FPL

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L-2005-008
10 CFR 50.90

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

RE: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendment
Adoption of Improved Standard Technical Specification (ISTS) Travelers

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4. The proposed amendments revise the Technical Specifications (TS) to incorporate eight generic changes that have been made to the Improved Standard Technical Specifications (ISTS), NUREG-1431, "Standard Technical Specifications for Westinghouse Plants." These generic changes, known as Travelers, were developed by the Technical Specifications Task Force (TSTF) and have been previously reviewed and approved by the Nuclear Regulatory Commission. The eight Travelers are:

1. TSTF-5, Rev. 1, "Delete safety limit violation notification requirements."
2. TSTF-93, Rev. 3, "Change the frequency of pressurizer heater testing from 92 days to [18] months."
3. TSTF-95, Rev. 0, "Revise completion time for reducing Power Range High trip setpoint from 8 to 72 hours."
4. TSTF-101, Rev. 0, "Change AFW pump testing frequency to be 'In accordance with the Inservice Testing Program.'" Also delete the plant specific pressure & flows to be consistent with the ISTS.
5. TSTF-258, Rev. 4, "Changes to Section 5.0, Administrative Controls."
6. TSTF-299, Rev. 0, "Administrative Controls Program 5.5.2.b Test Interval and Exception."
7. TSTF-308, Rev. 1, "Determination of Cumulative and Projected Dose Contributions in RECP."
8. TSTF-361, Rev. 2, "Allow standby SDC/RHR/DHR loop to inoperable to support testing."

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Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Adoption of Improved Standard Technical Specification (ISTS) Travelers

Enclosure 1 provides the proposed License Amendments description and justification. Enclosure 2 is the No Significant Hazards Considerations and Enclosure 3 provides the Environmental Consideration. Enclosure 4 provides the marked-up copies of the proposed Technical Specification changes and an information copy of the proposed changes to the TS Bases. Enclosure 5 is a copy of the re-typed TS pages. Finally, Enclosure 6 provides a List of Regulatory Commitments. FPL has determined that the proposed License Amendments do not involve a significant hazards consideration pursuant to 10 CFR 50.92.

The proposed license amendments are similar in nature to other NRC approved industry license amendments associated with adopting Travelers. Specific references are included in Enclosure 1.

The Turkey Point Plant Nuclear Safety Committee and the Florida Power & Light Company Nuclear Review Board have reviewed the proposed amendments. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

FPL requests that the proposed License Amendments be approved by December 31, 2005. Please issue the amendment to be effective on the date of issuance and to be implemented within 60 days of receipt by FPL.

Please contact Mr. Walter J. Parker, Licensing Manager, Turkey Point Units 3 and 4 at 305-246-6632 if there are any questions about this submittal.

Very truly yours,

Terry Jones

Terry O. Jones
Vice President
Turkey Point Plant

Enclosures

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point
USNRC Project Manager, Turkey Point
W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Adoption of Improved Standard Technical Specification (ISTS) Travelers

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]

Sworn to and subscribed before me this

20 day of JANUARY, 2005

Darius D. Reid
Name of Notary Public (Type or Print)

DARIUS D. REID
Notary Public - State of Florida
My Commission Expires Jan 3, 2007
Commission # DD156688
Bonded By National Notary Assn.

Terry O. Jones is personally known to me.

Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendments
Adoption of Improved Standard Technical Specification (ISTS) Travelers

TABLE OF CONTENTS

Enclosures

- ENCLOSURE 1 — Proposed License Amendments Description and Justification
- ENCLOSURE 2 — No Significant Hazards Considerations
- ENCLOSURE 3 — Environmental Consideration
- ENCLOSURE 4 — Proposed Mark-Up of Affected Technical Specification and Bases Pages
- ENCLOSURE 5 — Re-typed Technical Specification Pages and Bases Pages (for information only)
- ENCLOSURE 6 — List of Regulatory Commitments

ENCLOSURE 1

PROPOSED LICENSE AMENDMENTS

DESCRIPTION AND JUSTIFICATION

Introduction

The proposed changes revise the Turkey Point Technical Specifications (TS) to adopt eight generic improvements, known as Travelers. These Travelers have been incorporated into subsequent revisions of the Improved Standard Technical Specifications (ISTS), including NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," and are applicable to the Turkey Point TS as described below. These changes have been reviewed generically and approved by the Nuclear Regulatory Commission (NRC).

Background

Since Revision 1 of the ISTS was published in 1995, the industry and the NRC staff have identified additional ISTS improvements. These improvements are proposed by the industry Technical Specifications Task Force (TSTF) and referenced by a TSTF number. Following industry acceptance and NRC staff approval, the NRC incorporates the Traveler into the ISTS. In most cases, these changes are generally applicable to individual plants and may be adopted by license amendment into plant-specific Technical Specifications.

The Turkey Point TS are not based on the ISTS. However, these Travelers, including the justification accepted by the NRC, are applicable to the Turkey Point TS, as described below. Note that when quoting text from the ISTS, text that is plant-specific is surrounded by brackets (i.e., "[" and "]").

Description and Justification of the Proposed Changes

Each Traveler is treated as a separate and complete entity in order to facilitate the presentation and review of the proposed changes. The following format is used for the descriptions and justifications of the proposed changes:

- Traveler Title – identifies the Traveler number and title
- NRC Approval – provides the reference for the approval of the Traveler
- Description of Proposed Turkey Point TS Change – provides a summary description of the proposed TS change associated with this Traveler
- Comparison of the Turkey Point TS Requirements to the ISTS and Differences between the Proposed TS Changes and the Traveler – provides a description of the specific changes to the ISTS that were made by the Traveler together with a description of the proposed specific changes to the Turkey Point TS. An assessment of the differences between the proposed TS changes and the Traveler is also provided.
- Justification – provides the justification for the proposed TS change

- Turkey Point TS Pages Affected – lists the TS pages that are being changed
- Turkey Point TS Bases Pages Affected – lists the TS Bases pages that are being changed
- Licensing Precedent – provides references for previous license amendment requests based on this Traveler, and the associated approval by the NRC.

This is the presentation format for each of the eight Travelers and the associated Turkey Point TS changes.

Traveler Title: TSTF-5, Revision 1, "Delete safety limit violation notification requirements."

NRC Approval: Letter from C. I. Grimes (NRC) to J. Davis (NEI) dated September 27, 1996.

Description of Proposed Turkey Point TS Change
The proposed change deletes notification, reporting, and restarts requirements from the TS if a safety limit is violated. Section 6.7 of the Turkey Point TS is deleted and is being replaced with the term "DELETED," and the references to Specification 6.7.1 are being deleted from TS 2.1.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler
TSTF-5 deleted ISTS Safety Limit (SL) Violation action 2.2.3, which required notifying the NRC Operations Center within 1 hour in accordance with 10 CFR 50.72, action 2.2.4, which required notifying the [Plant Superintendent and Vice President - Nuclear Operations] within 24 hours, action 2.2.5, which required submitting a Licensee Event Report to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President - Nuclear Operations] pursuant to 10 CFR 50.73 within 30 days, and action 2.2.6, which stated that operation of the unit shall not be resumed until authorized by the NRC.

Turkey Point TS Section 2.1, "Safety Limits," Specification 2.1.1, Action, and Specification 2.1.2, Action, state that if a Safety Limit is exceeded, comply with the requirements of Specification 6.7.1. Turkey Point TS Section 6.7, "Safety Limit Violation," Specification 6.7.1 requires notifying the NRC Operations Center within one hour in accordance with 10 CFR 50.72, notifying the Chief Nuclear Officer (CNO) and the Company Nuclear Review Board (CNRB) within 24 hours, preparing and submitting a Licensee Event Report (LER) pursuant to 10 CFR 50.73, and submitting the LER to the CNRB and the CNO within 30 days, and prohibits critical operation of the unit until authorized by the NRC.

The ISTS requirements that were deleted by TSTF-5 and the Turkey Point TS requirements are equivalent. The remaining Turkey Point TS and ISTS Safety Limit Violation actions are equivalent.

Justification
This change deletes requirements from TS that are duplicative or contained in other regulations or required to comply with regulations, as described below:

Title 10 of the Code of Federal Regulations, Part 50.36(c)(l)(i)(A) states, "If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73."

Specification 6.7.1.a requires in part that the NRC Operations Center be notified by telephone as soon as practical and in all cases within one hour after a safety limit has been violated, in accordance with 10 CFR 50.72. This part of the specification duplicates 10 CFR 50.36(c)(1)(I)(A) and 10 CFR 50.72. 10 CFR 50.36(c)(1)(I)(A) requires notifying the Commission in accordance with 10 CFR 50.72 if any safety limit is exceeded.

Specification 6.7.1.a also requires that the CNO and the CNRB be notified within 24 hours of a safety limit violation. The 24 hour limit to inform the CNO and the CNRB may be deleted without significant consequence, because an event as significant as the violation of a safety limit will be reported to company management and the review board in accordance with company policies and procedures. The timing and reporting to management and internal oversight organizations is a Company-internal procedural issue not appropriate for Technical Specifications.

Specification 6.7.1.b requires that a Licensee Event Report (LER) be prepared in accordance with 10 CFR 50.73. This specification duplicates the requirements of CFR 50.36(c)(l)(i)(A) which requires the submission of a Licensee Event Report to the Commission as required by § 50.73 and by 10 CFR 50.73 (a)(2)(I)(A), which requires the submittal of a LER for, "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."

Specification 6.7.1.c requires in part that the LER be submitted to the Commission in accordance with 10 CFR 50.73. This action is duplicative of the requirements of 10 CFR 50.36(c)(l)(i)(A) and is merely a directive to follow a federal regulation. This action does not need to be stated in the TS.

Specification 6.7.1.c also requires that the LER be submitted to the CNRB and the CNO within 30 days after discovery of the event. The Topical Quality Assurance Report (TQAR) and Chapter 12 of the Turkey Point Updated Final Safety Analysis Report (UFSAR) describe the process for on-site and off-site safety committee reviews of reportable events. The timing of reporting to management and internal oversight organizations is a Company-internal procedural issue not appropriate for the Technical Specifications.

Specification 6.7.1.d requires that critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory Commission. This specification duplicates the requirements of 10 CFR 50.36(c)(1)(i)(A) which requires that "Operation must not be resumed until authorized by the Commission" if any safety limit is exceeded.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
Index page xvi
2-1
2-2 (for information only)
6-12

Turkey Point TS Bases Pages Affected
None

Licensing Precedent
Grand Gulf Unit 1 requested adoption of TSTF-5, Revision 1, on August 20, 1999. The request was approved by the NRC on June 30, 2000 as Amendment 142.

Traveler Title: TSTF-93, Revision 3, "Change the Frequency of Pressurizer Heater Testing from 92 Days to [18] Months"

NRC Approval: The Traveler was approved on October 3, 1997 and was incorporated in Revision 2 of NUREG-1431, dated April 2, 2002.

Description of Proposed Turkey Point TS Change
The proposed change to Specification 3.4.3, "Pressurizer," revises the pressurizer heater testing frequency in Surveillance 4.4.3.2 from 92 days to 18 months and eliminates details on how the test is performed.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler
TSTF-93 revised ISTS Surveillance Requirement 3.4.9.2, which states, "Verify capacity of each required group of pressurizer heaters is > [125] kW." The Surveillance Frequency of 92 days was changed to [18] months. The Surveillance was modified by adding a Reviewer's Note which states, "The frequency for performing pressurizer heater capacity testing shall be either 18 months or 92 days, depending on whether or not the plant has dedicated safety-related heaters. For dedicated safety-related heaters, which do not normally operate, 92 days is applied. For non-dedicated safety-related heaters, which normally operate, 18 months is applied."

Turkey Point TS Surveillance Requirement 4.4.3.2 states, "The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days." The Surveillance Requirement is being revised to be consistent with the ISTS by stating, "The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 125 kW at least once per 18 months." The Turkey Point pressurizer heaters are non-dedicated safety-related heaters which normally operate.

The ISTS Surveillance Requirement affected by this Traveler and the revised Turkey Point Surveillance Requirement are equivalent.

Justification
The frequency for Turkey Point Surveillance Requirement 4.4.3.2 is being changed to once per 18 months. NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," Section 6.6, "Pressurizer Heaters," states, "Most pressurizer heaters are in constant use, both the proportional and to some extent the backup heaters. Therefore, operators should be aware of problems that may arise with pressurizer heaters. In addition, pressurizer heaters are fairly reliable." The TS surveillance tests for both units for the last five years were reviewed to assess the historical test results and the performance of the pressurizer heaters. The results of the review confirmed that the TS surveillance requirement was always met. The NRC recommended in NUREG-1366 that pressurizer heaters be tested once each refueling interval at plants that meet this criteria. TSTF-93 implemented this recommendation in the ISTS. As described above, the Reviewer's Note states that plants that have non-dedicated safety-related heaters, the appropriate surveillance requirement interval is 18 months. The Turkey Point pressurizer heaters are non-dedicated safety-related heaters

that normally operate. Therefore, the appropriate interval for the Turkey Point surveillance requirement is 18 months.

The existing Turkey Point Surveillance Requirement specifies how to perform the surveillance by stating "by energizing the heaters and measuring circuit current." This phrase is being deleted. It is not necessary for the TS to state the method of verifying that a limit is met. Information on acceptable methods for performing the testing is being located in the Bases. This presentation is consistent with the ISTS.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
3/4 4-9

Turkey Point TS Bases Pages Affected
Page 52

Licensing Precedent
There are no identified previous licensing actions for adoption of TSTF-93. However, TSTF-93 is based on a recommendation in NUREG-1366. NUREG-1366 was distributed to licensees as an NRC recommended improvement under Generic Letter 93-06, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operations," Item 6.6. There are multiple precedents for adopting this change. An example is Byron Units 1 and 2 and Braidwood Units 1 and 2 which requested adoption of the same changes that are in TSTF-93 on October 3, 1995. The request was approved by the NRC in a letter dated April 10, 1996 as Byron Amendment 82 and Braidwood Amendment 84.

Traveler Title: TSTF-95, Revision 0, "Revise Completion Time for Reducing Power Range High Trip Setpoint from 8 to 72 Hours"

NRC Approval: Letter from C. I. Grimes (NRC) to J. Davis (NEI) dated September 27, 1996.

Description of Proposed Turkey Point TS Change

The proposed change extends the time allowed to reduce the Power Range Neutron Flux - High trip setpoints to 72 hours following a power reduction due to Axial Flux Difference, $F_Q(Z)$, $F^N_{\Delta H}$, or Quadrant Power Tilt Ratio exceeding the limit. The affected specifications are 3.2.1, "Axial Flux Difference, 3.2.2, "Heat Flux Hot Channel Factor - F_Q (Z)," 3.2.3, "Nuclear Enthalpy Rise Hot Channel Factor," 3.2.4, "Quadrant Power Tilt Ratio," 3.1.3 "Movable Control Assemblies," and 3.3.1 "Reactor Trip System Instrumentation."

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler

TSTF-95 revised the completion times for required actions that require reducing the Power Range Neutron Flux - High trip setpoint. ISTS Specification 3.2.1, $F_Q(Z)$, states that with $F_Q(Z)$ not within the limit, reduce thermal power > 1% RTP for each 1% $F_Q(Z)$ exceeds the limit and reduce the Power Range Neutron Flux - High trip setpoints > 1% for each 1% $F_Q(Z)$ exceeds the limit. ISTS Specification 3.2.2, $F^N_{\Delta H}$, states that with $F^N_{\Delta H}$ not within the limit, reduce thermal power to < 50% RTP and reduce the Power Range Neutron Flux - High trip setpoints to < 55% RTP. TSTF-95 extended the completion times to reduce the Power Range Neutron Flux - High trip setpoints from 8 hours to 72 hours.

Turkey Point TS 3.2.2, Action a, states that when $F_Q(Z)$ is not within the limit, reduce thermal power 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and reduce the Power Range Neutron Flux - High trip setpoints within the next 4 hours. TS 3.2.3, Action a.2, states that when $F^N_{\Delta H}$ exceeds the limit, reduce thermal power to less than 50% of rated thermal power within 2 hours and reduce the Power Range Neutron Flux - High trip setpoints to less than or equal to 55% of rated thermal power within the next 4 hours. The action times to reduce the Power Range Neutron Flux - High trip setpoints are being changed from the next 4 hours to 72 hours after entering the action.

The ISTS requirements affected by this Traveler and the revised Turkey Point TS requirements are equivalent.

In addition, Turkey Point TS 3.2.1, Action a.2, states that when the Axial Flux Difference is outside the limits, reduce thermal power to less than 50% of rated thermal power within 30 minutes and reduce the Power Range Neutron Flux - High trip setpoint to less than 55% of rated thermal power within the next 4 hours. TS 3.2.4, Actions a.2.b), a.3, b.3, and c.2, also require reducing thermal power for various conditions in which the Quadrant Power Tilt Ratio limit is exceeded and reducing the Power Range Neutron Flux - High trip setpoint within the next 4 hours. The equivalent specifications in the ISTS, 3.2.3, Axial Flux Difference (AFD) and 3.2.4, Quadrant Power Tilt Ratio (QPTR), do not require reducing the Power Range Neutron Flux - High trip setpoint. The Turkey Point TS 3.2.1 and 3.2.4 actions are being revised to allow 72 hours after exceeding the limit to reduce the Power Range Neutron Flux - High trip setpoint consistent with the changes made by TSTF-95 and the proposed changes to TS 3.2.2 and 3.2.3, described above.

Similarly, Turkey Point TS 3.1.3.1, Action (d)(3)(a) states that when a full length rod is declared inoperable, and the shutdown margin requirement of Specification 3.1.1.1 is satisfied, power operation may continue provided that the thermal power level is reduced to less than or equal to 75% RTP and within the next 4 hours, the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RTP. The equivalent specification in the ISTS, Rod Group Alignment Limits, Section 3.1.4, action B, requires the thermal power be reduced to 75% RTP, but does not require reducing the Power Range Neutron Flux – High trip setpoint. The Turkey Point TS 3.1.3.1 actions are being revised to allow 72 hours after exceeding the limit to reduce the Power Range Neutron Flux - High trip setpoint consistent with the changes made by TSTF-95 and the proposed changes to TS 3.2.2 and 3.2.3, described above.

In addition, Turkey Point TS 3.3.1, Table 3.3-1, action 2, for Power Range, Neutron Flux listed as a "functional unit" in Table 3.3-1, "Reactor Trip System Instrumentation" requires that with the number of operable channels one less than the total number of channels, startup and/or power operation may proceed provided certain conditions are satisfied. "Condition c" is to either restrict thermal power to less than or equal to 75% of RTP and adjust the Power Range Neutron Flux Trip Setpoint to less than or equal to 85% of RTP within 4 hours; or monitor the Quadrant Power Tilt Ratio per Specification 4.2.4.2. The Turkey Point action 2 is being revised to adjust the Power Range Neutron Flux Trip Setpoint to less than or equal to 85% of RTP within 72 hours, rather than 4 hours, as part of condition c. The equivalent ISTS TS is 3.3.1, "Reactor Trip Instrumentation," Condition D and does not require reducing the Power Range Neutron Flux – High trip setpoint.

Justification

The action time of 4 hours to reduce the Power Range Neutron Flux-High trip setpoints after a power reduction presents an unjustified burden on the operation of the plant. The proposed action time of 72 hours after exceeding the limit will allow time to perform a second flux map to confirm the results, or determine that the condition was temporary, without implementing an unnecessary trip setpoint change. Following a significant power reduction, at least 24 hours are required to reestablish steady state xenon prior to taking a flux map. An additional 8 to 12 hours is required to obtain a flux map, and analyze the data. Extending the time to reduce the trip setpoints will allow time for this analysis.

Delaying the reduction in the trip setpoints until after the condition can be confirmed eliminates a potential trip initiator. A significant potential for human error is created by requiring the trip setpoints to be reduced within the same time frame that a unit power reduction is taking place. Setpoint adjustment is estimated to take approximately 4 hours per channel, which includes establishment of plant condition supportive of removing a channel from service, tripping of bistables, setpoint adjustments, and channel restoration. An additional 2 hours is estimated for necessary initial preparations, such as pre-job brief, calibration equipment checks, and obtaining tools and approvals. Therefore, to reduce the trip setpoint for all 4 channels can reasonably take 18 hours.

The proposed action time of 72 hours after exceeding the limit is reasonable given the time to reduce power, providing 36 hours to perform and analyze a flux map to confirm the limit is exceeded, and 18 hours to reduce the trip setpoints.

The 72 hour action time is sufficient considering the small likelihood of a severe transient in this period and the preceding prompt reduction in thermal power in accordance with the TS actions.

The ISTS Actions for Axial Flux Difference, Rod Group Alignment Limits, Reactor Trip System Instrumentation, and Quadrant Power Tilt Ratio do not require reducing the Power Range Neutron Flux - High trip setpoints. Nevertheless, it is proposed to conservatively maintain an action to reduce the Power Range Neutron Flux - High trip setpoints for Turkey Point, but to change the action time to 72 hours after exceeding the limit. The justification for the change to provide 72 hours to reduce the Power Range Neutron Flux - High trip setpoint is equally applicable to the Axial Flux Difference, Rod Group Alignment Limits, Reactor Trip System Instrumentation, and Quadrant Power Tilt Ration Actions in the Turkey Point Technical Specifications because the reason for reducing the trip setpoints does not affect the time required to perform the task. Therefore, this change is being made to those actions as well.

TSTF-95 does not affect ISTS 3.1.4 Rod Group Alignment Limits, 3.2.3, Axial Flux Difference (AFD), ISTS 3.2.4, Quadrant Power Tilt Ratio (QPTR), or 3.3.1 Reactor Trip System Instrumentation because the Actions for those specifications do not require reducing the Power Range Neutron Flux – High trip setpoints if the respective limits are not met. The Traveler was only directed at ISTS specification actions that required reducing those trip setpoints. However, TSTF-95 is applicable to Turkey Point TS 3/4.2.1, Axial Flux Difference, 3/4.1.3 Movable Control Assemblies, 3/4.2.4, Quadrant Power Tilt Ratio, and 3/4.3.1 Reactor Trip System Instrumentation, because these specifications require reducing the Power Range Neutron Flux – High trip setpoints if their respective limits are not met. The justification for extending the action time to 72 hours after exceeding the limit for these specifications is the same as described in the Traveler. The specific reasons for reducing the trip setpoints in Turkey Point TS 3/4.2.1, 3/4.1.3, 3/4.2.4 and 3/4.3.1 are different than the ISTS actions changed by the Traveler; that is, different parameters are not meeting their TS limits. However, the action requirement is exactly the same, and so it is appropriate to allow the same amount of time to complete the action.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected

3/4 1-18	3/4 2-13
3/4 2-1	3/4 2-14
3/4 2-4	3/4 2-15
3/4 2-11	3/4 3-5

Turkey Point TS Bases Pages Affected
None

Licensing Precedent
There are no identified previous licensing actions for individual adoption of TSTF-95. However, many plants of Westinghouse design that have converted to the ISTS have adopted TSTF-95 during their ISTS conversion. For example, North Anna Units 1 and 2 submitted a license amendment request to convert to the ISTS on December 11, 2000. The request was approved by the NRC in a letter dated April 5, 2002 as Amendments 232 and 212.

Traveler Title: TSTF-101, Revision 0, "Change AFW pump testing frequency to be 'In accordance with the Inservice Testing Program'"

NRC Approval: Letter from C. I. Grimes (NRC) to J. Davis (NEI) dated September 27, 1996.

Description of Proposed Turkey Point TS Change
The proposed change to TS 3/4.7.1.2, "Auxiliary Feedwater System," revises the surveillance frequency for the Auxiliary Feedwater (AFW) pumps from 31 days on a Staggered Test Bases to a frequency set by the inservice testing (IST) program, and removes the specific flow rate limit. Procedural details are also being deleted from the Surveillance Requirements.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler
TSTF-101 revised ISTS 3.7.5, "Auxiliary Feedwater System." Surveillance Requirement 3.7.5.2 requires verifying the developed head for each AFW pump at the flow test point is greater than or equal to the required developed head. TSTF-101 revised the surveillance frequency from 31 days on a Staggered Test Basis to a frequency set by the Inservice Testing Program. The ISTS surveillance also contains a note which states that the Surveillance is not required to be performed for the turbine driven AFW pump until [24 hours] after ≥ [1000] psig in the steam generator. The purpose of the note is to allow entering the mode of applicability to establish the necessary conditions to perform the test.

Turkey Point Surveillance Requirement 4.7.1.2.1.a.1) requires verifying by control panel indication and visual observation of equipment that each steam-driven pump operates for 15 minutes or greater and verifying that flow is greater than or equal to 373 gpm at the entrance of the steam generators every 31 days on a Staggered Test Basis. Surveillance Requirement 4.7.1.2.1.a.1 also states that the provisions of Specification 4.0.4 are not applicable for entry into Modes 2 and 3. This surveillance is being replaced by a surveillance that is consistent with ISTS SR 3.7.5.2. The revised surveillance requires testing each AFW pump and verifying that the developed head at the flow test point is greater than or equal to the required developed head when tested in accordance with Specification 4.0.5. Turkey Point Specification 4.0.5 is equivalent to the Inservice Testing Program specified in the ISTS. Additionally, the exception to Specification 4.0.4 is being retained as part of the proposed change. Specification 4.0.4 states, "Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with a Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements." The exception to Specification 4.0.4 is equivalent to the ISTS note to allow entering the mode of applicability in order to establish the conditions to perform the surveillance.

The Turkey Point Surveillance 4.7.1.2.1.a.2) requires verifying the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test in Surveillance 4.2.1.2.a.1). This surveillance is being deleted consistent with ISTS 3.7.5.

The ISTS Surveillance Requirement affected by this Traveler and the revised Turkey Point Surveillance Requirement are equivalent.

Justification
This change will result in the testing frequency of the AFW pumps being in accordance with the inservice testing program (i.e., Specification 4.0.5) consistent with other pumps required by the Technical Specifications, such as charging pumps and containment spray pumps. This will eliminate any potential ambiguity associated with AFW pump testing as a result of ASME changes, and results in consistent presentation of pump testing throughout the Technical Specifications. This frequency for testing AFW pumps is consistent with the ASME Code requirements. Such inservice tests confirm component operability, trend performance, and detect incipient failures by indicating abnormal performance.

The other changes to the Surveillance make the testing requirements consistent with the testing requirements of other pumps in the TS, which do not specify:

- a minimum operating time
- flow rate
- that support systems, such as discharge valves and control valves, operate or
- the flow rate test point.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
3/4 7-3
3/4 7-4
3/4 7-5 (for information only)

Turkey Point TS Bases Pages Affected
Page 83
Page 84

Licensing Precedent
There are no identified previous licensing actions for individual adoption of TSTF-101. However, many plants of Westinghouse design that have converted to the ISTS have adopted TSTF-101 during conversion. For example, North Anna Unit 1 and 2 submitted a license amendment request to convert to the ISTS on December 11, 2000. The request was approved by the NRC in a letter dated April 5, 2002 as Amendments 232 and 212.

Traveler Title: TSTF-258, Revision 5, "Changes to Section 5.0, Administrative Controls"

NRC Approval: Letter from W. D. Beckner (NRC) to J. Davis (NEI) dated June 29, 1999.

Description of Proposed Turkey Point TS Change
The proposed change revises the Administrative Controls section of the Technical Specifications to adopt several improvements to unit staffing, working hour limitations, the Radioactive Effluent Controls Program, and reporting.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler
TSTF-258 made seven changes to the Administrative Controls section of the ISTS.

- It deleted requirements for a licensed operator to be present in the control room when fuel is in the reactor vessel and for a senior licensed operator to be in the control room when either unit is in Modes 1, 2, 3, or 4 in ISTS 5.2.2 since 10CFR50.54(k) and (m) require this.
- It eliminated details from working hour limitations from ISTS 5.2.2.
- It clarified the requirements for the Shift Technical Advisor function in ISTS 5.2.2.
- It modified the unit staff requirements in ISTS 5.3 to add elements required to be in the Technical Specifications by 10 CFR 55.4.
- It revised the Radioactive Effluent Controls Program in ISTS 5.5.4 to be consistent with 10 CFR 20 and to make other improvements.
- It revised the Monthly Operating Report in ISTS 5.6.4 to eliminate the requirement to report challenges to the pressurizer power operated relief valves and safety valves.
- It revised the High Radiation Area requirements in ISTS 5.7 based on 10 CFR 20 and the letter from C. Grimes (NRC) to J. Davis (NEI) dated April 9, 1997.

Turkey Point Specification 6.2.2.b contains the same requirement for at least one licensed operator to be in the control room when fuel is in either reactor as is deleted by TSTF-258. Specification 6.2.2.c contains the same requirement for at least one senior licensed operator to be in the control room when either unit is in Mode 1, 2, 3, or 4 as is deleted by TSTF-258. The TS requirements are being deleted to be consistent with the ISTS and the changes made in TSTF-258.

Turkey Point Specification 6.8.5 contains working hour limitations very similar to the changes made in TSTF-258. However, the changes in TSTF-258 related to periodic review of overtime to ensure excessive hours have not been assigned are not consistent with the Turkey Point TS. TSTF-258 replaced the requirement for the plant manager or his designee to review individual overtime monthly with a requirement for periodic independent review. Specification 6.8.5 requires a monthly review by the plant manager or his designee. The TS requirements are being revised to be consistent with the ISTS and the changes made in TSTF-258.

Turkey Point Specification 6.2.3, Shift Technical Advisor, contains requirements on the responsibilities and qualifications of engineering expertise on shift. The TS requirements are being revised to be consistent with the ISTS and the changes made in TSTF-258. The changes include revising the Section title from "Shift Technical Advisor" to "Shift Technical Advisor Function", and deleting the explicit presentation of the educational requirements for the Shift Technical Advisor function.

Turkey Point Specification 6.3, Facility Staff Qualifications, is being modified to include a statement that a senior licensed operator performs the functions described in 10 CFR 50.54(m). This change is administrative and establishes consistency with 10 CFR 55.4. This change is consistent with the addition to the ISTS made by TSTF-258.

The Turkey Point Specification 6.8.4.f, Radioactive Effluent Controls Program, requirements are consistent with the ISTS Radioactive Effluent Controls Program requirements affected by TSTF-258. The changes made to the Radioactive Effluent Controls Program by TSTF-258 are directly applicable to the Turkey Point TS. The TS requirements are being revised to be consistent with the ISTS and the changes made by TSTF-258. The changes include revising Turkey Point Specification 6.8.4.f.2 to be consistent with 10 CFR 20, revising Specifications 6.8.4.f.7 and 6.8.4.f.10 to include editorial clarifications that do not change the intent of the requirements and to be more consistent with the ISTS, and explicitly specifying that the provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequencies.

Turkey Point Specification 6.9.1.5, Monthly Operating Reports, contains a requirement to document all challenges to the pressurizer power operated relief valves (PORVs) or safety valves. TSTF-258 eliminated this requirement. The TS requirements are being revised to be consistent with the ISTS and the changes made by TSTF-258.

Turkey Point Specification 6.12, High Radiation Area, requirements are consistent with the ISTS High Radiation Area requirements that were replaced by TSTF-258. Turkey Point Specification 6.12 is not being revised at this time.

The affected ISTS requirements and the revised Turkey Point TS requirements are equivalent.

Justification

TSTF-258 deleted requirements in ISTS 5.2 for a licensed operator to be present in the control room when fuel is in the reactor vessel and for a senior licensed operator to be in the control room when either unit is in Modes 1, 2, 3, or 4. These requirements appear in Turkey Point Specifications 6.2.2.b and 6.2.2.c. The deleted requirements are not needed in the Technical Specifications as they are already required by the Code of Federal Regulations. The requirements of 10 CFR 50.54(m)(2)(iii) and 50.54(k) adequately provide for shift manning. These regulations, 50.54(m)(2)(iii), require "when a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." Further, 50.54(k) requires "An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility." The requirements will be met through compliance with these regulations and are not required to be reiterated in the Technical Specifications.

TSTF-258 eliminated specific working hour limitations from ISTS 5.2.2. Turkey Point Specification 6.8.5 is being revised to delete the requirement that controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the plant manager or his designee to ensure that excessive hours have not been assigned. There is no guidance in Generic Letter 82-12, "Nuclear Power Plant Staff Working Hours," that discusses these additional controls. The additional requirement to have the plant manager (or his designee) review individual overtime on a monthly bases is unnecessary since sufficient administrative controls and policies exist, as well as the role of the individual's supervisors, in preventing excessive or abuse of overtime. The requirement for the monthly review by the plant manager or his designee is being replaced by a requirement to perform a periodic independent review.

TSTF-258 clarified the requirements for the Shift Technical Advisor function in ISTS 5.2.2. Turkey Point Specification 6.2.3 is revised to eliminate the title of "Shift Technical Advisor" (STA) so that it does not imply that the STA and the Shift Supervisor must be different individuals. Option 1 of the Commission Policy Statement on Engineering Expertise on Shift is satisfied by assigning an individual with specified educational qualifications to each operating crew as one of the SROs (preferably the shift supervisor) required by 10 CFR 50.54(m)(2)(i) to provide the technical expertise on shift. This change is also consistent with note *** of Table 6.2-1, "Minimum Shift Crew Composition." In addition, the requirement that the advisor have a bachelor's degree or equivalent in a scientific or engineering discipline is being deleted. Educational requirements are specified in the Policy Statement on Engineering Expertise on Shift, which is referenced in Specification 6.2.3, and do not need to be repeated in the Technical Specifications. Eliminating this phrase is consistent with the ISTS requirements.

TSTF-258 modified the unit staff requirements of ISTS 5.3 to add elements required to be in the Technical Specifications by 10 CFR 55.4. The definitions in 10 CFR 55.4 state: "Actively performing the functions of an operator or senior operator means that an individual has a position on the shift crew that requires the individual to be licensed as defined in the facility's technical specifications, and that ...." TSTF-258 added to ISTS 5.3, "For the purpose of 10 CFR 55.4, a licensed Senior Operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m)." A similar paragraph is added to Turkey Point Specification 6.3. Adding a paragraph to Specification 6.3 referencing 10 CFR 55.4 ensures that there is no misunderstanding when complying with 10 CFR 55.4 requirements. Adding this paragraph is consistent with the recommendations in the April 9, 1997 letter from C. Grimes (NRC) to J. Davis (NEI).

TSTF-258 revised the Radioactive Effluent Controls Program in ISTS 5.5.4 to be consistent with 10 CFR 20 and to make other improvements. Turkey Point Specification 6.8.4.f.2 is being revised to specify a limit "conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402." Enclosure 3 to Generic Letter 89-01 provided model Technical Specifications that satisfied the requirements of 10 CFR 20.106, which was the current applicable regulatory requirement at the time. The Turkey Point TS requirements are consistent with Generic Letter 89-01. In 1991, Part 20 was revised. The change to Specification 6.8.4.f.2 is intended to eliminate possible confusion or improper implementation of the revised 10 CFR Part 20 requirements. Specifications 6.8.4.f.7 and 6.8.4.f.10 are being revised to include editorial clarifications that do not change the intent of the requirements and to be more consistent with the ISTS. The provisions of Specification 4.0.2 are being applied to the Radioactive Effluent Controls Program surveillance frequencies to allow for scheduling flexibility. Specification 4.0.2 permits a 25% extension of the specified frequency (31 days). Allowing a 25% extension of the frequency of performing the cumulative dose and projected dose calculation will have no affect on the outcome of the calculations. The provisions of Specification 4.0.3 are also being applied to the Radioactive Effluent Controls Program. Specification 4.0.3 specifies the requirements if a Surveillance Requirement is missed. Adding these requirements to the Radiological Effluent Controls Program establishes consistency with the other Surveillance Requirements in the TS, and is consistent with the ISTS and the changes made by TSTF-258.

TSTF-258 revised the monthly operating report requirements in ISTS 5.6.4 to eliminate the requirement to report challenges to the pressurizer power operated relief valves (PORVs) and safety valves. The reporting of pressurizer safety and relief valve failures and challenges is based on the guidance of NUREG-0694, "TMI-Related Requirements for New Operating Licensees." This guidance states, "Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report." NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report," requested the submittal of less information in the monthly operating report. The Generic Letter specifies information that needs to be reported to support the NRC Performance Indicator Program, and availability and capacity statistics. The Generic Letter does not specify that challenges to the pressurizer PORVs and safety valves should be reported. Given that the NRC no longer requires the reporting of this information for the Performance Indicator Program, it is acceptable to delete the requirement to provide documentation of all challenges to the PORVs or safety valves from Specification 6.9.1.5.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
xv
6-2
6-5
6-16
6-17
6-18
6-20

Turkey Point TS Bases Pages Affected
None

Licensing Precedent

Calvert Cliffs Units 1 and 2 requested adoption of TSTF-258, Revision 4, on June 11, 2002. The request was approved by the NRC on July 16, 2003 as Amendments 259 and 236.

Traveler Title: TSTF-299, Revision 0, "Administrative Controls Program 5.5.2.b Test Interval and Exception"

NRC Approval: Letter from W. D. Beckner (NRC) to A. R. Pietrangelo (NEI) dated October 31, 2000.

Description of Proposed Turkey Point TS Change
The proposed change clarifies the meaning of "refueling cycle" for system integrated leak test intervals in the Primary Coolant Sources Outside Containment program in Specification 6.8.4.a. In addition, the proposed change also specifies that the provisions of Specification 4.0.2 are applicable to these test intervals.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler
TSTF-299 revised the Primary Coolant Sources Outside Containment program in ISTS 5.5.2. The Primary Coolant Sources Outside Containment program requires that the program include integrated leak test requirements for each system at refueling cycle intervals or less. TSTF-299 revised the program to require the integrated leak test requirements for each system at least every 18 months. In addition, the program was modified to state that the requirements of Surveillance Requirement 3.0.2 are applicable.

Turkey Point Specification 6.8.4.a, Primary Coolant Sources Outside Containment, is consistent with the program in the ISTS and contains the same requirements. The Turkey Point program is being revised to require testing at least every 18 months instead of refueling cycle intervals or less. The program is also being revised to specify that the requirements of Specification 4.0.2 are applicable. Specification 4.0.2 is equivalent to ISTS Surveillance Requirement 3.0.2. Both allow the frequency of testing to be extended by 25%.

The ISTS requirements affected by this Traveler and the revised Turkey Point TS requirements are equivalent.

Justification
The fixed testing frequency of 18 months is more precise than the existing frequency of at least once per refueling cycle and is more consistent with similar requirements in the Turkey Point TS. The system leak testing is similar to a Surveillance Requirement. Specification 4.0.2 permits Surveillance Requirement frequencies to be extended by 25%. For consistency with the other Surveillance Requirements, the provisions of Specification 4.0.2 are being applied to the system leak testing. The applicability of Specification 4.0.2 must be explicitly stated in Specification 6.8.4.a because Specification 4.0.2 only applies to the Surveillance Requirement sections (Sections 4.1 through 4.10). The Specification 4.0.2 provision also provides the needed flexibility to perform the leak testing during a refueling outage should the previous fuel cycle be extended due to a lengthy forced shutdown. The revised test interval combined with the provisions of Specification 4.0.2 is equivalent to the existing requirement, provided the interval between refueling outages is no greater

than 22.5 months (18 months plus 25%) for plants on an 18 month fuel cycle. As Turkey Point Units 3 and 4 are on an 18 month fuel cycle, this change is administrative.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
6-14

Turkey Point TS Bases Pages Affected
None

Licensing Precedent
Calvert Cliffs Units 1 and 2 requested adoption of TSTF-299, Revision 0, on June 11, 2002. The request was approved by the NRC on July 16, 2003 as Amendments 259 and 236.

Traveler Title: TSTF-308, Revision 1, "Determination of Cumulative and Projected Dose Contributions in RECP"

NRC Approval: June 27, 2000 as referenced in the NRC Safety Evaluation contained in the letter from G. S. Vissing (NRC) to P. E. Katz (Calvert Cliffs) dated July 16, 2003.

Description of Proposed Turkey Point TS Change

The proposed change to Specification 6.8.4.f, "Radioactive Effluent Controls Program," clarifies the requirements for the determination of cumulative and projected dose contributions from radioactive effluents.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler

TSTF-308 revised the Radioactive Effluent Controls Program requirements for the determination of cumulative and projected dose contributions to describe the actual intent of the requirement (ISTS Specification 5.5.4.e). The ISTS stated, "Determination of cumulative and projected dose contributions from radioactive effluent for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days." TSTF-308 revised the requirement to state, "Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days."

Turkey Point TS 6.8.4.f, "Radioactive Effluent Controls Program," is consistent with the program in the ISTS. The current paragraph (6.8.4.f.5) regarding determination of cumulative and projected dose contributions from radioactive effluents is worded the same as the paragraph that was revised by TSTF-308. The wording being adopted in the Turkey Point TS is consistent with the change made by TSTF-308.

The ISTS Radioactive Effluent Controls Program requirements affected by this Traveler and the revised Turkey Point TS requirements are equivalent.

Justification

TSTF-308 was proposed because there was concern that the text of this requirement can be misinterpreted to require determining projected dose contribution for the current calendar quarter and current calendar year every 31 days. The current Turkey Point wording was specified in Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETSs) in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETSs to the Offsite Dose Calculation Manual or to the Process Control Program." Generic Letter 89-01 provided new programmatic controls for radioactive effluents and radiological environmental monitoring that were to be incorporated into the TS to conform to the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I of 10 CFR Part 50. The pre-Generic Letter 89-01 requirements required determining the cumulative dose contributions every 31 days, and required determining projected dose contributions every 31 days, but did not require determining projected dose contributions for the current calendar quarter and current calendar year every 31 days.

The NRC Staff's draft standard technical specifications for 4-loop Westinghouse plants (documented in a August 14, 1987 letter to Texas Utilities) included Radioactive Effluent Technical Specifications. Surveillance 4.11.1.2 for liquid effluents states, "Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM) at least once per 31 days." Surveillance 4.11.1.3.1 for the liquid radioactive waste treatment system states, "Doses due to liquid releases from each unit to unrestricted areas shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when liquid radioactive waste treatment systems are not being fully utilized."

Generic Letter 89-01 appears to have combined these two surveillance requirements for cumulative and projected doses. In combining these requirements in Generic Letter 89-01, the new program element can be interpreted to require determining projected dose contribution for the current calendar quarter and current calendar year every 31 days. This was not the NRC's intention. TSTF-308 clarified the requirement by stating, "Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days."

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
6-16

Turkey Point Bases Pages Affected
None

Licensing Precedent
Calvert Cliffs Units 1 and 2 requested adoption of TSTF-308, Revision 1, on June 11, 2002. The request was approved by the NRC on July 16, 2003 as Amendments 259 and 236.

Traveler Title: TSTF-361, Revision 2, "Allow Standby SDC/RHR/DHR Loop to be Inoperable to Support Testing"

NRC Approval: Letter from W. D. Beckner (NRC) to A. R. Pietrangelo (NEI) dated October 31, 2000.

Description of Proposed Turkey Point TS Change
The proposed change adds a note to the residual heat removal requirements during Mode 6 low water level operations (TS 3.9.8.2) which allows one required RHR loop to be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is operable and in operation.

Comparison of the Turkey Point TS Requirements to the ISTS and Differences Between the Proposed TS Changes and the Traveler
TSTF-361 revised ISTS 3.9.6, which requires two residual heat removal (RHR) loops to be operable and one loop to be in operation in Mode 6 when the water level less than 23 feet above the top of the reactor vessel flange. TSTF-361 added an LCO note which allows one of the required RHR loops to be inoperable for up to 2 hours for required surveillance testing provided the other loop is operable and in operation.

Turkey Point TS 3.9.8.2 requires two independent RHR loops to be operable and at least one loop to be in operation in Mode 6 when the water level above the top of the reactor vessel flange is less than 23 feet. A footnote is being added which allows one required RHR loop to be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is operable and in operation.

The ISTS requirements affected by this Traveler and the revised Turkey Point TS requirements are equivalent.

Justification
Turkey Point Specification 3.9.8.2 currently does not allow the non-operating RHR loop to be made inoperable to support surveillance testing. The allowance is needed to provide the flexibility to perform surveillance testing while ensuring that there is reasonable time for operators to respond to and mitigate any expected failures. Therefore, to support required outage activities and still maintain the plant in a safe condition, this footnote should be added to Specification 3.9.8.2.

As discussed above, the justification presented in the Traveler is applicable to Turkey Point. The Traveler is being adopted by Turkey Point with no significant changes.

Turkey Point TS Pages Affected
3/4 9-9

Turkey Point TS Bases Pages Affected
Page 110

Licensing Precedent

Calvert Cliffs Units 1 and 2 requested adoption of TSTF-361, Revision 2, on September 20, 2002. The request was approved by the NRC on February 25, 2003 as Amendments 256 and 233.

ENCLOSURE 2

NO SIGNIFICANT HAZARDS CONSIDERATIONS

Introduction

The proposed amendments revise the Turkey Point Technical Specifications to adopt eight generic changes that have been made to NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," (the ISTS) and that have been previously reviewed and approved by the Nuclear Regulatory Commission.

A summary listing of the proposed changes to the Technical Specifications follows.

- 2.1.1, Reactor Core, and 2.1.2, Reactor Coolant System Pressure
  References to Specification 6.7.1 are being deleted.

- 3/4.1.3, Movable Control Assemblies
  The time allowed to reduce the Power Range Neutron Flux – High trip setpoints is being revised from 4 hours following the required power reduction to 72 hours after entering the action.

- 3/4.2.1, Axial Flux Difference
  The time allowed to reduce the Power Range Neutron Flux – High trip setpoints is being revised from 4 hours following the required power reduction to 72 hours after entering the action.

- 3/4.2.2, Heat Flux Hot Channel Factor, $F_Q(Z)$
  The time allowed to reduce the Power Range Neutron Flux – High trip setpoints is being revised from 4 hours following the required power reduction to 72 hours after entering the action.

- 3/4.2.3, Nuclear Enthalpy Rise Hot Channel Factor
  The time allowed to reduce the Power Range Neutron Flux – High trip setpoints is being revised from 4 hours following the required power reduction to 72 hours after entering the action.

- 3/4.2.4, Quadrant Power Tilt Ratio
  The time allowed to reduce the Power Range Neutron Flux – High trip setpoints is being revised from 4 hours following the required power reduction to 72 hours after entering the action.

- 3/4.3.1, Reactor Trip System Instrumentation
  The time allowed to reduce the Power Range Neutron Flux – High trip setpoints is being revised from 4 hours following the required power reduction to 72 hours after entering the action.

- 3/4.4.3, Pressurizer
  The surveillance interval for the pressurizer heaters is being revised from 92 days to 18 months. Additionally, procedural details are being removed.

- 3/4.7.1.2, Auxiliary Feedwater System
  The surveillance requirement and surveillance interval for the auxiliary feedwater pumps are being changed. Additionally, procedural details are being removed.

- 3/4.9.8.2, Residual Heat Removal and Coolant Circulation, Low Water Level
  A footnote is being added to the Limiting Condition for Operation that would allow one required Residual Heat Removal (RHR) loop to be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is operable and in operation.

- 6.1, Responsibility, 6.2, Organization, 6.3, Facility Staff Qualifications, 6.14, Offsite Dose Calculation Manual (ODCM)
  Specific personnel titles are being replaced with generic titles.

- 6.2.2, Plant Staff
  Several requirements for licensed operators are being deleted.

- 6.2.3, Shift Technical Advisor
  This section is being revised to be consistent with ISTS Specification 5.2.2.f.

- 6.3.3
  This section is being added to indicate that a senior reactor operator and a licensed reactor operator perform the functions described in 10 CFR 50.54(m).

- 6.7, Safety Limit Violation
  This section is being deleted.

- 6.8.4.a, Primary Coolant Sources Outside Containment
  The surveillance interval is being revised to be consistent with the other Technical Specification surveillance intervals. The applicability of Specification 4.0.2 is being extended to this specification.

- 6.8.4.f, Radioactive Effluent Controls Program
  This section is being revised to be consistent with ISTS Specification 5.5.4.

- 6.8.5
  This section is being revised to be consistent with ISTS Specification 5.2.2.d, to allow a periodic independent review instead of a monthly review by the plant manager or his designee to ensure that excessive hours have not been assigned

- 6.9.1.5, Monthly Operating Reports
  The requirement to include documentation of all challenges to the power operated relief valves or safety valves is being deleted.

Determination of No Significant Hazards Consideration

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the 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 as follows:

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

The proposed changes revise administrative requirements, actions, action times, surveillance requirements and surveillance frequencies. The revised requirements are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased by the proposed changes. The Technical Specifications continue to require the systems, structures, and components associated with the revised requirements to be operable. Therefore, any mitigation functions assumed in the accident analyses will continue to be performed. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed amendments do not involve a significant increase in the probability or consequences of any 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 previously evaluated.

The proposed changes do not alter the design or physical configuration of the plant. No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed changes do not affect any other plant equipment. Therefore, operation of the facility in accordance with the proposed amendments does not create the possibility of a new or different kind of accident from any 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 changes do not change the design or function of plant equipment. The proposed changes do not significantly reduce the level of assurance that any associated plant equipment will be available to perform its function. The proposed changes provide operating flexibility without significantly affecting plant operation. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in the margin of safety.

Based on the above, we have determined that the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

ENCLOSURE 3

ENVIRONMENTAL CONSIDERATIONS

Environmental Consideration

The proposed license amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released off-site, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendments involve no significant hazards consideration, and therefore, meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Hence, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendments.

ENCLOSURE 4

PROPOSED MARK-UP OF

AFFECTED TECHNICAL SPECIFICATIONS

AND BASES PAGES

INDEX

ADMINISTRATIVE CONTROLS

SECTION	PAGE
6.1 RESPONSIBILITY	6-1
6.2 ORGANIZATION	6-1
6.2.1 ONSITE AND OFFSITE ORGANIZATION	6-1
6.2.2 PLANT STAFF	6-2
TABLE 6.2-1 MINIMUM SHIFT CREW COMPOSITION	6-4
6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION	6-5
6.3 FACILITY STAFF QUALIFICATIONS	6-5
6.4 DELETED	6-5
6.5 DELETED	6-5

INDEX

ADMINISTRATIVE CONTROLS

SECTION	PAGE
6.6 DELETED	6-12
6.7 SAFETY LIMIT VIOLATION DELETED	6-12
6.8 PROCEDURES AND PROGRAMS	6-13
6.9 REPORTING REQUIREMENTS	6-18
6.9.1 ROUTINE REPORTS	6-18
Startup Report	6-18
Annual Reports	6-19
Annual Radiological Environmental Operating Report	6-20
Annual Radioactive Effluent Release Report	6-20
Monthly Operating Report	6-20
Peaking Factor Limit Report	6-21
Core Operating Limits Report	6-21
6.9.2 SPECIAL REPORTS	6-22
6.10 DELETED	6-23

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1, for 3 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes and comply with the requirements of Specification 6.7.1.

(FOR INFORMATION ONLY)

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within the Allowed Rod Misalignment between the Analog Rod Position Indication ** and the group step counter demand position within one hour after rod motion. The Allowed Rod Misalignment shall be defined as:

a. for THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 18 steps, and

b. for THERMAL POWER greater than 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 12 steps.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps and THERMAL POWER greater than 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER and confirm that all indicated rod positions are within the Allowed Rod Misalignment, or

   3. Be in HOT STANDBY within the following 6 hours.

c. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 18 steps and THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Be in HOT STANDBY within the following 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.
**During Unit 4 Cycle 21, the position of Rod F-8 Shutdown Bank B will be determined every 8 hours by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed, until the repair of the indication system for this rod is completed.

d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the Allowed Rod Misalignment of Specification 3.1.3.1, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the Allowed Rod Misalignment of Specification 3.1.3.1, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

   a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next ~~4~~ hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.d.3.c and 3.1.3.1.d.3.d below are demonstrated, and 72

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

   c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

   d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position * of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time invervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. **

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 92 days.

* During Unit 4 Cycle 21, the position of Rod F-8 Shutdown Bank B will be determined every 8 hours by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed state, until the repair of the indication system for this rod is completed.

** During Unit 4 Cycle 21, the position of rod F-8, Shutdown Bank B, may be monitored by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed state and it will not provide an input into the Rod Position Deviation Monitor. The use of the alternate method for rod F-8 does not require the 4 hour comparison of demanded versus actual position per 4.1.3.1.1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed Relaxed Axial Offset Control (RAOC) operational space as defined in the CORE OPERATING LIMITS REPORT (COLR), or

b. within a +/- 2% or +/- 3% target band about the target flux difference during Base Load operation.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION***:

a. For RAOC operation with the indicated AFD outside of the limits specified in the COLR, either

   1. Restore the indicated AFD to within the RAOC limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoint to less than or equal to 55% of RATED THERMAL POWER within ~~the next 4 hours~~. 12 hours after exceeding the limits

b. For Base Load operation above P_T** with the indicated AFD outside of the applicable target band about the target flux difference, either

   1. Restore the indicated AFD to within the Peaking Factor Limit Report target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than P_T and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless indicated AFD is within the limits specified in the COLR.

* See Special Test Exceptions Specification 3.10.2.

** P_T = Reactor Power at which predicted F_Q would exceed its limit (consistent with Specification 4.2.2.1).

*** The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band.

(FOR INFORMATION ONLY)

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the alarm used to monitor the AFD is OPERABLE, and

   2) At least once per hour for the first 6 hours after restoring the alarm used to monitor the AFD to OPERABLE status.*

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the alarm used to monitor the AFD is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.3 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.2 above or by linear interpolation between the most recently measured value and the predicted value at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

* Performance of a functional test to demonstrate OPERABILITY of the alarm used to monitor the AFD may be substituted for this requirement.

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F^L_Q(Z)$ shall be limited by the following relationships:

$$F^M_Q(Z) \le \frac{[F_Q]^{Lx}}{P}[K(Z)] \text{ for } P > 0.5$$

$$F^M_Q(Z) \le \frac{[F_Q]^{Lx}}{0.5}[K(Z)] \text{ for } P \le 0.5$$

where: $[F_Q]^L$ = F_Q limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}},$$

$[F_Q]^M$ = The Measured Value, and

K(Z) for a given core height, is specified in the K(Z) curve, defined in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F^M_Q(Z)$ exceeding its limit:

72 hours after exceeding the limit.

a. Reduce THERMAL POWER at least 1% for each 1% $F^M_Q(Z)$ exceeds $F^L_Q(Z)$ within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within ~~the next 4 hours~~. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1%$F^M_Q(Z)$ exceeds the $F^L_Q(Z)$; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced power limit required by ACTION a., above; THERMAL POWER may then be increased provided $F^M_Q(Z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$F^N_{\Delta H} \leq F^{RTP}_{\Delta H}[1.0 + PF_{\Delta H}(1-P)]$,

Where: $F^{RTP}_{\Delta H}$ = $F_{\Delta}H$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$PF_{\Delta}H$ = Power Factor Multiplier for $F_{\Delta}H$ as specified in the CORE OPERATING LIMITS REPORT

$P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

72 hours after exceeding the limit

1. Restore $F^N_{\Delta H}$ to within the above limit, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and /or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within the limit of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% or RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the ~~next 4~~ hours.

72 hours after exceeding the limit

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within ~~the next 4~~ hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

 1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

  a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

  b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

 2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

 3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

72 hours after exceeding the limit

 4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

 1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

  a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

  b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

72 hours after exceeding the limit

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within ~~the next 4 hours~~, and

3. Identify and correct the cause of the out-of -limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

4.2.4.3 If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report in accordance with 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

(FOR INFORMATION ONLY)

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirement specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

(FOR INFORMATION ONLY)

TABLE 3.3-1 (Continued)

TABLE NOTATION

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

*** Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1(UF-4A1) or UF-3B1 (UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within ~~4~~ hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

72

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of indicated level, and at least two groups of pressurizer heaters each having a capacity of at least 125 kW and capable of being supplied by emergency power.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

18 months

to be at least 125 kW

** 14 days if the inoperability is associated with an inoperable diesel generator.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN Within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) ~~Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or~~

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

~~equal to 373 gpm to the entrance of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3;~~

~~2) Verifying by control panel indication and visual observation of equipment that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test above;~~

1) ~~3)~~ Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

2) ~~4)~~ Verifying that power is available to those components which require power for flow path operability.

b. At least once per 18 months by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each Auxiliary Feedwater Actuation test signal, and

Insert

2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each Auxiliary Feedwater Actuation test signal.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 1 by verifying normal flow to each steam generator.

INSERT to Page TS 3/4 7-4

c. By verifying the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head when tested in accordance with Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3.

(FOR INFORMATION ONLY)

TABLE 3.7-3

AUXILIARY FEEDWATER SYSTEM OPERABILITY

UNIT	TRAIN	STEAM SUPPLY FLOWPATH[3]	PUMP	DISCHARGE WATER FLOWPATH[3]
3	1	SG 3C via MOV-3-1405 or SG 3B via MOV-3-1404[1]	A or C[2]	SG 3A via CV-3-2816 SG 3B via CV-3-2817 SG 3C via CV-3-2818
3	2	SG 3A via MOV-3-1403 or SG 3B via MOV-3-1404[1]	B or C[2]	SG 3A via CV-3-2831 SG 3B via CV-3-2832 SG 3C via CV-3-2833
4	1	SG 4C via MOV-4-1405 or SG 4B via MOV-4-1404[1]	A or C[2]	SG 4A via CV-4-2816 SG 4B via CV-4-2817 SG 4C via CV-4-2818
4	2	SG 4A via MOV-4-1403 or SG 4B via MOV-4-1404[1]	B or C[2]	SG 4A via CV-4-2831 SG 4B via CV-4-2832 SG 4C via CV-4-2833

NOTES:

[1]Steam admissíon valves MOV-3-1404 and MOV-4-1404 can be aligned to either train (but not both) to restore OPERABILITY in the event MOV-3-1403 or MOV-3-1405, or MOV-4-1403 or MOV-4-1405 are inoperable.

[2]During single and two unit operation, one pump shall be OPERABLE in each train and the third auxiliary feedwater pump shall be OPERABLE and capable of being powered from, and supplying water to either train, except as noted in ACTION 3 of Technical Specification 3.7.1.2. The third auxiliary feedwater pump (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable.

[3]If any local manual realignment of valves is required when operating the auxiliary feedwater pumps, a dedicated individual, who is in communication with the control room, shall be stationed at the auxiliary feedwater pump area. Upon instructions from the control room, this operator would realign the valves in the AFW system train to its normal operational alignment.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation. (*)

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

Insert

INSERT to Page TS 3/4 9-9

* One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is OPERABLE and in operation.

PLANT STAFF

6.2.2 The plant organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

DELETED

b. ~~At least one licensed Operator shall be in the control room when fuel is in either reactor.~~

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

f. DELETED

h. The Operations Supervisor shall hold a Senior Reactor Operator License.

i. The Operations Manager shall either:

1. hold or have held a Senior Reactor Operator License on the Turkey Point Plant; or,

2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or

3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)

* The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

FUNCTION

6.2.3 SHIFT TECHNICAL ADVISOR

An individual

to the unit operations shift crew

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit and the opposite unit. ~~The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and~~ shall meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

This individual

delete

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for

6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.1.2 The Operations Manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

6.3.1.3 The licensed Operators and Senior Operators who shall also meet or exceed the minimum qualifications of the supplemental requirements specified in 10 CFR Part 55, and ANSI 3.1, 1981.

6.3.1.4 The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. *Education: Minimum of a high school diploma or equivalent*

b. *Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant*

c. Training: Complete the Multi-Discipline Supervisor training program

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.4 Deleted

6.5 Deleted

Insert 6.3.3

INSERT to Page TS 6-5

6.3.3 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1.3, perform the functions described in 10 CFR 50.54(m).

ADMINISTRATIVE CONTROLS

6.6 DELETED

~~6.7 SAFETY LIMIT VIOLATION~~

~~6.7.1 The following actions shall be taken in the event a Safety Limit is violated:~~

~~a. In accordance with 10 CFR 50.72, the NRC Operations Center, shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The Chief Nuclear Officer, and the Company Nuclear Review Board (CNRB) shall be notified within 24 hours.~~

~~b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.~~

~~c. The License Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the CNRB, and the Chief Nuclear Officer within 30 days after discovery of the event.~~

~~d. Critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory Commission.~~

6.7 DELETED

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at ~~refueling cycle intervals or less~~.

least every 18 months.

The provisions of Specification 4.0.2 are applicable.

b. DELETED

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

f. Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

2. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the 10 CFR 20, Appendix B, Table 2, Column 2 limits, concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

Insert 1

5. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;~~

6. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary ~~to 500 mrem per year to the whole body, 3000 mrem per year to the skin and 1500 mrem per year to any organ from Iodine 131, Iodine 133 tritium and all radionuclides in particulate form with half live greater than 8 days.~~

Insert 2

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY, conforming to 10 CFR §50, Appendix I;

INSERTS to Page TS 6-16

Insert 1

Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

Insert 2

shall be in accordance with the following:

a. For noble gases: a dose rate less than or equal to 500 mrems/yr to the whole body and a dose rate less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, iodine-133, tritium, and all radionuclide in particulate form with half-lives greater than 8 days: a dose rate less than or equal to 1500 mrems/yr to any organ.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

10. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

, beyond the site boundary,

g. Deleted Insert

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following deviations or exemptions:

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment interval pressure for the design basis loss of coolant accident, P_a, is 49.9 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding $1.0\ L_a$, the As-left leakage rate acceptance criterion is $\leq 0.75\ L_a$, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

INSERT to Page TS 6-17

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

- The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is $\leq 0.05\ L_a$, when pressurized to P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

l. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. Change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.8.4 l.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

personnel

6.8.5 Administrative procedures shall be developed and implemented to limit the working hours of ~~plant staff~~ who perform safety-related functions, e.g. licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel. The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized by the applicable department manager or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures ~~such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee~~ to ~~assure~~ that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized. ensure

to require a periodic independent review be conducted

ADMINISTRATIVE CONTROLS

ANNUAL REPORTS (Cont'd)

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the Offsite Dose Calculation Manual (ODCM), and in (2) 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT**

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience (including documentation of all challenges to the PORVs or safety valves) shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

*A single submittal may be made for a multiple unit station.

**A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

(FOR INFORMATION ONLY)

Procedure No.:	Procedure Title:	Page: 52
0-ADM-536	Technical Specification Bases Control Program	Approval Date: 5/1/03

ATTACHMENT 1
(Page 42 of 102)

TECHNICAL SPECIFICATION BASES

3/4.4 REACTOR COOLANT SYSTEM (Continued)

3/4.4.2 SAFETY VALVES (Continued)

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code. The pressurizer code safety valves' lift settings allows a +2%, -3% setpoint tolerance for OPERABILITY; however, the valves are reset to within ±1% during the surveillance to allow for drift.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the maximum water volume parameter is restored to within its limit following expected transient operation. The maximum water volume (1133 cubic feet) ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that both backup pressurizer heater groups be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

Insert

3/4.4.4 RELIEF VALVES

The opening of the power-operated relief valves (PORVs) fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2 or 3. Equipment necessary to establish PORV operability in Modes 1 and 2 is limited to Vital DC power and the Instrument Air system. Equipment necessary to establish block valve operability is limited to an AC power source. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a PORV fail in the open position.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used as a back-up for the steam generator tube rupture and to support plant shutdown in the event of an Appendix R fire. These functions are considered to be important-to-safety, or Quality Related per the FPL Quality Assurance program.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

(FOR INFORMATION ONLY)

INSERT to TS Bases Page 52

The Surveillance is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The frequency of 18 months is considered adequate to detect heater degradation and has been shown by industry operating experience to be acceptable.

(FOR INFORMATION ONLY)

Procedure No.:	Procedure Title:	Page:
0-ADM-536	Technical Specification Bases Control Program	83 Approval Date: 5/1/03

ATTACHMENT 1
(Page 73 of 102)

TECHNICAL SPECIFICATION BASES

3/4.7 PLANT SYSTEMS (Continued)

3/4.7.1.1 SAFETY VALVES (Continued)

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure (including tolerance and accumulation) - (Btu/lbm)

N = Number of loops in plant

The values calculated from this algorithm must then be adjusted lower for use in TS 3.7.1.1 to account for instrument and channel uncertainties.

Operation with less than all four MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. Table 3.7-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the surveillance to allow for drift.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety-related sources. Auxiliary feedwater can be supplied through redundant lines to the safety-related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. Each steam driven auxiliary feedwater pump has sufficient capacity for single and two unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

ACTION statement 2 describes the actions to be taken when both auxiliary feedwater trains are inoperable. The requirement to verify the availability of both standby feedwater pumps is to be accomplished by verifying that both pumps have successfully passed their monthly surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If no alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke ACTION statement 1 for the other train. If both standby feedwater pumps are made available before one auxiliary feedwater train is returned to an OPERABLE status, then the affected unit(s) shall be placed in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

(FOR INFORMATION ONLY)

Procedure No.:	Procedure Title:	Page:
0-ADM-536	Technical Specification Bases Control Program	84
		Approval Date: 5/1/03

ATTACHMENT 1
(Page 74 of 102)

TECHNICAL SPECIFICATION BASES

3/4.7 PLANT SYSTEMS (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

ACTION statement 3 describes the actions to be taken when a single auxiliary feedwater pump is inoperable. The requirement to verify that two independent auxiliary feedwater trains are OPERABLE is to be accomplished by verifying that the requirements for Table 3.7-3 have been successfully met for each train within the last surveillance interval. The provisions of Specification 3.0.4 are not applicable to the third auxiliary feedwater pump provided it has not been inoperable for longer than 30 days. This means that a unit(s) can change OPERATIONAL MODES during a unit(s) heatup with a single auxiliary feedwater pump inoperable as long as the requirements of ACTION statement 3 are satisfied.

The specified flow rate acceptance criteria conservatively bounds the limiting AFW flow rate modeled in the single unit loss of normal feedwater analysis. Dual unit events such as a two unit loss of offsite power require a higher pump flow rate, but it is not practical to test both units simultaneously. The monthly flow surveillance test specified in 4.7.1.2.1 is considered to be a general performance test for the AFW system and does not represent the limiting flow requirement for AFW. Check valves in the AFW system that require full stroke testing under limiting flow conditions are tested under Technical Specification 4.0.5. Insert C

The monthly testing of the auxiliary feedwater pumps will verify their operability. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps.

3/4.7.1.3 CONDENSATE STORAGE TANK

There are two (2) seismically designed 250,000 gallons condensate storage tanks. A minimum indicated volume of 210,000 gallons is maintained for each unit in MODES 1, 2 or 3. The OPERABILITY of the condensate storage tank with the minimum indicated volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately 23 hours or maintain the Reactor Coolant System at HOT STANDBY conditions for 15 hours and then cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed in operation.

The minimum indicated volume includes an allowance for instrument indication uncertainties and for water deemed unusable because of vortex formation and the configuration of the discharge line.

(FOR INFORMATION ONLY)

INSERT to TS Bases Page 84

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Specification 4.0.5, the Inservice Testing Program. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The surveillance states that Specification 4.0.4 is not applicable to entry into Modes 2 and 3 when testing the steam driven AFW pump. This allowance is required because there is insufficient steam pressure to perform the test prior to entering the applicability of the specification.

(FOR INFORMATION ONLY)

Procedure No.:	Procedure Title:	Page: 110
0-ADM-536	Technical Specification Bases Control Program	Approval Date: 5/1/03

ATTACHMENT 1
(Page 100 of 102)

TECHNICAL SPECIFICATION BASES

3/4.9 REFUELING OPERATIONS (Continued)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core. Insert

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment ventilation penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL – REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient shielding will be available during fuel movement and for removal of iodine in the event of a fuel handling accident. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 HANDLING OF SPENT FUEL CASK

Limiting spent fuel decay time from last time critical to a minimum of 1,525 hours prior to moving a spent fuel cask into the spent fuel pit will ensure that potential offsite doses are a fraction of 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

The restriction to allow only a single element cask to be moved into the spent fuel pit will ensure the maintenance of water inventory in the unlikely event of an uncontrolled cask descent. Use of a single element cask which nominally weighs about twenty-five tons will also increase crane safety margins by about a factor of four.

(FOR INFORMATION ONLY)

INSERT to TS Bases Page 110

Specification 3.9.8.2 is modified by a footnote that allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include the core time to boil, that there is no draining operation to further reduce RCS water level, and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

ENCLOSURE 5

RE-TYPED TECHNICAL SPECIFICATION PAGES

and

BASES PAGE
(for information only)

INDEX

ADMINISTRATIVE CONTROLS

SECTION	PAGE
6.1 RESPONSIBILITY	6-1
6.2 ORGANIZATION	6-1
6.2.1 ONSITE AND OFFSITE ORGANIZATION	6-1
6.2.2 PLANT STAFF	6-2
TABLE 6.2-1 MINIMUM SHIFT CREW COMPOSITION	6-4
6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION	6-5
6.3 FACILITY STAFF QUALIFICATIONS	6-5
6.4 DELETED	6-5
6.5 DELETED	6-5

ADMINISTRATIVE CONTROLS

SECTION PAGE

6.6 DELETED 6-12

6.7 DELETED 6-12

6.8 PROCEDURES AND PROGRAMS 6-13

6.9 REPORTING REQUIREMENTS 6-18

6.9.1 ROUTINE REPORTS 6-18

Startup Report 6-18
Annual Reports 6-19
Annual Radiological Environmental Operating Report 6-20
Annual Radioactive Effluent Release Report 6-20
Monthly Operating Report 6-20
Peaking Factor Limit Report 6-21
Core Operating Limits Report 6-21

6.9.2 SPECIAL REPORTS 6-22

6.10 DELETED 6-23

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1, for 3 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the Allowed Rod Misalignment of Specification 3.1.3.1, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the Allowed Rod Misalignment of Specification 3.1.3.1, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 72 hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.d.3.c and 3.1.3.1.d.3.d below are demonstrated, and

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and F_Q (Z) and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position * of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time invervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. **

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 92 days.

* During Unit 4 Cycle 20, the position of Rod C-9 Shutdown Bank A will be determined every 8 hours by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed state, until the repair of the indication system for this rod is completed.

** During Unit 4 Cycle 20, the position of rod C-9, Shutdown Bank A, may be monitored by verifying gripper coil parameters of the Control Rod Drive Mechanism to determine it has not changed state and it will not provide an input into the Rod Position Deviation Monitor. The use of the alternate method for rod C-9 does not require the 4 hour comparison of demanded versus actual position per 4.1.3.1.1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed Relaxed Axial Offset Control (RAOC) operational space as defined in the CORE OPERATING LIMITS REPORT (COLR), or

b. within a +/- 2% or +/- 3% target band about the target flux difference during Base Load operation.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION***:

a. For RAOC operation with the indicated AFD outside of the limits specified in the COLR, either

   1. Restore the indicated AFD to within the RAOC limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoint to less than or equal to 55% of RATED THERMAL POWER within 72 hours after exceeding the limits.

b. For Base Load operation above P_T** with the indicated AFD outside of the applicable target band about the target flux difference, either

   1. Restore the indicated AFD to within the Peaking Factor Limit Report target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than P_T and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless indicated AFD is within the limits specified in the COLR.

---

* See Special Test Exceptions Specification 3.10.2.

** P_T = Reactor Power at which predicted F_Q would exceed its limit (consistent with Specification 4.2.2.1).

*** The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q^{(Z)}$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q^L(Z)$ shall be limited by the following relationships:

$$F_Q^M(Z) \leq \frac{[F_Q]^{Lx}}{P}[K(Z)] \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{[F_Q]^{Lx}}{0.5}[K(Z)] \text{ for } P \leq 0.5$$

where: $[F_Q]^L$ = F_Q limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}},$$

$[F_Q]^M$ = The Measured Value, and

K(Z) for a given core height, is specified in the K(Z) curve, defined in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F_Q^M(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^M(Z)$ exceeds $F_Q^L(Z)$ within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within 72 hours after exceeding the limits: POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1% $F_Q^M(Z)$ exceeds the $F_Q^L(Z)$; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced power limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q^M(Z)$ is demonstrated through incore mapping to be within its limit.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H}\,[1.0 + PF_{\Delta H}\,(1-P)],$$

Where: $F^{RTP}_{\Delta H}$ = $F_{\Delta}H$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$PF_{\Delta}H$ = Power Factor Multiplier for $F_{\Delta}H$ as specified in the CORE OPERATING LIMITS REPORT

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

1. Restore $F^N_{\Delta H}$ to within the above limit, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within 72 hours after exceeding the limits.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and /or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within the limit of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% or RATED THERMAL POWER.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within 72 hours after exceeding the limits.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within 72 hours after exceeding the limits; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within 72 hours after exceeding the limits; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within 72 hours after exceeding the limits; and

3. Identify and correct the cause of the out-of -limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

4.2.4.3 If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report in accordance with 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

TABLE 3.3-1 (Continued)

TABLE NOTATION

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

*** Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1(UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 72 hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of indicated level, and at least two groups of pressurizer heaters each having a capacity of at least 125 kW and capable of being supplied by emergency power.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 125 kw at least once per 18 months.

** 14 days if the inoperability is associated with an inoperable diesel generator.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN Within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

2) Verifying that power is available to those components which require power for flow path operability.

b. At least once per 18 months by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each Auxiliary Feedwater Actuation test signal.

c. By verifying the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head when tested in accordance with Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 1 by verifying normal flow to each steam generator.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation*.

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

* One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is OPERABLE and in operation.

PLANT STAFF

6.2.2 The plant organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. DELETED

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

f. DELETED

h. The Operations Supervisor shall hold a Senior Reactor Operator License.

i. The Operations Manager shall either:

1. hold or have held a Senior Reactor Operator License on the Turkey Point Plant; or,

2. have held a Senior Reactor Operator License on a similar plant (i.e., another pressurized water reactor); or

3. have completed the Turkey Point Plant Senior Management Operations Training Course. (i.e., certified at an appropriate simulator for equivalent senior operator knowledge level.)

* The Health Physics Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

6.2.3 SHIFT TECHNICAL ADVISOR FUNCTION

6.2.3.1 An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit and the opposite unit. This individual shall meet the qualifications specified by the 1985 NRC Policy Statement on Engineering Expertise on Shift.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for

6.3.1.1 The Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.1.2 The operations manager whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i.

6.3.1.3 The licensed Operators and Senior Operators who shall also meet or exceed the minimum qualifications of the supplemental requirements specified in 10 CFR Part 55, and ANSI 3.1, 1981.

6.3.1.4 The Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant

c. Training: Complete the Multi-Discipline Supervisor training program

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.3.3 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1.3, perform the functions described in 10 CFR 50.54(m).

6.4 Deleted

6.5 Deleted

6.6 DELETED

6.7 DELETED

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at least every 18 months.

The provisions of Specification 4.0.2 are applicable.

b. DELETED

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

PROCEDURES AND PROGRAMS (Continued)

f. Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

2. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 – 20.2402;

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

5. Determination of cumulative dose from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

6. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

    a. For noble gases: a dose rate less than or equal to 500 mrems/yr to the whole body and a dose rate less than or equal to 3000 mrems/yr to the skin, and

    b. For iodine-131, iodine-133, tritium, and all radionuclide in particulate form with half-lives greater than 8 days: a dose rate less than or equal to 1500 mrems/yr to any organ.

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY, conforming to 10 CFR §50, Appendix I;

PROCEDURES AND PROGRAMS (Continued)

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Deleted

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following deviations or exemptions:

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment interval pressure for the design basis loss of coolant accident, P_a, is 49.9 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding $1.0\ L_a$, the As-left leakage rate acceptance criterion is $\leq 0.75\ L_a$, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than 0.60 L_a, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.

- The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than 0.6 L_a, at all times when containment integrity is required.

3) Overall air lock leakage acceptance criteria is $\leq$ 0.05 L_a, when pressurized to P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

i. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. Change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.8.4 i.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.8.5 Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety-related functions, e.g. licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel. The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized by the applicable department manager or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

ANNUAL REPORTS (Cont'd)

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the Offsite Dose Calculation Manual (ODCM), and in (2) 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT**

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

*A single submittal may be made for a multiple unit station.

**A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4 REACTOR COOLANT SYSTEM (Continued)

3/4.4.2 SAFETY VALVES (Continued)

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code. The pressurizer code safety valves' lift settings allows a +2%, -3% setpoint tolerance for OPERABILITY; however, the valves are reset to within ±1% during the surveillance to allow for drift.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the maximum water volume parameter is restored to within its limit following expected transient operation. The maximum water volume (1133 cubic feet) ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that both backup pressurizer heater groups be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation. The Surveillance is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The frequency of 18 months is considered adequate to detect heater degradation and has been shown by industry operating experience to be acceptable.

3/4.4.4 RELIEF VALVES

The opening of the power-operated relief valves (PORVs) fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2 or 3. Equipment necessary to establish PORV operability in Modes 1 and 2 is limited to Vital DC power and the Instrument Air system. Equipment necessary to establish block valve operability is limited to an AC power source. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a PORV fail in the open position.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used as a back-up for the steam generator tube rupture and to support plant shutdown in the event of an Appendix R fire. These functions are considered to be important-to-safety, or Quality Related per the FPL Quality Assurance program.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.7 PLANT SYSTEMS (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

ACTION statement 3 describes the actions to be taken when a single auxiliary feedwater pump is inoperable. The requirement to verify that two independent auxiliary feedwater trains are OPERABLE is to be accomplished by verifying that the requirements for Table 3.7-3 have been successfully met for each train within the last surveillance interval. The provisions of Specification 3.0.4 are not applicable to the third auxiliary feedwater pump provided it has not been inoperable for longer than 30 days. This means that a unit(s) can change OPERATIONAL MODES during a unit(s) heatup with a single auxiliary feedwater pump inoperable as long as the requirements of ACTION statement 3 are satisfied.

The flow surveillance test specified in 4.7.1.2.1.C is considered to be a general performance test for the AFW system and does not represent the limiting flow requirement for AFW. Check valves in the AFW system that require full stroke testing under limiting flow conditions are tested under Technical Specification 4.0.5.

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Specification 4.0.5, the Inservice Testing Program. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The surveillance states that Specification 4.0.4 is not applicable to entry into Modes 2 and 3 when testing the steam driven AFW pump. This allowance is required because there is insufficient steam pressure to perform the test prior to entering the applicability of the specification.

Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps.

3/4.7.1.3 CONDENSATE STORAGE TANK

There are two (2) seismically designed 250,000 gallons condensate storage tanks. A minimum indicated volume of 210,000 gallons is maintained for each unit in MODES 1, 2 or 3. The OPERABILITY of the condensate storage tank with the minimum indicated volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately 23 hours or maintain the Reactor Coolant System at HOT STANDBY conditions for 15 hours and then cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed in operation.

The minimum indicated volume includes an allowance for instrument indication uncertainties and for water deemed unusable because of vortex formation and the configuration of the discharge line.

ATTACHMENT 1
(Page 100 of 102)

TECHNICAL SPECIFICATION BASES

3/4.9 REFUELING OPERATIONS (Continued)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

Specification 3.9.8.2 is modified by a footnote that allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include the core time to boil, that there is no draining operation to further reduce RCS water level, and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment ventilation penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL – REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient shielding will be available during fuel movement and for removal of iodine in the event of a fuel handling accident. The minimum water depth is consistent with the assumptions of the safety analysis.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.9 REFUELING OPERATIONS (Continued)

3/4.9.12 HANDLING OF SPENT FUEL CASK

Limiting spent fuel decay time from last time critical to a minimum of 1,525 hours prior to moving a spent fuel cask into the spent fuel pit will ensure that potential offsite doses are a fraction of 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

The restriction to allow only a single element cask to be moved into the spent fuel pit will ensure the maintenance of water inventory in the unlikely event of an uncontrolled cask descent. Use of a single element cask which nominally weighs about twenty-five tons will also increase crane safety margins by about a factor of four.

Requiring that spent fuel decay time from last time critical be at least 120 days prior to moving a fuel assembly outside the fuel storage pit in a shipping cask will ensure that potential offsite doses are a fraction of 10 CFR 100 limits should a dropped cask and ruptured fuel assembly release activity directly to the atmosphere.

3/4.9.13 RADIATION MONITORING

The OPERABILITY of the containment radiation monitors ensures continuous monitoring of radiation levels to provide immediate indication of an unsafe condition.

3/4.9.14 SPENT FUEL STORAGE

The spent fuel storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure: a) $K_{eff} \leq 0.95$ with a minimum soluble boron concentration of 650 ppm present, and b) $K_{eff} < 1.0$ when flooded with unborated water for normal operations and postulated accidents.

The spent fuel racks are divided into two regions. Region I racks have a 10.6 inch center-to-center spacing and Region II racks have a 9.0 inch center-to-center spacing. Because of the larger center-to-center spacing and poison (B^{10}) concentration of Region I cells, the only restriction for placement of fuel is that the initial fuel assembly enrichment is equal to or less than 4.5 weight percent of U-235. The limiting value of U-235 enrichment is based upon the assumptions in the spent fuel safety analyses and assures that the limiting criteria for criticality is not exceeded. Prior to placement in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.9-1, placement in a Region II cell is authorized. These positive controls assure that fuel enrichment limits assumed in the safety analyses will not be exceeded.

ENCLOSURE 6

LIST OF REGULATORY COMMITMENTS

There are no regulatory commitments made by FPL in this submittal.