are not increased.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The revised service period does not create the possibility of a new or different kind of accident from any accident previously evaluated because the recalculation of an acceptable service period does not affect accident initiation sequences. No new operating configuration is being imposed by the calculations that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated have been created as a result of these revisions.

(3) **Operation of the facility in accordance with the proposed license amendments would not involve a significant reduction in a margin of safety.**

Calculations were performed to determine the service period appropriate for the existing curves. The changes to service period do not involve a reduction in a margin of safety because the margin of safety associated with the heatup/cooldown curves, as verified by the results of the analyses, are unchanged. Therefore, the proposed change to the service period does not involve a significant reduction in the margin of safety.

Based on the above discussion, it has been determined that the proposed changes to the Technical Specifications to revise the service period for heatup/cooldown curves are acceptable. The revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated; they neither create the possibility of a new or different kind of accident from any accident previously evaluated, nor involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.6 MODIFICATION TO SURVEILLANCE REQUIREMENT FOR EMERGENCY CONTAINMENT COOLING SYSTEM

### Description of Proposed License Amendments

FPL proposes to modify the actuation logic for the Emergency Containment Cooling (ECC) units for Turkey Point Units 3 and 4. Previously, all three ECC units were automatically started on a safety injection (SI) signal. The revised design and Technical Specification Surveillance Requirements would require only two ECCs to automatically start on an SI signal, and that the third (swing) ECC unit be maintained in an operable condition and available for manual starting.

The ECC units are designed to remove heat from containment and transfer it to the Component Cooling Water System (CCWS) following a postulated loss of coolant accident (LOCA) or main steam line break (MSLB). To support post-LOCA long-term containment cooling and to maintain the containment pressure and temperature during a LOCA or MSLB within their design values, a minimum of two ECC units are required to operate, at least one of the two "automatic" ECC's must start in response to an SI signal with a second ECC being started within 24 hours following the event. The evaluations for the thermal power uprate also determined that, if more than two ECC units automatically start following an SI signal, the amount of heat discharged from the ECC units to the CCWS could cause the CCWS to exceed its design temperature during the injection and/or recirculation phases of the LOCA.

### INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

### Discussion

**(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The purpose of the ECC units is to help mitigate the consequences of an accident (i.e., to help maintain the containment pressure and temperature within their design values following a design basis accident). The ECC units do not operate during normal operation of the plant. Failure of the ECC units would not initiate a plant transient or accident. Therefore, the proposed change involving the ECC units would not affect the probability of occurrence of an accident previously evaluated.

Evaluations demonstrate that, with two ECC units operating during a LOCA or MSLB, the containment pressure and temperature will be maintained within their design values. These evaluations also demonstrate that, with two ECC units operating during a LOCA or MSLB, the temperature of the CCWS will be maintained within its design temperature. Therefore, the proposed change involving the ECC units would not affect the consequences of an accident previously evaluated.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The purpose of the ECC units is to mitigate design basis accidents, and failure of the ECC units would not cause a plant transient or accident. Furthermore, a single failure of an ECC unit during a LOCA or MSLB would not lead to a new or different kind of accident. Although the revised Technical Specifications require two ECC units to start automatically on a LOCA signal, they would also require that all three ECC units be operable. On a single failure of an operating ECC unit, there would be sufficient time to start the standby ECC unit to accomplish the design function of the ECC system. Therefore, the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) **Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The proposed change in the actuation logic of the ECC units would not cause either the containment pressure and temperature or the CCWS temperature to exceed their design values. While the energy released into containment and subsequently transferred to the CCWS will increase as a result of the thermal uprate, this increase is insignificant and will not result in either the containment or CCWS exceeding a design limit. Therefore, the proposed change would not affect the margin of safety.

Based on the above discussion, it has been determined that the proposed changes to the ECCs automatic actuation logic are acceptable. These revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated; they neither create the possibility of a new or different kind of accident from any accident previously evaluated, nor involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

### Description of Proposed License Amendments

The Technical Specifications issued with the operating license for Turkey Point Units 3 and 4 did not include any Limiting Condition for Operation associated with the Control Room Emergency Ventilation System. The Atomic Energy Commission (AEC) requested inclusion of such Technical Specifications in 1974 and provided model Technical Specifications for inclusion in the Turkey Point plant licenses. These model Technical Specifications were based on the removal of greater than or equal to 90% radioactive methyl iodide. The Technical Specifications approved by the NRC in April 1982 included a methyl iodide removal efficiency of 90%.

To assure consistency between testing efficiency and analysis assumptions for control room doses post-accident, the required methyl iodide removal efficiency is being increased to 99%. This increase is consistent with the recommendations of Regulatory Guide 1.52, and supports the analysis for control room doses post-accident. Since this change is clearly conservative, personnel safety will not be adversely impacted.

### INTRODUCTION

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendment.

### Discussion

(1) **Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed change does not affect the integrity of the fission product barriers utilized for mitigation of dose consequences as a result of an accident. Only the iodide removal efficiency of the control room emergency ventilation system is increased, and this change is in the conservative direction.

To assure consistency between testing efficiency and analysis assumptions for post-

accident control room doses, the methyl iodide removal efficiency required to be demonstrated by laboratory test, is being increased from 90% to 99%. This increase in testing efficiency is consistent with the recommendations set by the NRC staff in Regulatory Guide 1.52 to support analysis efficiencies for elemental iodine and methyl iodide removal of 95%, respectively. Testing performed to verify methyl iodide removal efficiency will be performed under conditions representative of the control room environment.

Since this change in removal efficiency is in the conservative direction, plant safety will not be adversely impacted.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed change to the control room emergency ventilation system iodide removal efficiency does not create the possibility of a new or different kind of accident from any accident previously evaluated because operation of the control room emergency ventilation system is not identified in any accident initiation sequence. The system is provided to minimize operator exposure to airborne radioactivity released as a result of an accident. The new operating configuration has been evaluated and no new limiting single failures have been identified as a result of the proposed modification. Therefore, it is concluded that no new or different kind of accidents from any accident previously evaluated have been created as a result of these revisions.

(3) **Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The proposed changes do not involve a reduction in the margin of safety because the margin of safety associated with this change is in the conservative direction. Thus, plant safety will not be adversely impacted and the margin of safety required for the affected safety analysis is maintained. The adequacy of the revised Technical Specification values to maintain the plant in a safe operating condition has been confirmed, since the testing will be done to a more conservative criteria (i.e., 99% efficiency). Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above discussion, it has been determined that the proposed Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated; create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety; therefore, the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

## 2.8 RELOCATION OF $F_Q(Z)$ AND $F_{\Delta H}$ LIMITS FROM TECHNICAL SPECIFICATIONS TO CORE OPERATING LIMITS REPORT AND EDITORIAL CORRECTIONS

**Description of Proposed License Amendments**

Generic Letter (GL) 88-16, dated October 4, 1988, encouraged licensees to amend the Technical Specifications related to cycle specific parameters. The GL provided guidance for relocation of certain cycle-dependent core operating limits from a licensee's Technical Specifications to the COLR. This would allow changes to the values of the core operating limits without prior NRC approval (i.e., license amendment), as long as an NRC approved methodology for the parameter limit calculation is followed. The proposed Technical Specification changes will relocate cycle specific parameter limits from the Technical Specifications to the Core Operating Limits Report (COLR). In accordance with the recommendations of GL 88-16 , FPL proposes the addition of the following parameters to the COLR:

(a) Heat Flux Hot Channel Factor, $F_Q(Z)$, and

(b) Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$ (which includes the Power Factor Multiplier).

These parameters are added to the Axial Flux Difference, Rod Bank Insertion Limits and the K(Z) curve currently included in the COLR in accordance with TS 6.9.1.7.

In addition, editorial corrections are proposed to ensure consistency within the Technical Specifications.

**INTRODUCTION**

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendments.

**Discussion**

(1) **Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The relocation of the values for $F_Q$ and $F_{\Delta H}$ from the Technical Specifications to the Core Operating Limits Report is administrative in nature and has no impact on the probability or consequences of any Design Bases Event (DBE) occurrence which was previously evaluated. The determination of the $F_Q$ and $F_{\Delta H}$ limits will be performed using methodology approved by the NRC and poses no significant increase in the probability or consequences of any accident previously evaluated.

The changes being proposed as editorial in nature do not affect assumptions contained in the safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, these proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) **Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The relocation of the $F_Q$ and $F_{\Delta H}$ limits from the Technical Specifications to the Core Operating Limits Report is administrative in nature and has no impact, nor does it contribute in any way to the possibility of a new or different kind of accident from any accident previously evaluated.

The determination of the $F_Q$ and $F_{\Delta H}$ limits will be performed using NRC-approved methodology and are submitted to the NRC as a revision to the COLR to allow the NRC staff to trend peaking factors. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken if the $F_Q$ and $F_{\Delta H}$ limits are exceeded. Therefore, the proposed amendments does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

The editorial changes proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the facility, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, these changes do not create the possibility of a new or different kind of accident.

**(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.**

The relocation of the $F_Q$ and $F_{\Delta H}$ limits from the Technical Specifications to the Core Operating Limits Report is administrative in nature and has no impact on the margin of safety. The determination of the $F_Q$ and $F_{\Delta H}$ limits will be performed using methodology approved by the NRC and does not constitute a significant reduction in the margin of safety.

The supporting Technical Specification values are defined by the accident analyses which are performed to conservatively bound the operating conditions defined by the Technical Specifications. Performance of analysis and evaluation have confirmed that the operating envelope defined by the Technical Specifications continues to be bounded by the analytical basis, which in no case exceeds the acceptance limits. Therefore, the margin of safety provided in the analyses in accordance with the acceptance limits is maintained and not significantly reduced.

The changes being proposed as editorial in nature do not relate to or modify the safety margins defined in, and maintained by the Technical Specifications. Therefore, the proposed changes which correct administrative errors and clarify existing Technical Specification requirements do not involve any reduction in a margin of safety.

Based on the above discussion, it has been determined that the proposed changes to the Technical Specifications are acceptable. These revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated; they neither create the possibility of a new or different kind of accident from any accident previously evaluated, nor involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.