FPL

AUG 1 5 2003

L-2003-140
10 CFR 50.90
10 CFR 50.91

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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to Request for Additional Information for
Proposed License Amendments – Administrative Update of Technical Specifications

By letter L-2002-210, dated December 20, 2002, Florida Power and Light Company (FPL) submitted a request to amend Appendices A and B of Facility Operating Licenses DPR-31 and DPR-41 to achieve consistency and improved content through editorial and administrative changes. In response to the NRC request for additional information dated June 5, 2003, Attachment 1 provides the additional information requested. As discussed with NRC Staff, Attachment 2 provides revisions to supercede sections of FPL's L-2002-210 submittal.

FPL has determined that the additional information provided herein does not change the conclusions reached in the original no significant hazards consideration determination provided in FPL letter L-2002-210.

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

Should there be any questions, please contact us.

Very truly yours,

Terry Jones

Terry O. Jones
Vice President
Turkey Point Plant

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant
Florida Department of Health

A001

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to Request for Additional Information for
Proposed License Amendments - Administrative Update of Technical Specifications

STATE OF FLORIDA)
) ss.
COUNTY OF MIAMI-DADE)

Terry O. Jones, being first duly sworn, deposes and says:

That he is Vice President, Turkey Point 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.

Terry O. Jones
Terry O. Jones

Subscribed and sworn to before me this

15 day of August, 2003.

Olga Hanek
Olga Hanek
Name of Notary Public (Type or Print)

OLGA HANEK
MY COMMISSION # CC 926970
EXPIRES: June 18, 2004
Bonded Thru Notary Public Underwriters

Terry O. Jones, is personally known to me.

L-2003-140

ATTACHMENT 1

PROPOSED RESPONSE TO THE NRC REQUEST
FOR ADDITIONAL INFORMATION REGARDING
AMENDMENT REQUESTS FOR AN ADMINISTRATIVE
UPDATE OF TECHNICAL SPECIFICATIONS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1. Change #7: Concerning the deletion of "below an average coolant temperature of 275°F," it appears that the Bases and Technical Specifications (TSs) disagree in wording.

- **TSs use average coolant temperature of 275°F.**
- **Bases use less than or equal to coolant temperature of 275°F.**

Provide an explanation for the discrepancy. (Title 10 of the Code of Federal Regulation (10 CFR) Section 50, Appendix G, and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Appendix G, contain guidance for the origin and use of a coolant temperature of 275°F)

Response:

Administrative Change #7, as proposed in FPL letter L-2002-210, "Proposed License Amendments Administrative Update of Technical Specifications" was submitted to delete the phrase "below an average coolant temperature of 275°F" from the wording of LCO 3.4.9.3, since it was redundant to the applicable plant condition already specified in the Applicability section of the TS. This change was considered to be administrative in nature.

NRC and FPL review of the proposed change to TS 3.4.9.3 noted an editorial error in a previously submitted license amendment which had caused the TS wording to deviate from the wording of the TS BASES. FPL letter L-89-201, "Proposed License Amendment Revised Technical Specifications" submitted on June 5, 1989 and approved in the August 28, 1990 NRC letter, "Turkey Point Units 3 and 4 – Issuance of Amendments Re: Replacement of Current Technical Specifications with Revised Technical Specifications" (Amendment numbers 137 and 132), changed the wording of the previously existing old-style formatted Overpressure Mitigating System (OMS) TS 3.15 (Amendment numbers 79 and 73) from requiring OMS operability at "RCS temperature less than or equal to 275°F" to the new Revised TS formatted wording of "below an RCS average coolant temperature of 275°F" in the LCO. "With RCS average coolant temperature less than or equal to 275°F" was also used in Action Statement b. of TS 3.4.9.3. The changes as submitted, were considered administrative but were not specifically justified as such by FPL. However, these changes were categorized as administrative in Table I of the Safety Evaluation for Turkey Point Amendments 137 and 132, dated August 28, 1990.

These changes were subsequently carried through in FPL's GL 90-06 response, L-92-285, "Proposed License Amendments Response to Generic Letter 90-06 Resolution of Generic Issues 70 and 94" submitted on November 25, 1992, and approved in NRC letter "Turkey Point Units 3 and 4 – Issuance of Amendements Re: Power-Operated Relief Valve Reliability and Low Temperature Overpressure Protection – Generic Letter 90-06" on June 30, 1994, in which FPL chose the option under the guidance of GL 90-06 to keep the previously existing OMS TS 3.4.9.3 LCO statement as it existed. The condition of "(below an average coolant temperature of 275°F)" was also included in the Applicability section and the Action Statement 3.4.9.3 b. in the amendment.

FPL agrees with the NRC observation and suggests that the proper characterization for OMS applicability is resident in the TS BASES document for MODE 4 when the temperature of any RCS cold leg is less than or equal to 275°F. FPL will amend Administrative Change #7, as submitted in L-2002-210, to change the APPLICABILITY wording of TS 3.4.9.3 to read: [MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.] ACTION b. of TS 3.4.9.3 will read: [With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.]

These changes conform to the TS APPLICABILITY wording recommendations of GL 90-06 for Westinghouse plants. These changes are supported by analysis previously provided for TS Amendments 208/202, FPL letter L-2000-146, "Resubmittal of Proposed License Amendments "Revised Pressure-Temperature (P/T) Curves, and Cold Overpressure Mitigation System (COMS) Setpoints"". This amendment justifies the present OMS TSs, based on ASME Code Case N-641, "Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection Requirements, Section XI, Division 1." These changes also provide consistency in wording, conforming to the TS BASES for OMS and related RCS TSs 3.4.1.3, 3.4.1.4.1, and 4.4.9.3.1.

The proposed changes are made for consistency with already established technical bases and with other sections of the TSs. These changes are therefore considered administrative in nature. The proposed changes do not affect the No Significant Hazards Consideration Determination of L-2002-210.

Changes to L-2002-210, reflecting the revised Administrative Change #7 are provided in Attachment 2 of this letter. Enclosures 1, 4 and 5 of the L-2002-210 will be superceded by new Enclosures 1, 4 and 5, provided in Attachment 2 of this letter.

2. **Section 50.90 of 10 CFR requires that amendments to the license be fully described. Section 3.2 of the submittal discusses the types of editorial changes made and indicates that no specific justifications for the changes are provided. However, no marked-up TS pages were provided to show what editorial changes were made.**

   Provide the mark-ups of the editorial changes not submitted with the original application.

Response:

At the time of the proposed license amendment submittal of FPL letter L-2002-210, dated December 20, 2002, Florida Power and Light Company (FPL) understood that a complete clean copy of the entire Technical Specifications, containing all editorial changes, would be sufficient for NRC review and approval. As provided in Section 3.2 of the submittal, the changes were purely editorial, and included deletion of blank pages, grammatical and spelling corrections and roll-up of text within sections. This was meant for cleanup of text and search capability purposes only.

The markup of Technical Specification text is available; but the process to produce it involved numerous iterations and reviews by many departments; as such, it is not easily retrievable at this time.

For purposes of completing this submittal process for the proposed administrative changes, FPL requests withdrawal of the editorial change request reflected in Enclosure 5 of FPL letter L-2002-210. All references to editorial changes in the Table of Contents and Enclosures 1, 2 and 5 of the L-2002-210 will be superceded by new Enclosures 1, 2 and 5, provided in Attachment 2 of this letter.

These changes are considered administrative in nature and do not affect the No Significant Hazards Consideration Determination of L-2002-210.

ATTACHMENT 2
CHANGES TO FPL LETTER L-2002-210, PROPOSED LICENSE AMENDMENTS UPDATE OF TECHNICAL SPECIFICATIONS

Summary of Revisions to Enclosures for Submittal under FPL Letter L-2002-210

Table of Contents (supercedes page 4 of 4, of FPL letter L-2002-210)

Enclosure 1 (supercedes Enclosure 1 of FPL letter L-2002-210)

Enclosure 2 (supercedes Enclosure 2 of FPL letter L-2002-210)

Enclosure 3 (no change from L-2002-210)

Enclosure 4 (supercedes Enclosure 4 of FPL letter L-2002-210)

Enclosure 5 (supercedes Enclosure 5 of FPL letter L-2002-210)

SUMMARY OF REVISIONS TO ENCLOSURES FOR SUBMITTAL UNDER FPL LETTER L-2002-210

New versions of the Table of Contents and Enclosures 1, 2, and 5, deleting editorial change proposals and references to those changes are provided in Attachment 2 of this letter. In addition, changes to Enclosures 1, 4 and 5 are provided incorporating the changes to Administrative Change #7 as described in FPL's response to NRC RAI #1. The Environmental Consideration of Enclosure 3 will remain unchanged.

The enclosures herein are provided to supercede those enclosures previously submitted in FPL letter L-2002-210. The revised documents are as follows:

The Table of Contents, L-2002-210, page 4 of 4

- The Table of Contents, page 4 of 4, of L-2002-210 has been changed to rename Enclosure 5 as "Clean Copy of Technical Specifications Pages Incorporating Administrative Changes."

Enclosure 1, L-2002-210

- Enclosure 1 has been changed to delete all references to and descriptions of editorial changes.
- Enclosure 1 has also been changed to incorporate the revised Administrative Change #7 as described in the response to NRC RAI #1.

Enclosure 2, L-2002-210

- Enclosure 2's Introduction has been changed to delete all references to editorial changes as described in FPL's response to NRC RAI #2.

Enclosure 3, L-2002-210

- Enclosure 3 remains unchanged.

Enclosure 4, L-2002-210

- Enclosure 4 has been changed to incorporate the revised mark up TS page changes of Administrative Change #7 as described in FPL's response to NRC RAI #1.

Enclosure 5, L-2002-210

- Enclosure 5 has been renamed "Clean Copy of Technical Specifications Pages Incorporating Administrative Changes" and has been changed to provide clean copy TS pages of only the administrative changes proposed in Enclosure 1 as described in FPL's response to NRC RAI #2.
- Enclosure 5 also provides the clean copy TS pages for Administrative Change #7 as described in the response to NRC RAI #1.

TABLE OF CONTENTS

Enclosures

ENCLOSURE 1

PROPOSED LICENSE AMENDMENTS APPLICATION

1.0 Description of Proposed Changes

Florida Power and Light Company (FPL) requests that Appendices A and B of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4 be amended to achieve improved Technical Specifications (TS) consistency and content through administrative changes.

The proposed amendments consist of changes to correct errors in the TS, including equation reformatting, wording changes, and other clarifications. These administrative changes are made for consistency with previously approved TS amendments, and other parts of the existing TS. There are no substantive changes made in these Proposed License Amendments.

A description and justification of the amendments request is provided in Enclosure 1. The no significant hazards consideration determination and environmental impact analysis in support of the proposed Technical Specification changes are provided in Enclosures 2 and 3, respectively. Enclosure 4 provides the proposed marked up Technical Specification pages of corrections to the identified administrative errors. Enclosure 5 provides a clean copy version of the TS changes incorporating the administrative corrections.

2.0 Background

Through periodic document reviews, several administrative inconsistencies were noted throughout the TS. These inconsistencies were cross-checked and verified to be administrative errors, based on review of previous amendments and the present TS. These changes are presented as part of these amendments and are justified as administrative changes only.

3.0 Description and Justification of Proposed Changes

3.1 Administrative Changes

Periodic reviews of the Turkey Point TS have identified several administrative errors. Marked-up copies of the administrative corrections are provided in Enclosure 4. Corrections made are justified as follows:

1. **Change:** On TS Index, page xiv, delete [5.5 METEOROLOGICAL TOWER LOCATION....5-4] and renumber the affected sections. On TS page 5-4, delete associated section and wording [5.5 METEOROLOGICAL TOWER LOCATION 5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.] and renumber the affected sections. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #1. The clean copy of this change is provided on page xiv of Enclosure 5.

   Justification: License Amendments 219 and 213, approved February 12, 2002, NRC SER "Turkey Point Units 3 and 4- Issuance of Amendments Regarding Removal of Site Area and Plant Area Maps from Technical Specifications (TAC Nos. MB1968 and MB 1969)" approved the relocation of TS Figure 5.1-1, "Site Area Map" and Figure 5.1.3 "Map Defining Unrestricted Areas and Site Boundary for Radioactive Gaseous and Liquid Effluents" from the TS to the Turkey Point Updated Final Safety Analysis Report (UFSAR). Section 5.5 of the TS, which also refers to the relocated Figure 5.5-1 was overlooked and should have been removed as part of the license amendment.

This change is administrative in nature. It removes outdated references providing greater clarity and consistency to the TS.

2. **Change:** On TS Index pages i and ii, delete definitions 1.15, 1.16, and 1.21, reorder definitions 1.37 and 1.38 alphabetically, and renumber affected entries. On TS pages 1-3 and 1-4, remove deleted definitions 1.15, 1.16, and 1.21. On TS pages 1-6 and 1-6a, relocate [DIGITAL CHANNEL OPERATIONAL TEST] and [CORE OPERATING LIMITS REPORT] definitions 1.37 and 1.38 to insert them alphabetically in the DEFINITIONS text. Renumber all affected definitions. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #2. The clean copy of this change is provided on pages i, ii, 1-4, and 1-6 of Enclosure 5.

   Justification: Definitions 1.15, 1.16, and 1.21 were deleted in previous license amendments and are thus irrelevant to the TS. Definitions 1.37 and 1.38 were added out of alphabetical order in previous amendments and are being placed in alphabetical order to conform to the format of the Definitions section of the TS. This change is administrative in nature, deleting irrelevant wording and providing consistency to the TS Index and Definitions.

3. **Change:** On TS page 2-7, the equation :

$$\Delta T \quad \left\{\frac{1+\tau_1 S}{1+\tau_2 S}\right\} \frac{(\quad 1 \quad)}{1+\tau_3 S} \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \; [\,T \frac{(\quad 1 \quad)}{1+\tau_6 S} \quad - T'\,] + K_3(P - P') - f_1(\Delta I)\right\}$$

   will be replaced with:

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{1}{(1+\tau_6 S)} - T'\right] + K_3\,(P - P') - f_1(\Delta I)\right\}$$

   The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #3. The clean copy of this change is provided on page 2-7 of Enclosure 5.

   Justification: Early versions of the Turkey Point TS were typed using type fonts available at the time. Word processing now allows TS to be written using more appropriate mathematical symbols. This change also conforms to the equation formatting of the Overtemperature ΔT formula on page 3.3.1-17 of NUREG-1431, Vol. 1, Rev. 2, "Standard Technical Specifications Westinghouse Plants." This change is administrative in nature, improving the consistency and readability and does not alter the content or the intended interpretation of the formula.

4. **Change:** On TS page 2-9, the equation :

$$\Delta T \quad \left\{\frac{1+\tau_1 S}{1+\tau_2 S}\right\} \frac{(\ 1\ )}{(1+\tau_3 S)} \le \Delta T_0 \left\{K_4 - K_5 \frac{(\tau_7 S)}{1+\tau_7 S} \quad \frac{(\ 1\ )}{(1+\tau_6 S)} \ T - K_6 \quad \left[T \frac{(\ 1\ )}{(1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

will be replaced with:

$$\Delta T \quad \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_0 \left\{K_4 - K_5 \ \frac{\tau_7 S}{1+\tau_7 S} \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \frac{1}{1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #4. The clean copy of this change is provided on page 2-9 of Enclosure 5.

Justification: Early versions of the Turkey Point TS were typed using type fonts available at the time. Word processing now allows TS to be written using more appropriate mathematical symbols. This change also conforms to the equation formatting of the Overpower ΔT formula on page 3.3.1-18 of NUREG-1431, Vol. 1, Rev. 2, "Standard Technical Specifications Westinghouse Plants." This change is administrative in nature, improving the consistency and readability and does not alter the content or the intended interpretation of the formula.

5. **Change:** On TS Index page iv, remove [Heat Tracing.....3/4 1-16], remove entire page 3/4 1-16. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #5. The clean copy of this change is provided on page 3/4 1-16 of Enclosure 5.

Justification: License Amendments 144 and 139, approved on July 16, 1991, NRC SER "Turkey Point Units 3 and 4- Issuance of Amendments Re: Boric Acid Concentration Reduction (TAC Nos. 79192 and 79193)" approved the reduction of boric acid concentration and the removal of heat tracing requirements for the boric acid tanks and boric acid makeup system piping and valves. The footnote of TS 3.1.2.6 states that the TS is no longer applicable once boric acid tanks inventory and boric acid source and flow paths inventories have been diluted to less than or equal to 3.5 weight percent (wt%). Plant Change/Modifications completed on August 3, 1991 for both units 3 and 4 reduced the concentration of boric acid in the boric acid system to less than 3.5 wt% thus eliminating the boric acid heat tracing system requirements of TS 3.1.2.6.

This change is administrative in nature, removing outdated requirements and completes the implementation of previously submitted and approved License Amendments 144 and 139.

6. **Change:** On TS page 3/4 4-32, change the title of Figure 3.4-3 from [FPL 32 EFPY HEATUP CURVES] to [FPL 32 EFPY COOLDOWN CURVES]. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #6. The clean copy of this change is provided on page 3/4 4-32 of Enclosure 5.

   Justification: The description provided below the Figure 3.4-3 on page 3/4 4-32 describes the figure as "Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations," as such, the title above the figure is erroneously labeled as a heatup curve when it is actually intended to be used for cooldown limitations requirements.

   This change is administrative in nature, correcting an error in the titling of a figure.

7. **Change:** On TS page 3/4 4-36, delete the words [below an average coolant temperature of 275°F] from Limiting Condition For Operation TS 3.4.9.3. Change the wording in TS 3.4.9.3 APPLICABILITY to [MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.] Change the wording of TS 3.4.9.3 ACTION b. to [With one PORV inoperable in MODE 4 (when the temperature of any cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.] The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #7. The clean copy of this change is provided on page 3/4 4-34 of Enclosure 5.

   Justification: The Limiting Condition For Operability wording of TS 3.4.9.3 "below an average coolant temperature of 275°F" is redundant to the applicable plant condition specified in the Applicability section of the TS. This change is being made to eliminate the redundant applicability statement to clarify the APPLICABILITY and ACTION sections of TS 3.4.9.3.

   NRC and FPL review of previous changes to TS 3.4.9.3 noted an editorial error in the Revised Technical Specifications (RTS) project license amendment submittal L-89-201, "Proposed License Amendment Revised Technical Specifications" on June 5, 1989. This submittal was approved in the August 28, 1990 NRC letter, "Turkey Point Units 3 and 4 – Issuance of Amendments Re: Replacement of Current Technical Specifications with Revised Technical Specifications." This amendment changed the wording of the previously existing old-style formatted Overpressure Mitigating System (OMS) TS 3.15 (amendment numbers 79 and 73) from requiring OMS operability at "RCS temperature less than or equal to 275°F" to the new RTS formatted wording (amendment numbers 137 and 132) of "below an RCS average coolant temperature of 275°F" in the LCO. "RCS average coolant temperature less than or equal to 275°F" was also used in Action Statement b. of TS 3.4.9.3. The changes as submitted, were categorized as administrative in nature but were not specifically justified in the submittal.

These changes were subsequently carried through, with no further changes made, in FPL's GL 90-06 response, L-92-285, "Proposed License Amendments Response to Generic Letter 90-06 Resolution of Generic Issues 70 and 94" submitted on November 25, 1992, and approved in NRC letter "Turkey Point Units 3 and 4 – Issuance of Amendments Re: Power-Operated Relief Valve Reliability and Low Temperature Overpressure Protection – Generic Letter 90-06" on June 30, 1994, in which FPL chose the option under the guidance of GL 90-06 to keep the previously existing OMS TS 3.4.9.3 LCO statement as it existed. The condition of "(below an average coolant temperature of 275°F)" was also included in the Applicability section and Action Statement 3.4.8.3 b. in the amendment.

FPL considers the changes proposed to be administrative in nature, providing consistency and clarification to the existing TS. These changes conform to the TS Applicability wording recommendations of GL 90-06 for Westinghouse plants. The proposed changes are supported by analysis previously provided for TS Amendments 208/202, FPL letter L-2000-146, "Resubmittal of Proposed License Amendments "Revised Pressure-Temperature (P/T) Curves, and Cold Overpressure Mitigation System (COMS) Setpoints"". This amendment justifies the present OMS TSs based on ASME Code Case N-641, "Alternative Pressure Temperature Relationship and Low temperature Overpressure Protection Requirements, Section XI, Division 1." These changes will also provide consistent wording conforming to related RCS TSs 3.4.1.3, 3.4.1.4.1, and 4.4.9.3.1.

8. **Change:** On TS page 3/4 6-1, in Surveillance Requirement 4.6.1.1 a., replace the phrase [secured in their positions;] with [secured in their closed positions;]. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #8. The clean copy of this change is provided on page 3/4 6-1 of Enclosure 5.

   Justification: Interpretation of the TS as presently written could create confusion as to what the intended position of valves, blind flanges, or deactivated automatic valves would be by stating they should be "secured in their positions." Specifying that the components should be "secured in their <u>closed</u> positions" removes any confusion as to what the intended final position of the valves should be. This improvement is administrative in nature and conforms to the content of ACTION A.1 of TS 3.6.3, Containment Isolation Valves, of NUREG-1431, Vol. 1, Rev. 2, Standard Technical Specifications Westinghouse Plants. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #8. The clean copy of this change is provided on page 3/4 6-1 of Enclosure 5.

9. **Change:** On TS page 3/4 7-11, TS 3.7.1.6 ACTION Statement b., replace the phrase [With both Standby Steam Generator Feedwater Pumps, restore at least one pump to OPERABLE status...] to [With both Standby Steam Generator Feedwater Pumps inoperable, restore at least one pump to OPERABLE status...]. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #9. The clean copy of this change is provided on page 3/4 7-10 of Enclosure 5.

   Justification: The addition of the word "inoperable" to the TS corrects the grammatical logic and reinforces the intent of the TS as it is written. This improvement is administrative logic and reinforces the intent of the TS as it is written. This improvement is administrative in nature, providing greater consistency and readability to the TS.

10. **Change:** On TS page 3/4 9-7, delete [*] from TS 3.9.7 and associated footnote at the bottom of page 3/4 9-7. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #10. The clean copy of this change is provided on page 3/4 9-7 of Enclosure 5.

Justification: License Amendments 108 and 102, issued August 29, 1984, approved the prohibition of travel of heavy loads over irradiated fuel assemblies with the exception of a temporary crane for use during proposed reracking of the spent fuel pools. Plant Change/Modifications completed installation of new high-density fuel racks in the Unit 3 and 4 spent fuel pits on September 7, 1989. Completion of the reracks for both units makes this exception to the TS no longer relevant.

This change is administrative in nature, deleting irrelevant information from the TS.

11. **Change:** On the note at the bottom of TS page 3/4 9-12, replace [safety evaluation] with [10 CFR 50.59 evaluation]. The mark-up version of these changes is shown in Enclosure 4 as Administrative Change #11. The clean copy of this change is provided on page 3/4 9-12 of Enclosure 5.

Justification: The phrase "safety evaluation" is not sufficiently descriptive to ensure that the intent of the TS note is met. Replacing the words "safety evaluation" with "10 CFR 50.59 evaluation" will ensure that a complete and adequate safety assessment is performed in accordance with the requirements of 10 CFR 50.59. This change is administrative in nature, improving the clarity of TS requirements.

3.2 **(Deleted)**

ENCLOSURE 2
NO SIGNIFICANT HAZARDS CONSIDERATIONS

Introduction

Florida Power and Light Company (FPL) requests that Appendixes A and B of Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4 be amended to achieve consistency and improved content through administrative changes. Administrative changes include equation reformatting, wording changes, and other clarifications.

No Significant Hazards Considerations

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a 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. A discussion of these standards as they relate to this change request follows.

1. ***Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?***

   No. The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed amendments are purely administrative or editorial in nature. These amendments make no substantive Technical Specification changes and do not affect any assumptions contained in plant safety analyses, the physical design and/or operation of the plant; and they do not affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

2. ***Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?***

   No. The use of the administratively changed Technical Specifications do not create the possibility of a new or different kind of accident from any previously evaluated, since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes and clarifications, since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

3. ***Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?***

   No. The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The changed Technical Specifications, which correct administrative and editorial errors, and clarify existing Technical Specification requirements, do not reduce any of the margins of safety.

Based on the reasoning presented above, FPL has determined that the requested changes involve no significant hazards consideration.

ENCLOSURE 4

PROPOSED MARK-UP OF ADMINISTRATIVE CHANGES TO AFFECTED TECHNICAL SPECIFICATION PAGES

Administrative Change	Affected TS Pages
Change #1	xiv, 5-4, 5-5, 5-6, 5-7
Change #2	i, ii, 1-2, 1-3, 1-4, 1-5, 1-6, 1-6a
Change #3	2-7
Change #4	2-9
Change #5	iv, 3/4 1-16
Change #6	3/4 4-32
Change #7	3/4 4-36
Change #8	3/4 6-1
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Change #11	3/4 9-12

Enclosure 4
Administrative Change #1

INDEX

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4 or ZIRLO™, except that replacement of fuel rods by filler rods consisting of stainless steel, or by vacant rod positions, may be made in fuel assemblies if justified by cycle-specific reload analysis using NRC-approved methodology. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches. Should more than 30 individual rods in the core, or 10 fuel rods in any fuel assembly, be replaced per refueling, a Special Report discussing the rod replacements shall be submitted to the Commission within 30 days after cycle startup.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig ± 1%, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The nominal water and steam volume of the Reactor Coolant System is 9343 cubic feet at a nominal T_{avg} of 574.2°F.

~~5.5 METEOROLOGICAL TOWER LOCATION~~

~~5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.~~

5.~~6~~5 FUEL STORAGE

5.~~6~~5.1 CRITICALITY

5.~~6~~5.1.1 The spent fuel storage racks are designed to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison and shall be maintained with:

a. k_{eff} equivalent to less than 1.0 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in WCAP-14416-P.

b. A k_{eff} equivalent to less than or equal to 0.95 when flooded with borated to 650 ppm water, which includes a conservative allowance for uncertainties as described in WCAP-14416-P.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for two region fuel storage racks.

d. The maximum enrichment loading for fuel assemblies is 4.5 weight percent of U-235.

5.~~6~~5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

5.~~6~~5.1.3 Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks. Administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

DRAINAGE

5.~~6~~5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.~~6~~5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1404 in two region storage racks

5.~~7~~6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.~~7~~6.1 The components identified in Table 5.~~7~~6-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.~~7~~6-1.

TABLE 5.~~7~~6-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at < 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from < 200°F to > 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from > 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	> 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	150 leak tests.	Pressurized to ≥ 2435 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3100 psig.
Secondary Coolant System	6 loss of secondary pressure	Loss of Secondary pressure
	50 leak tests	Pressurized to ≥ 1085 psig
	35 hydrostatic pressure tests.	Pressurized to ≥ 1356 psig.

Enclosure 4
Administrative Change #2

INDEX

DEFINITIONS

Insert (A) then (B)

INDEX

DEFINITIONS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

Insert (A) then (B)

DOSE EQUIVALENT I-131

1.~~10~~ 12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.~~11~~ 13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

FREQUENCY NOTATION

1.~~12~~14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GAS DECAY TANK SYSTEM

1.~~13~~15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.~~14~~16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

~~1.15 (Deleted)~~

~~1.16 (Deleted)~~

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 13.5 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.20 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

~~1.21 (Deleted)~~

PURGE - PURGING

1.~~22~~21 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

DEFINITIONS

QUADRANT POWER TILT RATIO

1.~~23~~ 22 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.~~24~~ 23 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2300 MWt.

REPORTABLE EVENT

1.~~25~~ 24 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.~~26~~ 25 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full-length rod cluster~~ assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.~~27~~ 26 The SITE BOUNDARY shall mean that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

1.~~28~~ 27 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.~~29~~ 28 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.~~30~~ 29 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.~~31~~ 30 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.~~32~~ 31 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.~~33~~ 32 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.~~34~~ 33 An UNRESTRICTED AREA shall mean an area, access to which is neither limited nor controlled by the licensee.

VENTILATION EXHAUST TREATMENT SYSTEM

1.~~35~~ 34 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.~~36~~ 35 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

B

remove & insert on pg. 1-2

DIGITAL CHANNEL OPERATIONAL TEST

1.~~37~~ 11 A DIGITAL CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock, and/or trip functions.

(A) removed & insert on pg 1-2

CORE OPERATING LIMITS REPORT

1.~~38~~ 10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with NRC approved methodology. Unit operation within these operating limits is addressed in individual specifications. The COLR is submitted to the NRC in accordance with the requirements of 6.9.1.7.

Enclosure 4
Administrative Change #3

1

$$\frac{(1+\tau_1 S)}{(1+\tau_2 S)} \qquad \left(\frac{1}{1+\tau_3 S}\right) \qquad \frac{1}{(1+\tau_6 S)}$$

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)\left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)}\left[T\left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = Lead/Lag compensator on measured ΔT; τ_1=0s, τ_2 =0s

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured ΔT; τ_3 = 0s

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

K_1 = 1.24;

K_2 = $0.017/^{o}F$;

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 25s, τ_5 = 3s;

T = Average temperature, ^{o}F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg}; τ_6 = 0s

T' ≤ $577.2^{o}F$ (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.001/psig;

P = Pressurizer pressure, psig;

Enclosure 4
Administrative Change #4

$$\frac{(1+\tau_1 S)}{(1+\tau_2 S)} \qquad \left(\frac{1}{1+\tau_3 S}\right) \qquad \left(\frac{1}{1+\tau_6 S}\right) \qquad \frac{1}{1+\tau_6 S}$$

NOTE 3: OVERPOWER ΔT

$$\Delta T \left\{\frac{1 + \tau_1 S}{1 + \tau_2 S}\right\}\left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left\{\frac{\tau_7 S}{1 + \tau_7 S}\right\}\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T\left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 $\leq$ 1.10,

K_5 $\geq$ 0.02/^oF for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

Enclosure 4
Administrative Change #5

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

(*)3.1.2.6 At least two independent channels of heat tracing shall be OPERABLE for the boric acid storage tank and for the heat traced portions of the associated flow paths required by Specification 3.1.2.2.

APPLICABILITY: MODES 1, 2, 3 and 4
MODES 5 and 6 (~~when the boric~~ acid storage tank is the borated water source per Specification 3.1.2.4)

ACTION:

MODES 1, 2, 3 and 4

With only one channel of heat tracing on either the boric acid storage tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6

With only one channel of heat tracing on either the boric acid storage tank or on the heat traced portion of an associated flow path OPERABLE, operations involving CORE ALTERATIONS or positive reactivity additions may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, suspend all activities involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each heat tracing channel for the boric acid storage tank and associated flow path required by Specification 3.1.2.2 shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 7 days by verifying the tank and flow path temperatures to be greater than or equal to 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

*This is no longer applicable once boric acid tanks inventory and boric acid source and flow paths inventories have been diluted to less than or equal to 3.5 weight percent (wt%).

Enclosure 4
Administrative Change #6

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. #71249)

LIMITING ART VALUES AT 32 EFPY: 1/4T, 262°F
3/4T, 218°F

COOLDOWN

FPL 32 EFPY ~~HEATUP~~ CURVES

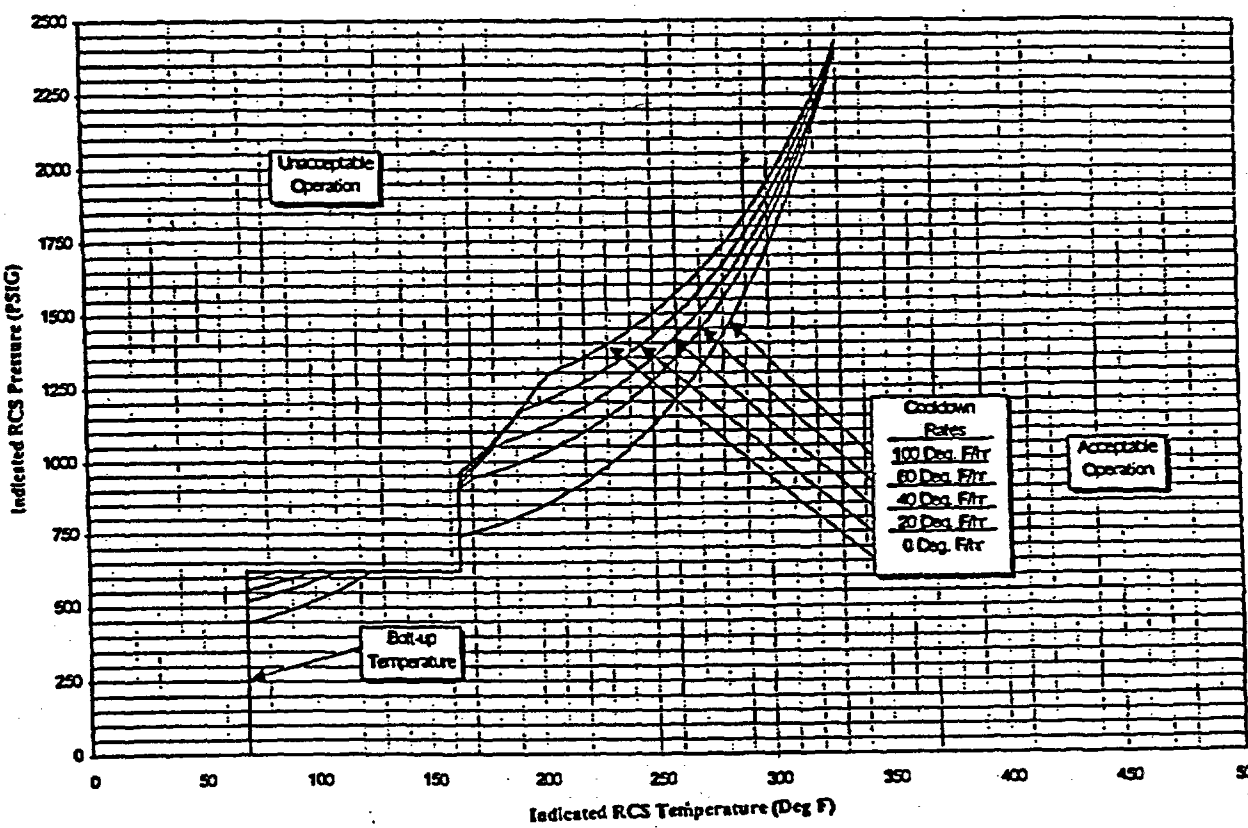

FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rate of 0, 20, 40, 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

Enclosure 4
Administrative Change #7

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and ~~below an RCS average coolant temperature of 275°F~~ at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 468 psig, or

b. The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY: MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F) ~~(below an RCS average coolant temperature of 275°F)~~, 5, and 6 with the reactor vessel head on.

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F) ~~(below an RCS average coolant temperature of 275°F)~~, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in Modes 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

e. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

f. The provisions of Specification 3.0.4 are not applicable.

Enclosure 4
Administrative Change #8

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

*Exception may be taken under Administrative Controls for opening of valves and airlocks necessary to perform surveillance, testing requirements and/or corrective maintenance. In addition, Specification 3.6.4 shall be complied with.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

Enclosure 4
Administrative Change #9

STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least 135,000 gallons of water (indicated volume), shall be in the Demineralized Water Storage Tank.**

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one Standby Steam Generator Feedwater Pump inoperable, restore the inoperable pump to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both Standby Steam Generator Feedwater Pumps inoperable, restore at least one pump to OPERABLE status within 24 hours, or:

   1. Notify the NRC within the following 4 hours, and provide cause for the inoperability and plans to restore pump(s) to OPERABLE status and,

   2. Submit a SPECIAL REPORT per 3.7.1.6d.

c. With less than 135,000 gallons of water indicated in the Demineralized Water Storage Tank restore the available volume to at least 135,000 gallons indicated within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits at least once per 24 hours.

4.7.1.6.2 At least monthly verify the standby feedwater pumps are OPERABLE by testing in recirculation on a STAGGERED TEST BASIS.

4.7.1.6.3 At least once per 18 months, verify operability of the respective standby steam generator feedwater pump by starting each pump and providing feedwater to the steam generators.

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*These pumps do not require plant safety related emergency power sources for operability and the flowpath is normally isolated.

**The Demineralized Water Storage Tank is non-safety grade.

Enclosure 4
Administrative Change #10

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.*

APPLICABILITY: With fuel ~~assemblies in the storage pool.~~

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

~~b. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.9.7 Prior to crane operation over fuel assemblies in the spent fuel storage pool, verify that each load is 2000 pounds or less.

~~*Exception may be taken for the temporary construction crane to be used for the re-rack operation which may be carried over irradiated fuel to facilitate installation of the crane. Lift rigs which meet the design and operational requirements of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" will be used while performing this installation.~~

Enclosure 4

~~Administrative Change~~ #11

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10' the spent fuel storage pool.*

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

10 CFR 50.59

*The requirements of this specification may be suspended for more than 4 hours hours to perform maintenance provided a ~~safety~~ evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

ENCLOSURE 5

CLEAN COPY OF TECHNICAL SPECIFICATIONS PAGES INCORPORATING ADMINISTRATIVE CHANGES

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Enclosure 5
Clean Copy TS Change #1

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DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 204 fuel rods clad with Zircaloy-4 or ZIRLO™, except that replacement of fuel rods by filler rods consisting of stainless steel, or by vacant rod positions, may be made in fuel assemblies if justified by cycle-specific reload analysis using NRC-approved methodology. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches. Should more than 30 individual rods in the core, or 10 fuel rods in any fuel assembly, be replaced per refueling, a Special Report discussing the rod replacements shall be submitted to the Commission within 30 days after cycle startup.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig ± 1%, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The nominal water and steam volume of the Reactor Coolant System is 9343 cubic feet at a nominal T_{avg} of 574.2°F.

5.5 FUEL STORAGE

5.5.1 CRITICALITY

5.5.1.1 The spent fuel storage racks are designed to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison and shall be maintained with:

a. k_{eff} equivalent to less than 1.0 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in WCAP-14416-P.

b. A k_{eff} equivalent to less than or equal to 0.95 when flooded with borated to 650 ppm water, which includes a conservative allowance for uncertainties as described in WCAP-14416-P.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for two region fuel storage racks.

d. The maximum enrichment loading for fuel assemblies is 4.5 weight percent of U-235.

5.5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

DESIGN FEATURES

5.5.1.3 Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks. Administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

DRAINAGE

5.5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1404 in two region storage racks

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6-1.

TABLE 5.6-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	150 leak tests.	Pressurized to ≥ 2435 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3100 psig.
Secondary Coolant System	6 loss of secondary pressure	Loss of Secondary pressure
	50 leak tests	Pressurized to ≥ 1085 psig
	35 hydrostatic pressure tests.	Pressurized to ≥ 1356 psig.

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DEFINITIONS

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DEFINITIONS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or 0-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with NRC approved methodology. Unit operation within these operating limits is addressed in individual specifications. The COLR is submitted to the NRC in accordance with the requirements of 6.9.1.7.

DIGITAL CHANNEL OPERATIONAL TEST

1.11 A DIGITAL CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock, and/or trip functions.

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

E-AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 13.5 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.20 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PURGE - PURGING

1.21 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.22 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.23 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2300 MWt.

REPORTABLE EVENT

1.24 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.25 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.26 The SITE BOUNDARY shall mean that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

1.27 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.28 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.29 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.30 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.31 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.32 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.33 An UNRESTRICTED AREA shall mean an area, access to which is neither limited nor controlled by the licensee.

VENTILATION EXHAUST TREATMENT SYSTEM

1.34 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.35 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

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TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \quad \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \frac{1}{(1+\tau_6 S)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:	ΔT	=	Measured ΔT by RTD Instrumentation
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead/Lag compensator on measured ΔT; τ_1 = 0s, τ_2 = 0s
	$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT; τ_3 = 0s
	ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
	K_1	=	1.24;
	K_2	=	0.017/°F;
	$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
	τ_4 , τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 25s, τ_5 = 3s;
	T	=	Average temperature, °F;
	$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg}; τ_6 = 0s
	T'	≤	577.2 °F (Nominal T_{avg} at RATED THERMAL POWER);
	K_3	=	0.001/psig;
	P	=	Pressurizer pressure, psig;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 S}{1+\tau_7 S} \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \frac{1}{1+\tau_6 S} - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1,
$\frac{1}{1+\tau_3 S}$	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	≤	1.10,
K_5	≥	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 s}{1+\tau_7 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,
$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1,

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. #71249)

LIMITING ART VALUES AT 32 EFPY: 1/4 T, 262°F
3/4 T, 218°F

FPL 32 EFPY COOLDOWN CURVES

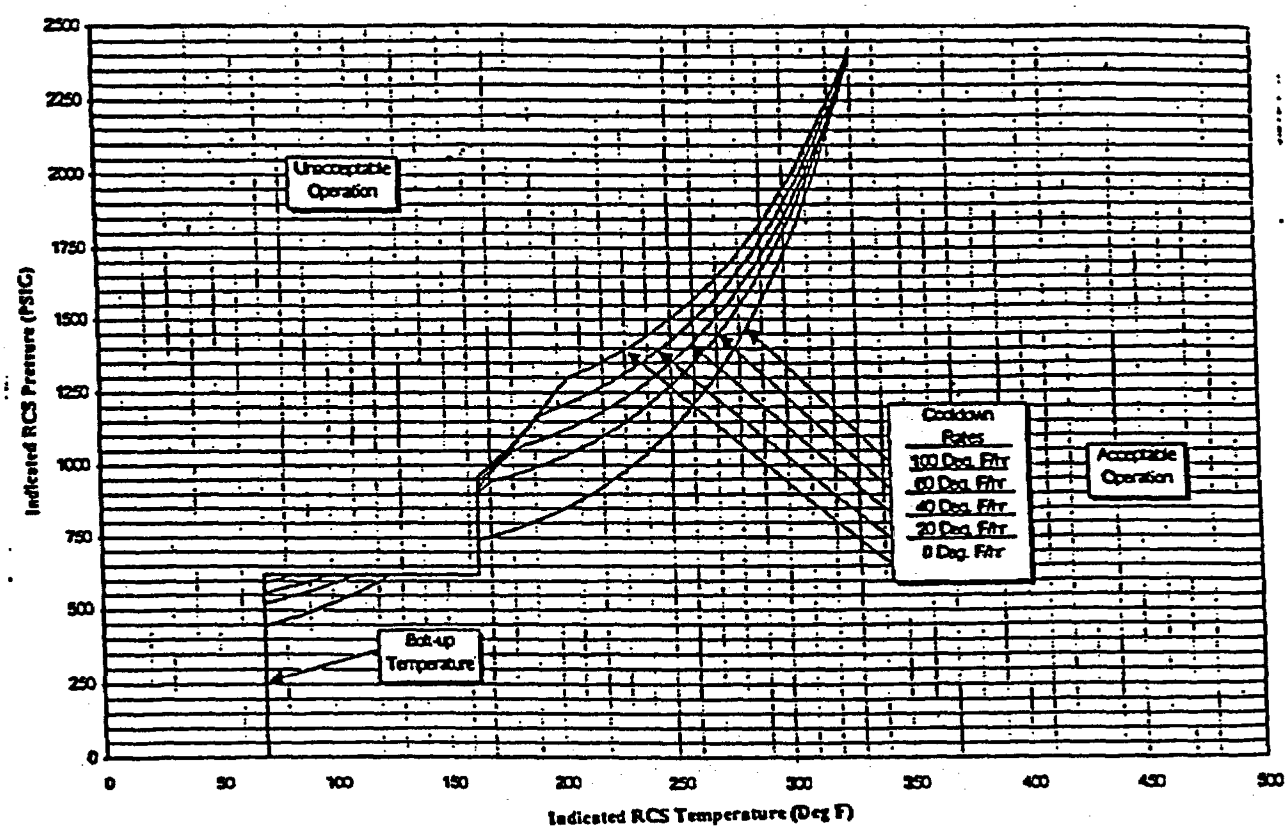

FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rate of 0, 20, 40, 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

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REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 468 psig, or

b. The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in Modes 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

e. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

f. The provisions of Specification 3.0.4 are not applicable.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

* Exception may be taken under Administrative Controls for opening of valves and airlocks necessary to perform surveillance, testing requirements and/or corrective maintenance. In addition, Specification 3.6.4 shall be complied with.

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least 135,000 gallons of water (indicated volume), shall be in the Demineralized Water Storage Tank.**

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one Standby Steam Generator Feedwater Pump inoperable, restore the inoperable pump to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both Standby Steam Generator Feedwater Pumps inoperable, restore at least one pump to OPERABLE status within 24 hours, or:

   1. Notify the NRC within the following 4 hours, and provide cause for the inoperability and plans to restore pump(s) to OPERABLE status and,

   2. Submit a SPECIAL REPORT per 3.7.1.6d.

c. With less than 135,000 gallons of water indicated in the Demineralized Water Storage Tank restore the available volume to at least 135,000 gallons indicated within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits at least once per 24 hours.

4.7.1.6.2 At least monthly verify the standby feedwater pumps are OPERABLE by testing in recirculation on a STAGGERED TEST BASIS.

4.7.1.6.3 At least once per 18 months, verify operability of the respective standby steam generator feedwater pump by starting each pump and providing feedwater to the steam generators.

*These pumps do not require plant safety related emergency power sources for operability and the flowpath is normally isolated.

**The Demineralized Water Storage Tank is non-safety grade.

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REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Prior to crane operation over fuel assemblies in the spent fuel storage pool, verify that each load is 2000 pounds or less.

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REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10' the spent fuel storage pool.*

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

*The requirements of this specification may be suspended for more than 4 hours hours to perform maintenance provided a 10 CFR 50.59 evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.