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SOUTHERN COMPANY

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant – Units 1 and 2
Application to Revise Technical Specifications to Adopt
Previously NRC-Approved Generic Technical Specification Changes

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is submitting a request for an amendment to the Technical Specifications (TS) for Vogtle Electric Generating Plant (VEGP) – Units 1 and 2.

The requested amendment will adopt various previously NRC-approved Technical Specifications Task Force (TSTF) Travelers. TSTF Travelers are generic changes to the Improved Standard Technical Specifications. These Travelers were chosen to increase the consistency between the Vogtle Technical Specifications and the Technical Specifications of the other plants in the SNC fleet. A list of the Travelers is located in Enclosure 1.

Enclosure 1 provides the basis for the proposed TS changes, and the Significant Hazards Consideration and Environmental Consideration determinations. Enclosure 2 provides the marked-up TS. Enclosure 3 contains example Bases changes that complement the proposed Technical Specification changes. The proposed Bases changes are provided for information only. The Bases will be revised under the Technical Specification Bases Control Program following NRC approval of the proposed Technical Specification changes. Enclosure 4 provides the clean-typed TS pages. Enclosure 5 provides a summary of the regulatory commitments made in this license amendment request.

As described in Enclosure 1, the proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards consideration.

SNC requests approval of the proposed license amendment by June 30, 2015, with the amendment being implemented within 90 days of issuance of the amendment.

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A copy of the proposed changes has been sent to J. H. Turner, the Georgia State Designee, in accordance with 10 CFR 50.91(b)(1).

Mr. C. R. Pierce states he is Regulatory Affairs Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

If you have any questions, please contact Ken McElroy at (205) 992-7369.

Respectfully submitted,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/EGA

Sworn to and subscribed before me this 18 *day of* July, *2014.*

Laura L. Crump
Notary Public

My commission expires: 10/8/2017

Enclosures:
1. Basis for Proposed Changes
2. Marked-Up Technical Specifications Pages
3. Example Marked-Up Technical Specifications Bases Pages
4. Clean-Typed Technical Specification Pages
5. Summary of Regulatory Commitments

cc: Southern Nuclear Operating Company
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Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
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Mr. V. M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Senior Project Manager – Vogtle
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State of Georgia
Mr. J. H. Turner, Environmental Director Protection Division

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Adoption of Previously NRC-Approved Generic Technical Specification Changes

Enclosure 1

Basis for Proposed Changes

Basis for Proposed Changes

1.0 Description

The requested amendment will adopt various previously NRC-approved Technical Specifications Task Force (TSTF) Travelers. TSTF Travelers are generic changes chosen to increase the consistency between the Vogtle Technical Specifications, the Improved Standard Technical Specifications (ISTS) for Westinghouse plants (NUREG-1431), and the Technical Specifications of the other plants in the SNC fleet. The requested Travelers are:

1. TSTF-2-A, Revision 1, "Relocate the 10 Year Sediment Cleaning of the Fuel Oil Storage Tank to Licensee Control" (Page E1-4)

2. TSTF-27-A, Revision 3, "Revise SR Frequency for Minimum Temperature for Criticality" (Page E1-7)

3. TSTF-28-A, Revision 0, "Delete Unnecessary Action to Measure Gross Specific Activity" (Page E1-11)

4. TSTF-45-A, Revision 2, "Exempt Verification of CIVs that are Locked, Sealed or Otherwise Secured" (Page E1-14)

5. TSTF-46-A, Revision 1, "Clarify the CIV Surveillance to Apply Only to Automatic Isolation Valves" (Page E1-17)

6. TSTF-87-A, Revision 2, "Revise 'RTBs Open' and 'CRDM De-energized' Actions to 'Incapable of Rod Withdrawal'" (Page E1-21)

7. TSTF-95-A, Revision 0, "Revise Completion Time for Reducing Power Range High Trip Setpoint from 8 to 72 Hours" (Page E1-25)

8. TSTF-110-A, Revision 2, "Delete SR Frequencies Based on Inoperable Alarms" (Page E1-28)

9. TSTF-142-A, Revision 0, "Increase the Completion Time When the Core Reactivity Balance is Not Within Limit" (Page E1-32)

10. TSTF-234-A, Revision 1, "Add Action for More Than One [D]RPI Inoperable" (Page E1-35)

11. TSTF-245, Revision 1, "AFW Train Inoperable When in Service" (Page E1-38)

12. TSTF-247-A, Revision 0, "Provide Separate Condition Entry for Each PORV and Block Valve" (Page E1-42)

13. TSTF-248-A, Revision 0, "Revise Shutdown Margin Definition for Stuck Rod Exception" (Page E1-46)

14. TSTF-266-A, Revision 3, "Eliminate the Remote Shutdown System Table of Instrumentation and Controls" (Page E1-49)

15. TSTF-272-A, Revision 1, "Refueling Boron Concentration Clarification" (Page E1-52)

16. TSTF-273-A, Revision 2, "Safety Function Determination Program Clarifications" (Page E1-55)

17. TSTF-284-A, Revision 3, "Add 'Met vs. Perform' to Technical Specification 1.4, Frequency" (Page E1-58)

18. TSTF-308-A, Revision 1, "Determination of Cumulative and Projected Dose Contributions in RECP" (Page E1-62)

19. TSTF-312-A, Revision 1, "Administrative Control of Containment Penetrations" (Page E1-65)

20. TSTF-314-A, Revision 0, "Require Static and Transient F_Q Measurement" (Page E1-70)

21. TSTF-340-A, Revision 3, "Allow 7-Day Completion Time for a Turbine-Driven AFW Pump Inoperable" (Page E1-73)

22. TSTF-343-A, Revision 1, "Containment Structural Integrity" (Page E1-76)

23. TSTF-349-A, Revision 1, "Add Note to LCO 3.9.5 Allowing Shutdown Cooling Loops Removal from Operation" (Page E1-81)

2.0 Proposed Changes, Justifications, and No Significant Hazards Determinations

Each Traveler is discussed in an individual analysis provided in Section 2.1 through 2.23. Each section contains the following topics:

Description of Proposed Change - This topic describes the effect of adopting the subject Traveler on the Vogtle Technical Specifications.

Differences Between the Proposed Change and the Approved Traveler - This topic describes differences between the changes proposed to the Vogtle Technical Specifications and the ISTS mark-ups provided in the approved Traveler.

Summary of the Approved Traveler Justification - This topic summarizes the justification utilized by the NRC when approving the Traveler.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification - This topic describes any differences between the Traveler justification utilized by the NRC when approving the Traveler and

the justification for adopting the Traveler in the Vogtle Technical Specifications.

License Commitments Required to Adopt this Change - Some Travelers require that licensees make regulatory commitments as a condition of adopting the change. This topic describes any such commitments being made by SNC as part of this request.

NRC Approval - This topic references the NRC letter, if any, approving the Traveler. It also provides example NRC approvals of plant-specific requests to adopt the Traveler. If the documents are in the NRC ADAMS system, the accession number (ACN) is given.

List of Affected Pages - This topic lists the Vogtle Technical Specification and Technical Specification Bases pages affected by the adoption of this Traveler.

Applicable Regulatory Requirements/Criteria - This topic describes how the justification satisfies the applicable regulatory requirements and criteria and provides a basis that the NRC staff may use to find the proposed amendment acceptable.

Significant Hazards Consideration - This topic provides an evaluation of whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment."

The affected marked-up Technical Specifications pages are in Enclosure 2. Retyped Technical Specification pages are in Enclosure 4.

Example mark-ups of the affected Technical Specification Bases pages are included for information only in Enclosure 3. The Bases will be revised under the Technical Specification Bases Control Program following NRC approval of the proposed Technical Specification changes.

To facilitate NRC review, each Traveler analysis will begin on a new page.

2.1 TSTF-2-A, Revision 1, "Relocate the 10 Year Sediment Cleaning of the Fuel Oil Storage Tank to Licensee Control"

Description of Proposed Change
The proposed change modifies Specification 3.8.3, "Diesel Fuel Oil, Lube Oil, Starting Air, and Ventilation," by removing Surveillance Requirement (SR) 3.8.3.7, which requires sediment cleaning of the fuel oil storage tanks every 10 years, from the Technical Specification and placing it under licensee control.

Differences Between the Proposed Change and the Approved Traveler
TSTF-2-A removes ISTS SR 3.8.3.6 from the Technical Specification. The equivalent SR in the Vogtle Technical Specifications is numbered SR 3.8.3.7.

Summary of the Approved Traveler Justification
The Technical Specifications are modified by removal of the SR directing performance of the 10 year diesel fuel oil storage tank cleaning that is specified in Regulatory Guide 1.137 to a document that is controlled by the licensee under 10 CFR 50.59. Fuel oil storage tank cleaning is a maintenance activity and is not a necessary surveillance to demonstrate operability of the diesel generators. As such, the SR does not meet the 10 CFR 50.36 description of a Surveillance Requirement and can be removed from the Technical Specifications and placed under licensee control.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
Administrative methods will be established to control performance of the 10 year diesel fuel oil storage tank cleaning activities that are currently described in SR 3.8.3.7.

NRC Approval
TSTF-2-A, Revision 1, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated July 16, 1998 (ACN ML9807280010). An example of a plant-specific NRC approval of the changes in TSTF-2-A is Catawba amendment number 206/200 dated July 10, 2003 (ACN ML031910598).

List of Affected Pages
3.8.3-3
3.8.3-4
B3.8.3-13
B3.8.3-14

Applicable Regulatory Requirements/Criteria
Under 10 CFR 50.36(c)(2)(ii), a limiting condition for operation must be included in Technical Specifications for any item meeting one or more of the four included criteria. As a result, existing Technical Specifications requirements that fall within or satisfy any of the criteria in 10 CFR 50.36 must be retained in the Technical Specifications, while those Technical Specifications requirements that do not fall

within or satisfy these criteria may be removed from the Technical Specifications and placed in other licensee controlled documents.

SR 3.8.3.7 is a maintenance activity, and is not a necessary surveillance to demonstrate operability of the diesel generators, and thus does not meet the criteria in 10 CFR 50.36 for retention in the Technical Specifications.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change removes the Surveillance Requirement for performing sediment cleaning of diesel fuel oil storage tanks every 10 years from the Technical Specifications and places it under licensee control. Diesel fuel oil storage tank cleaning is not an initiator of any accident previously evaluated. This change will have no effect on diesel generator fuel oil quality, which is tested in accordance with other Technical Specifications requirements. Removing the diesel fuel oil storage tank sediment cleaning requirements from the Technical Specifications will have no effect on the ability to mitigate an accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change removes the requirement to clean sediment from the diesel fuel oil storage tank from the Technical Specifications and places it under licensee control. The margin of safety provided by the fuel oil storage tank sediment cleaning is unaffected by this relocation because the quality of diesel fuel oil is tested in accordance with other Technical Specifications requirements. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.2 TSTF-27-A, Revision 3, "Revise SR Frequency for Minimum Temperature for Criticality"

Description of Proposed Change

The proposed change revises Specification 3.4.2, "RCS Minimum Temperature for Criticality," to modify the Frequency of SR 3.4.2.1. The Frequency is changed from "Once within 30 minutes and every 30 minutes thereafter when the T_{avg} - T_{ref} deviation alarm is not reset and any RCS loop T_{avg} < 561°F," to state, "In accordance with the Surveillance Frequency Control Program."

Differences Between the Proposed Change and the Approved Traveler

The frequency for ISTS SR 3.4.2.1, and its associated Note, are modified by TSTF-27-A. The changes in TSTF-27-A would modify the Frequency for SR 3.4.2.1 to a periodic frequency of 12 hours. As described in TS 5.5.21, Vogtle has adopted a Surveillance Frequency Control Program (SFCP) to control surveillances with periodic frequencies. The Frequency for SR 3.4.2.1, as modified by the changes identified in TSTF-27-A, will become a periodic frequency, and can be controlled under the SFCP. The Frequency for SR 3.4.2.1 is therefore modified to indicate that it is "In accordance with the Surveillance Frequency Control Program. The initial Frequency for this Surveillance will be 12 hours. The changes to SR 3.4.2.1 and the Bases for this SR are modified from that in TSTF-27-A to reflect this difference. NRC approval of the license change implementing the SFCP was provided in Amendment Numbers 158/140, dated January 19, 2011 (ACN ML102520083).

Summary of the Approved Traveler Justification

Specification 3.4.2, "RCS Minimum Temperature for Criticality," is designed to prevent criticality outside of the normal operating regime. There are no safety analyses that dictate the minimum temperature for criticality, but most low power accident analyses assume a specific starting temperature.

During the approach to criticality, reactor coolant system (RCS) temperature is closely watched. There are indications in the control room of deviations between actual and reference RCS temperature and on low RCS temperature to alert the operator if temperature is deviating from the program value. The Frequency of the SR only specifies how often temperature is logged, not how often it is watched. Therefore, the issue isn't whether or not the safety analysis assumptions are being protected, but how often RCS temperature is recorded in an operator's log. Therefore, this Traveler affects presentation and logging, not safety.

The current presentation can lead to inadvertently violating the SR Frequency with no effect on safety. The 30 minute SR Frequency "clock" continues even when RCS temperature is above the SR threshold or Applicability threshold temperature. Therefore, if temperature drops below the threshold value after more than 37 minutes (30 minutes + 25%) from the last time RCS temperature was logged, the SR Frequency has been violated. If temperature has unexpectedly decreased, the operator's attention should be on restoring temperature, not logging a value to meet a Surveillance. The operator is faced with making a decision of whether to focus his attention on the plant or on an administrative requirement. This is clearly adverse to safety. The other option is

to perform the surveillance every 30 minutes until temperature is well above the threshold value in order to ensure that the SR has been performed if temperature should drop. This is not a beneficial use of an operator's time during the critical phases of a startup.

The proposed Frequency for SR 3.4.2.1 is modified to indicate that it is "In accordance with the Surveillance Frequency Control Program. The initial Frequency for this Surveillance will be 12 hours. This will ensure that T_{avg} is logged at appropriate intervals (in addition to strip chart recorders and computer logging of temperature).

The requirement that RCS temperature must be above a certain value when the reactor is critical is stated in the LCO. This requirement will be monitored based on operating necessity whether or not it is specified in a Surveillance Requirement. Requiring that the value be logged based on conditional circumstances is poor human-factors design and diverts the operator's attention from his duties without a compensating safety benefit.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
The NRC did not issue a letter approving TSTF-27-A, Revision 3, however, it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-27-A, Revision 3 has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265).

List of Affected Pages
3.4.2-1
B3.4.2-3

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 28, Reactivity Limits, states:

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

There is no regulatory requirement that specifies the interval between measurement and logging of reactivity parameters, such as RCS temperature. The proposed Frequency for SR 3.4.2.1 is modified to indicate that it is "In accordance with the Surveillance Frequency Control Program." The initial Frequency for this Surveillance will be 12 hours. This will ensure that T_{avg} is logged at appropriate intervals (in addition to strip chart recorders and computer logging of temperature).

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the Surveillance Frequency for monitoring RCS temperature to ensure the minimum temperature for criticality is met. The Frequency is changed from a 30 minute Frequency when certain conditions are met to a periodic Frequency that it is controlled in accordance with the Surveillance Frequency Control Program. The initial Frequency for this Surveillance will be 12 hours. This will ensure that T_{avg} is logged at appropriate intervals (in addition to strip chart recorders and computer logging of temperature). The measurement of RCS temperature is not an initiator of any accident previously evaluated. The minimum RCS temperature for criticality is not changed. As a result, the mitigation of any accident previously evaluated is not affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change revises the Surveillance Frequency for monitoring RCS temperature to ensure the minimum temperature for criticality is met. The current, condition based Frequency represents a distraction to the control room operator during the critical period of plant startup. RCS temperature is closely monitored by the operator during the approach to criticality, and temperature is recorded on charts and computer logs. Allowing the operator to monitor temperature as needed by the situation and logging RCS temperature at a periodic Frequency that it is controlled in accordance with the Surveillance Frequency Control Program is sufficient to ensure that the LCO is met while eliminating a diversion of the operator's attention. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.3 TSTF-28-A, Revision 0, "Delete Unnecessary Action to Measure Gross Specific Activity"

Description of Proposed Change
The proposed change deletes Required Action B.1 of Specification 3.4.16, "RCS Specific Activity." Required Action B.1 requires performance of SR 3.4.16.2, which verifies that reactor coolant Dose Equivalent I-131 specific activity is $\leq$ 1.0 µCi/gm.

Differences Between the Proposed Change and the Approved Traveler
None.

Summary of the Approved Traveler Justification
Required Action B.1 requires performance of SR 3.4.16.2, which requires measurement of the Dose Equivalent I-131 specific activity. Measurement of Dose Equivalent I-131 specific activity must be performed in order to verify "restoration" of the specific activity to within limits and does not need to be otherwise required. Further, if the Condition is entered and the plant is in Mode 2 in 4 hours or less, the Required Action is in conflict with the Note of SR 3.4.16.2 which states that the SR is only required to be performed in Mode 1. Finally, this Required Action is an unnecessary burden on the operator because Required Action B.2 requires the plant to be in Mode 3 with Reactor Coolant System T_{avg} < 500°F within 6 hours. Required Action B.2 requires the plant to exit the Applicability of the Specification and should be the focus of the operator's attention.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
TSTF-28-A, Revision 0, was approved by the NRC as documented in a letter from Christopher Grimes (NRC) to James Davis (NEI), dated September 27, 1996 (ACN ML9610030183). TSTF-28-A has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265).

List of Affected Pages
3.4.16-1
B3.4.16-4

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 64, Monitoring Radioactivity Releases, states:

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

There is no regulatory requirement that specifies when potential radioactive effluents, such as Dose Equivalent I-131 specific activity, should be measured. The ISTS for Westinghouse Plants (NUREG-1431), does not require periodic Dose Equivalent I-131 specific activity measurement during a plant shutdown for Dose Equivalent I-131 specific activity not within limit. The proposed Required Actions are consistent with the NUREG-1431 Required Actions.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change eliminates Required Action B.1 of Specification 3.4.16, "RCS Specific Activity," which requires verifying that Dose Equivalent I-131 specific activity is within limits. Determination of Dose Equivalent I-131 is not an initiator of any accident previously evaluated. Determination of Dose Equivalent I-131 has no effect on the mitigation of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change eliminates a Required Action. The activities performed under the Required Action will still be performed to determine if the LCO is met or the plant will exit the Applicability of the Specification. In either case, the presence of the Required Action does not provide any significant margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.4 TSTF-45-A, Revision 2, "Exempt Verification of CIVs that are Locked, Sealed or Otherwise Secured"

Description of Proposed Change
The proposed change revises SR 3.6.3.3 and SR 3.6.3.4 in Specification 3.6.3, "Containment Isolation Valves," to exempt containment isolation valves (CIVs) from position verification if the valves are locked, sealed, or otherwise secured in position.

Differences Between the Proposed Change and the Approved Traveler
None.

Summary of the Approved Traveler Justification
The proposed change revises SR 3.6.3.3 and SR 3.6.3.4 in Specification 3.6.3, "Containment Isolation Valves," for containment isolation manual valves and blind flanges located inside and outside containment, by adding a provision to exempt from the position verification requirements CIVs that are locked, sealed, or otherwise secured in position. Because the SRs are intended to ensure the position of valves that could be inadvertently repositioned, it is not necessary to check the CIVs that are locked, sealed, or otherwise secured, because these valves were verified to be in the correct position upon being locked, sealed, or otherwise secured.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

In addition, the proposed change is consistent with other Vogtle Surveillance Requirements to verify the position of valves, such as SR 3.5.2.2 (Emergency Core Cooling System valves), SR 3.7.5.1 (Auxiliary Feedwater System valves), SR 3.6.6.1 (Containment Spray and Cooling System valves), SR 3.7.7.1 (Component Cooling Water System valves), SR 3.7.8.1 (Nuclear Service Cooling Water System valves), and SR 3.7.14.1 (Engineered Safety Features ESF Room Cooler and Safety-Related Chiller System valves).

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
TSTF-45-A, Revision 2, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated July 26, 1999 (ACN ML9907300113). An example of a plant-specific NRC approval of the changes in TSTF-45-A is San Onofre Units 2 and 3, Amendment Numbers 201/192 dated November 3, 2005 (ACN ML052780467).

List of Affected Pages
3.6.3-4
3.6.3-5
B3.6.3-10
B3.6.3-11

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criteria:

Criterion 16, Containment Design, states:

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 53, Provisions for Containment Testing and Inspection, states:

> The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important area, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.

In accordance with the requirement of Criterion 16 for an essentially leak-tight containment barrier, open containment isolation valves are designed to either close automatically when required or are periodically inspected to ensure they are closed. However, it is not necessary to periodically verify that containment isolation valves are closed to meet Criterion 16 if those valves are locked, sealed, or otherwise secured in the closed position.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change exempts containment isolation valves (CIVs) located inside and outside of containment that are locked, sealed, or otherwise secured in position from the periodic verification of valve position required by Surveillance Requirements 3.6.3.3 and 3.6.2.4. The exempted valves are verified to be in the correct position upon being locked, sealed, or secured. Because the valves are in the condition assumed in the accident analysis, the proposed change will not affect the initiators or mitigation of any accident previously evaluated. Therefore,

the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change replaces the periodic verification of valve position with verification of valve position followed by locking, sealing, or otherwise securing the valve in position. Periodic verification is also effective in detecting valve mispositioning. However, verification followed by securing the valve in position is effective in preventing valve mispositioning. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.5 TSTF-46-A, Revision 1, "Clarify the CIV Surveillance to Apply Only to Automatic Isolation Valves"

Description of Proposed Change
The proposed change modifies SR 3.6.3.5, and its associated Bases, to delete the requirement to verify the isolation time of "each power operated" containment isolation valve and only require verification of each "automatic power operated isolation valve."

Differences Between the Proposed Change and the Approved Traveler
None.

Summary of the Approved Traveler Justification
SR 3.6.3.5 requires verification that the isolation time of "each power operated and each automatic containment isolation valve is within limits." The Bases for this SR state that the "isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis." However, there are some valves credited as containment isolation valves that are power operated (i.e., can be remotely operated) that do not receive a containment isolation signal (e.g., a GDC 57 penetration). These power operated valves do not have an isolation time that is assumed in the accident analyses since they require operator action. The revised SR will clarify that it is only containment isolation valves (CIVs) that receive an automatic isolation signal that are in the scope of the SR. The associated Technical Specification Bases are also revised to reflect these changes. Deleting the reference to "power operated" isolation valve time testing reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analysis.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
The NRC did not issue a letter approving TSTF-46-A, Revision 1; however, it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-46-A, Revision 1 has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265). An example of a plant-specific NRC approval of the changes in TSTF-46-A is Peach Bottom Atomic Power Station, Units 2 and 3, Amendment Numbers 259/262 May 10, 2006 (ACN ML061070292).

List of Affected Pages
3.6.3-5
B3.6.3-11

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 16, Containment Design, states:

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

In accordance with the requirement of Criterion 16 for an essentially leak-tight containment barrier, CIVs are designed to either close automatically when required or are capable of being closed manually if the valve required to be operated. It is not necessary to verify closure times for CIVs that do not receive an automatic isolation signal, and for which no closure time is assumed ion the accident analysis.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, 10 CFR 50.36(c)(2)(ii)(C), states:

> *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The change affects power operated CIVs that do not receive a containment isolation signal, and that do not have an isolation time that is assumed in the accident analyses, since they require operator action. There is no regulatory requirement to establish or verify isolation times for CIVs that are not credited to automatically close in the accident analysis. The changes will not alter the CIV design, or the design of the isolation logic or circuitry. The CIVs will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

The proposed change revises the requirements in Technical Specification SR 3.6.3.5, and the associated Bases, to delete the requirement to verify the isolation time of "each power operated" containment isolation valve (CIV) and only require verification of closure time for each "automatic power operated isolation valve." The closure times for CIVs that do not receive an automatic closure signal are not an initiator of any design basis accident or event, and therefore the proposed change does not increase the probability of any accident previously evaluated. The CIVs are used to respond to accidents previously evaluated. Power operated CIVs that do not receive an automatic closure signal are not assumed to close in a specified time. The proposed change does not change how the plant would mitigate an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not result in a change in the manner in which the CIVs provide plant protection or introduce any new or different operational conditions. Periodic verification that the closure times for CIVs that receive an automatic closure signal are within the limits established by the accident analysis will continue to be performed under SR 3.6.3.5. The change does not alter assumptions made in the safety analysis, and is consistent with the safety analysis assumptions and current plant operating practice. There are also no design changes associated with the proposed changes, and the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides clarification that only CIVs that receive an automatic isolation signal are within the scope of the SR 3.6.3.5. The proposed change does not result in a change in the manner in which the CIVs provide plant protection. Periodic verification that closure times for CIVs that receive an automatic isolation signal are within the limits established by the accident analysis will continue to be performed. The proposed change does not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The

proposed change will not result in plant operation in a configuration outside the design basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.6 TSTF-87-A, Revision 2, "Revise 'RTBs Open' and 'CRDM De-energized' Actions to 'Incapable of Rod Withdrawal'"

Description of Proposed Change
The proposed change modifies Specification 3.4.5, "RCS Loops – Mode 3," Required Action C.2 and D.1, from "De-energize all control rod drive mechanisms" to "Place the Rod Control System in a condition incapable of rod withdrawal." It also modifies Specification 3.4.9, "Pressurizer." Required Action A.1, from requiring reactor trip breakers (RTBs) to be open after reaching MODE 3 to "Place the Rod Control System in a condition incapable of rod withdrawal," and to require full insertion of all rods.

Differences Between the Proposed Change and the Approved Traveler
None.

Summary of the Approved Traveler Justification
This change provides for a consistent presentation of the Required Actions. The specific method for ensuring that rods cannot be withdrawn is removed from the Technical Specifications. Since the revised Actions still assure rod withdrawal is precluded, this detail is not required to be in the Technical Specifications to provide adequate protection of the public health and safety. There is no overall effect from the change. The requirement that the control rods are inserted and are not capable of being withdrawn is maintained. Therefore, removing this detail from the Technical Specifications is acceptable.

This change (allowing alternate options to preclude rod withdrawal) is necessary to eliminate undesirable secondary effects of opening the RTBs. By opening the RTBs, plant interlock P-4 is tripped, which results in a trip of the main turbine and will close the main and bypass feedwater lines if RCS T_{avg} is below the low setpoint in MODE 3. Forcing reliance on AFW in this condition is not the intent, nor is it desirable, over continued use of normal Feedwater. Additionally, Condition C of LCO 3.4.5 and LCO 3.9.1 are modified to reflect the LCO. The status of the reactor trip breakers is not a requirement of the LCO; and is therefore inappropriate in the Condition. No technical changes result from this change.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
The NRC did not issue a letter approving TSTF-87-A, Revision 2; however, it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-87-A, Revision 2 has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265).

List of Affected Pages
3.4.5-2

3.4.9-1
B3.4.5-1
B3.4.5-2
B3.4.5-3
B3.4.5-4
B3.4.5-5
B3.4.9-3

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 28, Reactivity Limits, states:

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

There is no regulatory requirement that specifies the manner by which actions taken to prevent inadvertent rod withdrawal must be performed. As a result, it is not necessary to restrict the methods used to perform this function in order to meet Criterion 28.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, 10 CFR 50.36(c)(2)(ii)(B), states:

> *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> TS 3.4.9, "Pressurizer," satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, 10 CFR 50.36(c)(2)(ii)(C), states:

> *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The intent of Required Actions in LCO 3.4.5 and LCO 3.4.9 directing, "De-energize all control rod drive mechanisms" is to prevent introduction of positive reactivity by inadvertent rod withdrawal. Changing this direction to state, "Place the Rod Control System in a condition incapable of rod withdrawal" satisfies the

same intent in a less specific manner. There will be no changes to the design of the Reactor Coolant System or Rod Control System such that compliance with any of the regulatory requirements and guidance documents above would come into question. The Reactor Coolant System and Rod Control System will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   This change revises the Required Actions for LCO 3.4.5, "RCS Loops – Mode 3," Conditions C.2 and D.1, from "De-energize all control rod drive mechanisms," to "Place the Rod Control System in a condition incapable of rod withdrawal." It also revises LCO 3.4.9, "Pressurizer," Required Action A.1, from requiring Reactor Trip Breakers to be open after reaching MODE 3 to "Place the Rod Control System in a condition incapable of rod withdrawal," and to require full insertion of all rods. Inadvertent rod withdrawal can be an initiator for design basis accidents or events during certain plant conditions, and therefore must be prevented under those conditions. The proposed Required Actions for LCO 3.4.5 and LCO 3.4.9 satisfy the same intent as the current Required Actions, which is to prevent inadvertent rod withdrawal when an applicable Condition is not met, and is consistent with the assumptions of the accident analysis. As a result, the proposed change does not increase the probability of any accident previously evaluated. The proposed change does not change how the plant would mitigate an accident previously evaluated, as in both the current and proposed requirements, rod withdrawal is prohibited. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change provides less specific, but equivalent, direction on the manner in which inadvertent control rod withdrawal is to be prevented when the Conditions of LCO 3.4.5 and LCO 3.4.9 are not met. Rod

withdrawal will continue to be prevented when the applicable Conditions of LCO 3.4.5 and LCO 3.4.9 are met. There are no design changes associated with the proposed changes, and the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The change does not alter assumptions made in the safety analysis, and is consistent with the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides the operational flexibility of allowing alternate, but equivalent, methods of preventing rod withdrawal when the applicable Conditions of LCO 3.4.5 and LCO 3.4.9 are met. The proposed change does not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change will not result in plant operation in a configuration outside the design basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.7 TSTF-95-A, Revision 0, "Revise Completion Time for Reducing Power Range High trip Setpoint from 8 to 72 Hours"

Description of Proposed Change
The proposed change revises Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," Required Action A.2, and Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," Required Action A.1.2.2 to provide 72 hours Completion Time instead of 8 hours to reset the Power Range Neutron Flux - High trip setpoints to a lower value.

Differences Between the Proposed Change and the Approved Traveler
The ISTS contains several alternative specifications on $F_Q(Z)$ to reflect different methodologies. TSTF-95-A revised Specification 3.2.1B, "$F_Q(Z)$ (F_Q Methodology)," which is the equivalent Vogtle Specification is Specification 3.2.1.

Summary of the Approved Traveler Justification
The existing Completion Time of 8 hours to reduce the Power Range Neutron Flux-High trip setpoints presents an unjustified burden on the operation of the plant. A Completion Time of 72 hours 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, during which there is increased potential for a plant transient and human error. Following a significant power reduction, at least 24 hours are required to re-establish steady state xenon prior to taking a flux map, and approximately 8 to 12 hours to obtain a flux map, and analyze the data. A significant potential for human error can be created by requiring the trip setpoints to be reduced within the same time frame that a unit power reduction is taking place and within the current 8 hour period. Setpoint adjustment of the four channels is estimated to take approximately 12 hours. Further, setpoint changes should only be required for extended operation in this condition. Therefore, a Completion Time of 72 hours is proposed.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
TSTF-95-A, Revision 0, was approved by the NRC as documented in a letter from Christopher Grimes (NRC) to James Davis (NEI) dated September 27, 1996 (ACN ML9610030183). An example of a plant-specific NRC approval of the changes in TSTF-95-A is Beaver Valley Units 1 and 2 Amendment Number 274/155 dated February 27, 2006 (ACN ML060330636).

List of Affected Pages
3.2.1-1
3.2.2-1
B3.2.1-5
B3.2.2-5

Applicable Regulatory Requirements/Criteria
10 CFR 50.36(c)(2) states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

There is no regulatory requirement that specifies what remedial actions are to be taken when an LCO is not met. The ISTS for Westinghouse Plants (NUREG-1431) provides 72 hours to reduce the Power Range Neutron Flux-High trip setpoints when $F_Q(Z)$ or $F^N_{\Delta H}$ are not within limit. The proposed Completion Times are consistent with NUREG-1431.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change extends the time allowed to reduce the Power Range Neutron Flux - High trip setpoint when Specification 3.2.1, "Heat Flux Hot Channel Factor," or Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor," are not within their limits. Both specifications require a power reduction followed by a reduction in the Power Range Neutron Flux - High trip setpoint. Because reactor power has been reduced, the reactor core power distribution limits are within the assumptions of the accident analysis. Reducing the Power Range Neutron Flux - High trip setpoints ensures that reactor power is not inadvertently increased. Reducing the Power Range Neutron Flux - High trip setpoints is not an initiator to any accident previously evaluated. The consequences of any accident previously evaluated with the Power Range Neutron Flux - High trip setpoints not reduced are no different under the proposed Completion Time than under the existing Completion Time. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides additional time before requiring the Power Range Neutron Flux - High trip setpoint be reduced when the reactor core power distribution limits are not met. The manual reduction in reactor power required by the specifications provides the necessary margin of safety for this condition. Reducing the Power Range Neutron Flux - High trip setpoints carries an increased risk of a reactor trip. Delaying the trip setpoint reduction until the power reduction has been completed and the condition is verified will minimize overall plant risk. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.8 TSTF-110-A, Revision 2, "Delete SR Frequencies Based on Inoperable Alarms"

Description of Proposed Change

The proposed change eliminates Surveillance Frequencies based on inoperable alarms in Specification 3.1.4, "Rod Group Alignment Limits," SR 3.1.4.1; Specification 3.1.6, "Control Bank Insertion Limits," SR 3.1.6.2; Specification 3.2.3, "Axial Flux Difference (AFD)," SR 3.2.3.1; and Specification 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," SR 3.2.4.1.

Differences Between the Proposed Change and the Approved Traveler

The Vogtle Section 3.1 specification numbers are different from the ISTS Section 3.1 specification numbers. Vogtle Specification 3.1.4, "Rod Group Alignment Limits" is equivalent to Specification 3.1.5 in the ISTS, and Vogtle Specification 3.1.6, "Control Bank Insertion Limits" is equivalent to Specification 3.1.7 in the ISTS. This has no effect on the requested change.

The ISTS contains two alternative specifications for Axial Flux Difference to reflect different methodologies. TSTF-110-A revised Specification 3.2.3A, "AFD (CAOC Methodology)," and Specification 3.2.3B, "AFD (RAOC Methodology)." Vogtle Specification 3.2.3 is equivalent to ISTS Specification 3.2.3B.

The Bases changes identified in TSTF-110-A for SRs 3.1.5.1, 3.1.7.2, and 3.2.4.1 are either not adopted or adopted in an adaptive manner. The Bases descriptions for Vogtle SRs 3.1.4.1 and 3.1.6.2 are substantially different from the Bases text in TSTF-110-A, which is based on NUREG-1431, Revision 1. These differences result from the adoption of a Surveillance Frequency Control Program (SFCP), as described in TS 5.5.21, to control periodic surveillance frequencies. Adoption of the SFCP included deletion of Bases text that provided the basis for surveillance frequency if control of the frequency had been moved to the SFCP. NRC approval of the license change implementing the SFCP was provided in Amendment Numbers 158/140, dated January 19, 2011 (ACN ML102520083).

Summary of the Approved Traveler Justification

Surveillances on the rod position deviation monitor, the rod insertion limit monitor, the AFD monitor and the QPTR alarm contain a second, increased surveillance Frequency to be used when the associated alarms are inoperable. The requirement to perform the surveillances more frequently when the associate alarms are inoperable is removed from the Technical Specifications and the actions are placed in plant administrative practices since the alarms themselves do not directly relate to the LCO limits. This detail is not required to be in the Technical Specifications to provide adequate protection of the public health and safety. The alarms serve as indication only. Plant procedures dictate the appropriate actions to be taken under these conditions. There are no underlying reliability issues associated with relocating these alarms. There is no adverse effect in permitting the normal surveillance Frequency to be used instead of the Frequency associated with any alarms. There are no safety functions adversely effected by this change.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
The NRC did not issue a letter approving TSTF-110-A, Revision 2, however it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-110-A has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265). TSTF-110-A, Revision 2, was approved for Beaver Valley Units 1 and 2 in Amendment Numbers 225/102 dated August 30, 1999 (ACN ML003768467).

List of Affected Pages
3.1.4-3
3.1.6-3
3.2.3-1
3.2.4-4
B3.1.6-6
B3.2.3-1
B3.2.3-4
B3.2.4-7

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), Part 50,
10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

10 CFR 50.36(c)(2)(ii)(B) states:

> *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The rod position deviation monitor, the rod insertion limit monitor, the AFD monitor and the QPTR alarm are not process variables, design features, or operating restrictions that are an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The monitors and alarms are operator aids used to maintain the associated equipment and variables within established limits. Therefore, in accordance with 10 CFR 50.36, actions based on the availability of these monitors and alarms are not required to be retained in the Technical Specifications.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change removes surveillance Frequencies associated with inoperable alarms (rod position deviation monitor, rod insertion limit monitor, AFD monitor and QPTR alarm) from the Technical Specifications and places the actions in plant administrative procedures. The subject plant alarms are not an initiator of any accident previously evaluated. The subject plant alarms are not used to mitigate any accident previously evaluated, as the control room indications of these parameters are sufficient to alert the operator of an abnormal condition without the alarms. The alarms are not credited in the accident analysis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change removes surveillance Frequencies associated with inoperable alarms (rod position deviation monitor, rod insertion limit monitor, AFD monitor and QPTR alarm) from the Technical Specifications and places the actions in plant administrative procedures. The alarms are

not being removed from the plant. The actions to be taken when the alarms are not available are proposed to be controlled under licensee administrative procedures. As a result, plant operation is unaffected by this change and there is no effect on a margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.9 TSTF-142-A, Revision 0, "Increase the Completion Time When the Core Reactivity Balance is Not Within Limit"

Description of Proposed Change
The proposed change revises Specification 3.1.2, "Core Reactivity," Condition A, "Measured core reactivity not within limit," to extend the Completion Time to re-evaluate the core design and safety analysis and to determine whether the reactor core is acceptable for continued operation, and to establish appropriate operating restrictions, from 72 hours to 7 days.

Differences Between the Proposed Change and the Approved Traveler
The Vogtle Section 3.1 specification numbers are different from the ISTS Section 3.1 specification numbers. Vogtle Specification 3.1.2, "Core Reactivity" is equivalent to Specification 3.1.3 in the ISTS.

Summary of the Approved Traveler Justification
The Completion Time for actions taken when the core reactivity balance is not within limit is being increased from 72 hours to 7 days. The Required Actions require a reevaluation of core design and safety analysis and determination if the reactor core is acceptable for continued operation, and the establishment of appropriate operating restrictions and SRs within 72 hours. The 72 hours allocated to perform these actions is insufficient. Resolving a predicted versus measured reactivity anomaly is very complex. Data must be gathered, transmitted to the core design organization (which may be an offsite vendor, which would require additional administrative actions), evaluation by the core design organization, and implementation of appropriate controls. It is unlikely that these activities could be accomplished in 72 hours. Also, because exceeding this limit is very unlikely, it is important to allow sufficient time to properly analyze the causes. The proposed 7 day Completion Time is sufficient to perform these actions.

The proposed 7 day Completion Time is acceptable because of the conservatisms used in designing the reactor core and performing the safety analyses and the low probability of an accident or transient which would approach the core design limits occurring during the 7 day period.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
The NRC did not issue a letter approving TSTF-142-A, Revision 0, but it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-142-A has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265). TSTF-142-A, Revision 0, was approved for Millstone Unit 2 in Amendment Number 280 dated September 25, 2003 (ACN ML032130014).

List of Affected Pages
3.1.2-1
B3.1.2-5

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

There is no regulatory requirement that specifies what remedial actions are to be taken when an LCO is not met. The ISTS for Westinghouse Plants (NUREG-1431) provides 7 days to take action when the core reactivity balance is not within limit. The proposed Completion Time is consistent with the NUREG-1431 value.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change extends the Completion Time to take the Required Actions when measured core reactivity is not within the specified limit of the predicted values. The Completion Time to respond to a difference between predicted and measured core reactivity is not an initiator to any accident previously evaluated. The consequences of an accident during the proposed Completion Time are no different from the consequences of an accident during the existing Completion Time. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change provides additional time to investigate and to implement appropriate operating restrictions when measured core reactivity is not within the specified limit of the predicted values. The additional time will not have a significant effect on plant safety due to the conservatisms used in designing the reactor core and performing the safety analyses and the low probability of an accident or transient which would approach the core design limits during the additional time. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.10 TSTF-234-A, Revision 1, "Add Action for More Than One [D]RPI Inoperable"

Description of Proposed Change
The proposed change modifies Specification 3.1.7, "Rod Position Indication," to add a Condition for more than one inoperable digital rod position indicator (DRPI) per group, revise the Action Note to reflect the change, and to clarify the wording of Required Action B.1.

Differences Between the Proposed Change and the Approved Traveler
The Vogtle Section 3.1 specification numbers are different from the ISTS Section 3.1 specification numbers. Vogtle Specification 3.1.7, "Rod Position Indication," is analogous to Specification 3.1.8 in the ISTS.

The existing Vogtle Actions Note is worded differently than the ISTS Actions Note that is modified by TSTF-234-A due to a plant specific clarification. The proposed Actions Note is identical to the wording approved in TSTF-234-A. This has no effect on the proposed change or the justification.

TSTF-234-A contains the bracketed text "[, or B.1, as applicable]" in the Bases discussion for Condition C. This change was not adopted because it is not necessary to provide direction in the Bases that all applicable Conditions must be entered.

Summary of the Approved Traveler Justification
The proposed change adds a new Condition B which applies when more than one DRPI per group is inoperable. The proposed Required Actions allow 24 hours to restore all but one DRPI per group. The additional time to restore an inoperable DRPI is appropriate because the proposed Action would require that the control rods be under manual control, that Reactor Coolant System average temperature be monitored and recorded hourly, and that rod position be verified indirectly every 8 hours using the movable incore detectors, thereby assuring that the rod alignment and rod insertion LCOs are met. Therefore, the required shutdown margin will be maintained. Given the alternate position monitoring requirement, and other indirect means of monitoring changes in rod position (e.g., alarms on Reactor Coolant System average temperature deviation), a 24 hour Completion Time to restore all but one DRPI per group provides sufficient time to restore operability while minimizing shutdown transients during the time that the position indication system is degraded.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None.

NRC Approval
The NRC documented their approval of TSTF-234-A, Revision 1, in a letter from William D. Beckner (NRC) to James Davis (NEI) dated January 13, 1999 (ACN ML9901210038). TSTF-234-A, Revision 1. TSTF-234-A has been

adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265). TSTF-234-A, Revision 1, was approved for Kewaunee in Amendment Number 176 dated September 22, 2004 (ACN ML042230068).

List of Affected Pages
3.1.7-1
3.1.7-2
B3.1.7-4
B3.1.7-5
B3.1.7-6

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

There is no regulatory requirement that specifies what remedial actions are to be taken when an LCO is not met. The ISTS for Westinghouse Plants (NUREG-1431) provides remedial actions for more than one DPRI inoperable in a group. The proposed change is consistent with NUREG-1431.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change provides a Condition and Required Actions for more than one inoperable digital rod position indicator (DRPI) per rod group. The DRPIs are not an initiator of any accident previously evaluated. The DRPIs are one indication used by operators to verify control rod insertion following an accident, however other indications are available. Therefore, allowing a finite period to time to correct more than one inoperable DRPI prior to requiring a plant shutdown will not result in a significant increase in the consequences of any accident previously

evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides time to correct the condition of more than one DRPI inoperable in a rod group. Compensatory measures are required to verify that the rods monitored by the inoperable DRPIs are not moved to ensure that there is no effect on core reactivity. Requiring a plant shutdown with inoperable rod position indications introduces plant risk and should not be initiated unless the rod position indication cannot be repaired in a reasonable period of time. As a result, the safety benefit provided by the proposed Condition offsets the small decrease in safety resulting from continued operation with more than one inoperable DRPI. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.11 TSTF-245-A, Revision 1, "AFW Train Operable When in Service"

Description of Proposed Change
The proposed TS modifies Surveillance Requirement 3.7.5.1, 3.7.5.3, and 3.7.5.4 to add a Note stating that "AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation."

Differences Between the Proposed Change and the Approved Traveler
ISTS SR 3.7.5.3 contains a note stating that the SR is "Not applicable in MODE 4 when steam generator is relied upon for heat removal." The approved Traveler deletes and replaces this note. Vogtle SR 3.7.5.3 does not currently include this note, and will add the note identified in the approved Traveler under this change. Additionally, the Bases for SR 3.7.5.1 includes clarifying text that is supplemental to that provided in the ISTS. As a result of this change, the supplemental text is no longer necessary and is deleted. These differences do not affect the applicability of the traveler justification.

Summary of the Approved Traveler Justification
Auxiliary Feedwater (AFW) is a dual use system. As such, AFW valves may be positioned other than that required for decay and residual heat removal during Modes I (below 10% Rated Thermal Power), 2, 3, 4, and 5, when the AFW system is being used to maintain steam generator level. Adding a Note stating that an AFW train may be considered operable during alignment and operation for steam generator level control, if capable of being manually realigned to the AFW mode of operation would clarify the intended dual-use flexibility of the AFW system and prevent unnecessary Action entry.

The Note provides an exception that allows the AFW system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the affected AFW train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an expected function of the AFW system, operability (i.e., the intended safety function) should be maintained during these operations. Additionally, following a reactor trip, AFW flow provides the source of makeup to the steam generators. If excessive RCS cooldown is experienced and it is caused by a large amount of AFW flow, the Turbine Driven AFW Pump may be stopped in order to limit RCS cooldown. However, the Turbine Driven AFW Pump would still remain available for steam generator level control and could be restored by the operator should the need arise.

NUREG-1431 incorporates the changes identified in TSTF-245-A, and includes a Note in SRs 3.7.5.1, 3.7.5.3 and 3.7.5.4 stating that "AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation."

Vogtle Operating Procedures and Emergency Operating Procedures contain steps to support realignment of the AFW system from manual steam generator level control mode to the emergency operation mode when required.

With regard to the AFW system, the NRC staff has previously issued a determination[1] on the effects of manual operation on Operability of the AFW system, which concluded that manual operation does not render the AFW system inoperable, provided manual action can perform the same function. The NRC recognizes that AFW is a dual-use system and may be used during startup of the plant, normal shutdown, and hot standby conditions, and that it is control band operated during these conditions in the manual mode of operation. In such situations, the AFW system may be considered operable. The NRC letter can be found as an attachment to TSTF-245-A.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC did not issue a letter approving TSTF-245-A, Revision 1; however, it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-245-A, Revision 1 has been adopted by many plants as part of complete conversion to the ISTS, such as Beaver Valley Power Station (ACN ML050610351). An example of a plant-specific NRC approval of the changes in TSTF-245-A is Comanche Peak Units 1 and 2, Amendment Numbers 126/126 dated April 4, 2006 (ACN ML060860258).

List of Affected Pages
3.7.5-3
B3.7.5-7
B3.7.5-8

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), Part 50, 10 CFR 50.36(c)(2)(ii)(B), states:

> *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The proposed TS changes would allow the AFW train(s) to be considered operable during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. In practice, this would allow the AFW valves may be positioned other than that required for decay and residual heat removal during Modes I (below 10% Rated Thermal Power), 2, 3, 4, and 5, when the AFW system is being used to maintain steam generator level. The decay and residual heat loads will be low during the low

[1] Harold, Jeffrey F. (NRC) to Stephen E. King (Consolidated Edison), "Manual vs. Automatic Operation as it Relates to Auxiliary Feedwater Operability at Indian Point Nuclear Generating Unit No. 2 (TAC No. M98056)", dated May 23, 1997.

power and shutdown conditions where the Note would apply, and there is sufficient time to realign the AFW system from manual steam generator level control mode to the AFW mode if needed.

There will be no changes to the auxiliary feedwater system design such that compliance with the regulatory requirements and guidance document above would come into question. The auxiliary feedwater system will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the requirements in Technical Specification 3.7.5, "Auxiliary Feedwater (AFW) System," to clarify the operability of an AFW train when it is aligned for manual steam generator level control. The AFW System is not an initiator of any design basis accident or event, and therefore the proposed change does not increase the probability of any accident previously evaluated. The AFW System is used to respond to accidents previously evaluated. The proposed change does not affect the design of the AFW System, and no physical changes are made to the plant. The proposed change does not significantly change how the plant would mitigate an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not result in a change in the manner in which the AFW System provides plant protection. The AFW System will continue to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators. There are no design changes associated with the proposed changes, and the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be

installed). The change does not alter assumptions made in the safety analysis, and is consistent with the safety analysis assumptions and current plant operating practice. Manual control of AFW level control valves is not an accident initiator. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change provides the operational flexibility of allowing an AFW train(s) to be considered operable when it is not in the normal standby alignment and is temporarily incapable of automatic initiation, such as during alignment and operation for manual steam generator level control, provided it is capable of being manually realigned to the AFW heat removal mode of operation. The proposed change does not result in a change in the manner in which the AFW System provides plant protection. The AFW System will continue to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators. The proposed change does not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change will not result in plant operation in a configuration outside the design basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.12 TSTF-247-A, Revision 0, "Provide Separate Condition Entry for Each PORV and Block Valve"

Description of Proposed Change
The proposed change modifies Specification 3.4.11, "Pressurizer PORVs," to provide separate Condition entry for each PORV and each block valve.

Differences Between the Proposed Change and the Approved Traveler
None. However, TSTF-247-A provides options depending on the number of PORV and block valves that are included in the plant design. The design of Vogtle, Units 1 and 2, includes two PORVs and associated block valves. The options from TSTF-247-A for plants with three PORVs and associated block valves are not adopted.

Summary of the Approved Traveler Justification
The existing LCO 3.4.11 Conditions allow separate condition entry for each pressurizer power operated relief valve (PORV). The Conditions and Required Actions provide appropriate compensatory measures for separate condition entry for each inoperable PORV. The Conditions and Required Actions also provide appropriate compensatory actions for separate condition entry for each block valve. Therefore, the Actions Note is modified to allow separate condition entry for each block valve.

Condition F is modified to apply when both block valves are inoperable. The existing Actions are modified to no longer require that both PORVs be placed in manual control if both block valves are inoperable. This avoids a potential situation where a plant shutdown is required if one of the block valves cannot be restored within 2 hours, and the PORVs, which will be needed for Low Temperature overpressure protection, cannot perform their automatic pressure relief function. Deletion of Action F.1 removes an unnecessary requirement since separate condition entry for each block valve makes it redundant with Action C.1. Action F.3 is also eliminated (Restore remaining block valve(s) to operable status). With separate condition entry for each block valve this ACTION is no longer necessary.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC did not issue a letter approving TSTF-247-A, Revision 0; however, it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. TSTF-247-A, Revision 2 has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265). An example of a plant-specific NRC approval of the changes in TSTF-247-A is Callaway Unit 1, Amendment Number 188, dated November 25, 2008 (ACN ML082910895).

List of Affected Pages
3.4.11-1
3.4.11-2
3.4.11-3
B3.4.11-4
B3.4.11-6

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), Part 50, 10 CFR 50.36(c)(2)(ii)(C), states:

> *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

There will be no changes to the pressurizer PORV or block valve design or operation such that compliance with any of the regulatory requirements and guidance documents above would come into question. There will be no changes to the plant design or operations such that compliance with any of the regulatory requirements and guidance documents above would come into question. The plant and its systems will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the requirements in Technical Specification 3.4.11, "Pressurizer PORVs," to clarify that separate Condition entry is allowed for each block valve. Additionally, the Actions are modified to no longer require that the PORVs be placed in manual operation when both block valves are inoperable and cannot be restored to operable status within the specified Completion Time. This preserves the overpressure protection capabilities of the PORVs. The pressurizer block valves are used to isolate their respective PORV in the event it is experiencing excessive leakage, and are not an initiator of any design basis accident or event. Therefore the proposed change does not increase the probability of any accident previously evaluated. The PORV and block valves are

used to respond to accidents previously evaluated. The proposed change does not affect the design of the PORV and block valves, and no physical changes are made to the plant. The proposed change does not change how the plant would mitigate an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not result in a change in the manner in which the PORV and block valves provide plant protection. The PORVs will continue to provide overpressure protection, and the block valves will continue to provide isolation capability in the event a PORV is experiencing excessive leakage. There are no design changes associated with the proposed changes, and the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The change does not alter assumptions made in the safety analysis, and is consistent with the safety analysis assumptions and current plant operating practice. Operation of the PORV block valves is not an accident initiator. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes provide clarification that separate Condition entry is allowed for each block valve. Additionally, the Actions are modified to no longer require that the PORVs be placed in manual operation when both block valves are inoperable and cannot be restored to operable status within the specified Completion Time. This preserves the overpressure protection capabilities of the PORVs. The proposed change does not result in a change in the manner in which the PORV and block valves provide plant protection. The PORVs will continue to provide overpressure protection, and the block valves will continue to provide isolation capability in the event a PORV is experiencing excessive leakage. The proposed change does not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change will not result in plant operation in a configuration outside the design basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.13 TSTF-248-A, Revision 0, "Revise Shutdown Margin Definition for Stuck Rod Exception"

Description of Proposed Change
The proposed change revises the definition of Shutdown Margin to eliminate the requirement that shutdown margin calculations must assume the single rod cluster control assembly (RCCA) of highest worth is fully withdrawn if all RCCAs can be verified to be fully inserted by two independent means.

Differences Between the Proposed Change and the Approved Traveler
None

Summary of the Approved Traveler Justification
The Shutdown Margin definition states, "SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures."

The proposed change modifies the definition to include, "However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation."

The consideration of a stuck rod is provided to allow for a single failure of one rod to not insert when a scram is initiated. However, with positive indication that all rods are already fully inserted, such a provision is overly conservative. This change is consistent with the definition of Shutdown Margin provided in NUREG-1431.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC documented their approval of TSTF-248-A, Revision 0, in a letter from William D. Beckner (NRC) to Anthony R. Pietrangelo (NEI), dated October 31, 2000 (ACN ML003775261). An example of a plant-specific NRC approval of the changes in TSTF-248-A is Catawba Units 1 and 2, McGuire Units 1 and 2, and Oconee Units 1, 2, and 3 amendments, dated May 28, 2010 (ACN ML101390415).

List of Affected Pages
1.1-6

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criteria:

Criterion 25, Protection System Requirements for Reactivity Control Malfunctions, states:

> The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 27, Combined Reactivity Control Systems Capability, states:

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Typically the shutdown margin calculations assume the most reactive control rod fails insert into the core (i.e., a stuck rod). However, when it can be confirmed by two independent methods that all rods are inserted, it is not appropriate to include a margin for stuck rods.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change modifies the definition of Shutdown Margin to eliminate the requirement to assume the highest worth control rod is fully withdrawn when calculating Shutdown Margin if it can be verified by two independent means that all control rods are inserted. The method for calculating shutdown margin is not an imitator of any accident previously evaluated. If it can be verified by two independent means that all control rods are inserted, the calculated Shutdown Margin without the

conservatism of assuming the highest worth control rod is withdrawn is accurate and consistent with the assumptions in the accident analysis. As a result, the mitigation of any accident previously evaluated is not affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change modifies the definition of Shutdown Margin to eliminate the requirement to assume the highest worth control rod is fully withdrawn when calculating Shutdown Margin if it can be verified by two independent means that all control rods are inserted. The additional margin of safety provided by the assumption that the highest worth control rod is fully withdrawn is unnecessary if it can be independently verified that all controls rods are inserted. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.14 TSTF-266-A, Revision 3, "Eliminate the Remote Shutdown System Table of Instrumentation and Controls"

Description of Proposed Change
The proposed change removes the list of Remote Shutdown System instrumentation and controls in Specification 3.3.4, "Remote Shutdown System," from the Technical Specifications and places them in the Technical Specification Bases.

Differences Between the Proposed Change and the Approved Traveler
None

Summary of the Approved Traveler Justification
This change eliminates the table of instrumentation and controls referenced in the Specification for the Remote Shutdown System. The specific instruments and controls necessary for each Function provided by the Remote Shutdown System are currently listed in a Table in the Specifications. This change will eliminate the table and the information will be placed in the Bases. It is unnecessary to list the specific instruments and controls in the Technical Specifications to provide adequate assurance that the functions can be performed. GDC 19 requires that the remote shutdown capability be provided. The LCO provides references to the Functions, which are described in the Bases. This is sufficient to ensure that the system will be operable. Listing the specific instrumentation and controls is unnecessary and may lead to needless expenditure of licensee and NRC resources processing license amendments to revise the table when the information can be adequately controlled by the licensee.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC documented their approval of TSTF-266-A, Revision 3, in a letter from William D. Beckner (NRC) to James Davis (NEI), dated September 10, 1999 (ACN ML9909160189). An example of a plant-specific NRC approval of the changes in TSTF-266-A is South Texas Project, Units 1 and 2, Amendment Numbers 163/152 dated August 20, 2004 (ACN ML042370841).

List of Affected Pages
3.3.4-1
3.3.4-3
B3.3.4-2
B3.3.4-3
B3.3.4-5

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 19, Control Room, states:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Criterion 19 requires that remote shutdown capability be provided. The Remote Shutdown System Functions are described in the Updated Final Safety Analysis Report. The definition of "operable" in the Vogtle specifications states that a system shall be operable or have operability when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system to perform its specified safety function(s) are also capable of performing their related support function. This definition provides adequate guidance for determining what instrumentation and controls are necessary for a particular remote shutdown function. The ability to transfer control of a function from the control room to the remote shutdown panel is a required support function by the definition of operability. Therefore, listing specific instrumentation and controls is unnecessary and may lead to needless expenditure of licensee and NRC resources processing license amendments to revise the Remote Shutdown System details in the Technical Specifications when these details are not necessary to describe the actual regulatory requirements. Therefore, they can be removed from the Technical Specifications and placed in the Bases without a significant impact on safety.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change removes the list of Remote Shutdown System instrumentation and controls from the Technical Specifications and places them in the Bases. The Technical Specifications continue to require that the instrumentation and controls be operable. The location of the list of Remote Shutdown System instrumentation and controls is not an initiator to any accident previously evaluated. The proposed change will have no effect on the mitigation of any accident previously evaluated because the instrumentation and controls continue to be required to be operable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change removes the list of Remote Shutdown System instrumentation and controls from the Technical Specifications and places it in the Bases. The review performed by the NRC when the list of Remote Shutdown System instrumentation and controls is revised will no longer be needed unless the criteria in 10 CFR 50.59 are not met such that prior NRC review is required. The Technical Specification requirement that the Remote Shutdown System be operable, the definition of operability, the requirements of 10 CFR 50.59, and the Technical Specifications Bases Control Program are sufficient to ensure that revision of the list without prior NRC review and approval does not introduce a significant safety risk. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.15 TSTF-272-A, Revision 1, "Refueling Boron Concentration Clarification"

Description of Proposed Change
The proposed change adds an Applicability Note to Specification 3.9.1, "Boron Concentration." The Note clarifies that the LCO only applies to the refueling canal and the refueling cavity when those volumes are connected to the Reactor Coolant System.

Differences Between the Proposed Change and the Approved Traveler
None

Summary of the Approved Traveler Justification
Specification 3.9.1, "Boron Concentration," is revised to clarify that the boron concentration limits do not apply to the refueling canal and refueling cavity when these areas are not connected to the RCS. This Specification limits the boron concentrations of the RCS, the refueling canal, and the refueling cavity during refueling to ensure that the reactor remains subcritical. However, when the refueling canal and refueling cavity are isolated from the RCS, no potential for dilution exists. In this condition it is not necessary to place a limit on the boron concentration of water in the refueling cavity and the refueling canal. This change is consistent with the intent of the Specification and eliminates restrictions that have no effect on safety.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC documented their approval of TSTF-272-A, Revision 1, in a letter from William D. Beckner (NRC) to James Davis (NEI), dated December 12, 1999. An example of a plant-specific NRC approval of the changes in TSTF-272-A is Millstone Unit 2, Amendment Number 263, dated January 11, 2002 (ACN ML013440338).

List of Affected Pages
3.9.1-1
B3.9.1-3
B3.9.1-4

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 28, Reactivity Limits, states:

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor

coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 61, Fuel Storage and Handling and Radioactivity Control, states:
The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

The proposed change clarifies that the limits on Reactor Coolant System boron concentration are only applicable to those portions of the Reactor Coolant System that are in communication with the reactor core and can, therefore, affect core reactivity.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change modifies the Applicability of Specification 3.9.1, "Boron Concentration," to clarify that the boron concentration limits are only applicable to the refueling canal and the refueling cavity when those volumes are attached to the Reactor Coolant System (RCS). The boron concentration of water volumes not connected to the RCS are not an initiator of an accident previously evaluated. The ability to mitigate any accident previously evaluated is not affected by the boron concentration of water volumes not connected to the RCS. Therefore, the proposed

change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change modifies the Applicability of Specification 3.9.1, "Boron Concentration," to clarify that the boron concentration limits are only applicable to the refueling canal and the refueling cavity when those volumes are attached to the RCS. Technical Specification SR 3.0.4 requires that Surveillances be met prior to entering the Applicability of a Specification. As a result, the boron concentration of the refueling cavity or the refueling canal must be verified to satisfy the LCO prior to connecting those volumes to the RCS. The margin of safety provided by the refueling boron concentration is not affected by this change as the RCS boron concentration will continue to satisfy the LCO. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.16 TSTF-273-A, Revision 2, "Safety Function Determination Program Clarifications"

Description of Proposed Change
The proposed TS changes adds explanatory text to the LCO 3.0.6 Bases clarifying the "appropriate LCO for loss of function," and that consideration does not have to be made for a loss of power in determining loss of function. Explanatory text is also added to the programmatic description of the Safety Function Determination Program (SFDP) in Specification 5.5.15 to provide clarification of these same issues.

Differences Between the Proposed Change and the Approved Traveler
None.

Summary of the Approved Traveler Justification
TS 5.5.15, "Safety Function Determination Program," implements the requirements of LCO 3.0.6. The SFDP program description in TS 5.5.15 is revised to clarify in the requirements that consideration does not have to be made for a loss of power in determining loss of function. TS 5.5.15 is also revised to incorporate an editorial change for consistency in meaning. The Bases for LCO 3.0.6 is revised to provide clarification of the "appropriate LCO for loss of function," and that consideration does not have to be made for a loss of power in determining loss of function.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
TSTF-273-A, Revision 2, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated August 16, 1999. TSTF-273-A, Revision 2 has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML021200265). An example of a plant-specific NRC approval of the changes in TSTF-273-A, Revision 2 is Susquehanna Steam Electric Station, Units 1 and 2, Amendment Numbers 209/183 dated February 25, 2003 (ACN ML060860258).

List of Affected Pages
5.5-15
B3.0-9

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow

any remedial action permitted by the technical specifications until the condition can be met.

The SFDP, as described in TS 5.5.15, implements the requirements of Limiting Condition for Operation (LCO) 3.0.6, and ensures that loss of safety function is detected and appropriate actions are taken. There will be no changes to the plant design or operation such that compliance with the regulatory requirements and guidance document above would come into question. The plant and its systems will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed TS changes add explanatory text to the programmatic description of the Safety Function Determination Program (SFDP) in Specification 5.5.15 to clarify in the requirements that consideration does not have to be made for a loss of power in determining loss of function. The Bases for LCO 3.0.6 is revised to provide clarification of the "appropriate LCO for loss of function," and that consideration does not have to be made for a loss of power in determining loss of function. The changes are editorial and administrative in nature, and therefore do not increase the probability of any accident previously evaluated. No physical or operational changes are made to the plant. The proposed change does not change how the plant would mitigate an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes are editorial and administrative in nature and do not result in a change in the manner in which the plant operates. The loss of function of any specific component will continue to be addressed in its specific TS LCO and plant configuration will be governed by the required

actions of those LCOs. The proposed changes are clarifications that do not degrade the availability or capability of safety related equipment, and therefore do not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no design changes associated with the proposed changes, and the changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The changes do not alter assumptions made in the safety analysis, and are consistent with the safety analysis assumptions and current plant operating practice. Due to the administrative nature of the changes, they cannot be an accident initiator. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes to TS 5.5.15 are clarifications and are editorial and administrative in nature. No changes are made the LCOs for plant equipment, the time required for the TS Required Actions to be completed, or the out of service time for the components involved. The proposed changes do not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed changes will not result in plant operation in a configuration outside the design basis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.17 TSTF-284-A, Revision 3, "Add 'Met vs. Perform' to Technical Specification 1.4, Frequency"

Description of Proposed Change
The change inserts a discussion paragraph into Specification 1.4, and several new examples are added to facilitate the use and application of SR Notes that utilize the terms "met" and "perform." The changes also modify SR 3.4.11.1, SR 3.4.11.2, SR 3.4.12.4, and SR 3.4.9.2 to appropriately use "met" and "perform" exceptions.

Differences Between the Proposed Change and the Approved Traveler
TSTF-284-A, Revision 3 includes changes to SR 3.1.11.1 and SR 3.1.11.2 of ISTS Specification 3.1.11, "SDM Test Exceptions." This LCO allows suspension of SDM requirements in MODE 2 provided specific conditions are met in order facilitate measurement control rod worth and SDM. The Vogtle Technical Specifications do not include a Specification that is analogous to ISTS TS 3.1.11, "SDM Test Exceptions," or SRs that are analogous to ISTS SRs 3.1.11.1 and 3.1.11.2. Therefore, the TS and Bases changes identified in TSTF-284-A for ISTS 3.1.11 are not adopted.

Changes to the Actions Bases for Specification 3.4.11, Pressurizer PORVs," are not adopted. The changes described in the TSTF are related to a Note in the ISTS that provides an exception to LCO 3.0.4 that allows entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves in order to demonstrate their operability. Consistent with NUREG-1431, Vogtle Technical Specification 3.4.11, and its associated Bases, do not include the Note providing this exception to LCO 3.0.4.

The Bases changes identified in TSTF-284-A for SR 3.4.12.8 is not adopted. The Bases descriptions for corresponding Vogtle SR 3.4.12.4 is substantially different from the Bases text in TSTF-284-A, which is based on NUREG-1431, Revision 1. These differences result from the adoption of a Surveillance Frequency Control Program (SFCP), as described in TS 5.5.21, to control periodic surveillance frequencies. Adoption of the SFCP included deletion of Bases text that provided the basis for surveillance frequency if control of the frequency had been moved to the SFCP. NRC approval of the license change implementing the SFCP was provided in Amendment Numbers 158/140, dated January 19, 2011 (ACN ML102520083).

Summary of the Approved Traveler Justification
The change inserts a discussion paragraph into Specification 1.4, and several new examples are added to facilitate the use and application of SR Notes that utilize the terms "met" and "perform." The changes also modify SRs as necessary to appropriately use "met" and "perform" exceptions. The added examples parallel the existing example 1.4-3 of Notes that allow for the SR to be "Not required to be performed . . .". The examples will alleviate misunderstanding and provide explicit direction for these types of SR Notes.

NUREG-1433 (BWR/4 plants) and -1434 (BWR/6 plants) contain a discussion in Specification 1.4 regarding the use of "met" and "performed" in SR Notes. Similarly, the Writer's Guide provides a distinction between these phrases.

However, NUREG-1430 (B&W), -1431 (Westinghouse), and -1432 (Combustion Engineering) dis not originally contain this detail, even though various locations throughout these NUREGs provide Notes with "met" and "performed" distinctions. Inserting this material will provide for better use, application, and understanding of these Notes. Furthermore, this change will establish consistency between the NUREGs. With this clarification, several exceptions that are unclear or have incorrect usage of "met" and "perform" are also corrected.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
TSTF-284-A, Revision 3, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated February 16, 2000 (ACN ML003684596). TSTF-284-A, Revision 3 has been adopted by many plants as part of complete conversion to the ISTS, such as Beaver Valley Power Station (ACN ML070160593). An example of a plant-specific NRC approval of the changes in TSTF-284-A, Revision 3 is Columbia Generating Station, Amendment Number 205 dated December 13, 2007 (ACN ML073120270).

List of Affected Pages
1.4-1
1.4-4
3.4.11-3
3.4.12-4
3.9.4-2
B3.4.11-7
B3.4.12-13
B3.9.4-7

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

The changes insert a discussion paragraph into Specification 1.4, and several new examples are added to facilitate the use and application of SR Notes that utilize "met" and "perform." With this clarification, several exceptions that are unclear or have incorrect usage of "met" and "perform" are also corrected. The changes also modify SR 3.4.11.1, SR 3.4.11.2, SR 3.4.12.4, and SR 3.4.9.2 to appropriately use "met" and "perform" exceptions. The changes to LCO 1.4 clarify implementation of the requirements for LCOs that have "met" or "performed" exceptions. There will be no changes to the plant design or

operations such that compliance with any of the regulatory requirements and guidance documents above would come into question. The plant and its systems will continue to comply with all applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes insert a discussion paragraph into Specification 1.4, and several new examples are added to facilitate the use and application of SR Notes that utilize the terms "met" and "perform." The changes also modify SRs in multiple Specifications to appropriately use "met" and "perform" exceptions. The changes are administrative in nature because they provide clarification and correction of existing expectations, and therefore the proposed change does not increase the probability of any accident previously evaluated. No physical or operational changes are made to the plant. The proposed change does not significantly change how the plant would mitigate an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes are administrative in nature and do not result in a change in the manner in which the plant operates. The proposed changes provide clarification and correction of existing expectations that do not degrade the availability or capability of safety related equipment, and therefore do not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no design changes associated with the proposed changes, and the changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The changes do not alter assumptions made in the safety analysis, and are consistent with the safety analysis assumptions and current plant operating practice. Due to the administrative nature of the changes, they cannot be an accident initiator.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes are administrative in nature and do not result in a change in the manner in which the plant operates. The proposed changes provide clarification and correction of existing expectations that do not degrade the availability or capability of safety related equipment, or alter their operation. The proposed changes do not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed changes will not result in plant operation in a configuration outside the design basis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.18 TSTF-308-A, Revision 1, "Determination of Cumulative and Projected Dose Contributions in RECP"

Description of Proposed Change
The proposed change revises Specification 5.5.4, "Radioactive Effluent Controls Program," paragraph e, to describe the original intent of the dose projections.

Differences Between the Proposed Change and the Approved Traveler
None

Summary of the Approved Traveler Justification
The proposed change revises Specification 5.5.4, "Radioactive Effluent Controls Program," paragraph e, to describe the original intent of the dose projections.

The NRC's draft Standard Technical Specifications for four-loop Westinghouse plants (8/14/87 letter to Texas Utilities) included Radioactive Effluent Technical Specifications. The two Surveillances in those draft Standard Technical Specifications reflect the intent of Vogtle Specification 5.5.4, paragraph e. SR 4.11.1.2 for Dose stated, "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 ODCM at least once per 31 days." SR 4.11.1.3.1 for Liquid Radwaste Treatment System stated, "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 Radwaste Treatment Systems are not being fully utilized." Generic Letter 89-01 inappropriately combined these two Surveillance Requirements for cumulative and projected doses and can be interpreted to require determining projected dose contribution for the current calendar quarter and current calendar year every 31 days. Therefore, the proposed change clarifies the wording in 5.5.4.e to not require dose projections for a calendar quarter and a calendar year every 31 days.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC did not issue a letter approving TSTF-308-A, Revision 1, however it was incorporated by the NRC into Revision 2 of the ISTS NUREGs. An example of a plant-specific NRC approval of the changes in TSTF-308-A is Calvert Cliffs Units 1 and 2 Amendment Numbers 259/236 dated July 16, 2003 (ACN ML031330142).

List of Affected Pages
5.5-3

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 64, Monitoring Radioactivity Releases, states:

> Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

The proposed change is an administrative requirement related to monitoring effluent discharge. It clarifies the intent of the NRC's guidance published in Generic Letter 89-01.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration
SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises Specification 5.5.4, "Radioactive Effluent Controls Program," paragraph e, to describe the original intent of the dose projections. The cumulative and projection of doses due to liquid releases are not an assumption in any accident previously evaluated and have no effect on the mitigation of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed

change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change revises Specification 5.5.4, "Radioactive Effluent Controls Program," paragraph e, to describe the original intent of the dose projections. The cumulative and projection of doses due to liquid releases are administrative tools to assure compliance with regulatory limits. The proposed change revises the requirement to clarify the intent, thereby improving the administrative control over this process. As a result, any effect on the margin of safety should be minimal. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.19 TSTF-312-A, Revision 1, "Administrative Control of Containment Penetrations"

Description of Proposed Change
The proposed TS changes add a Note to the LCO for Specification 3.9.4, "Containment Penetrations," allowing penetration flow path(s) that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control.

Differences Between the Proposed Change and the Approved Traveler
Vogtle LCO 3.9.4.b was previously amended to allow the personnel and equipment airlocks to remain open during core alterations or movement of irradiated fuel assemblies within the containment, provided one airlock door was available and a designated individual was available to close the open airlock door(s) if needed. The scope of this previous amendment overlaps the scope of TSTF-312-A, and as a result LCO 3.9.4 and its associated Bases are not identical to those presented in TSTF-312-A. The Note for LCO 3.9.4 and the supplemental LCO text for Bases 3.9.4 are incorporated without change from TSTF-312-A. No additional changes to the LCO and Bases were necessary of made as a result of the existing allowance for the personnel and equipment airlock.

Summary of the Approved Traveler Justification
The proposed TS change adds a Note to the LCO for Specification 3.9.4, "Containment Penetrations," allowing "Penetration flow path(s) that have direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control." The Applicability for LCO 3.9.4 is during core alterations, and during movement of irradiated fuel assemblies within containment.

The changes proposed in TSTF-312-A, Revision 1 are consistent with those in Specification 3.6.3, "Containment Isolation Valves." TS 3.6.3, Actions Note 1, allows penetration flow path(s) (except for the 24 inch purge valves) to be unisolated intermittently under administrative control, and is Applicable in MODES 1, 2, 3, and 4. Under the applicable conditions for LCO 3.6.3, the accident analyses credit the primary containment as a release barrier. The proposed change to LCO 3.9.4 would be Applicable under significantly lower energy conditions than those that apply for LCO 3.6.3, and is therefore less risk significant. Adoption of this change is proposed to provide a consistent approach to containment boundary issues that utilizes previously approved and acceptable compensatory measures.

The proposed change also includes the addition of text to the LCO discussion in Bases 3.9.4 stipulating that the administrative controls that are put in place when penetrations flow path(s) are unisolated ensure that: 1) appropriate personnel are aware of the open status of the penetration flow path during core alterations or movement of irradiated fuel assemblies within the containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident (FHA).

TSTF-312-A includes a Reviewer's Note that identifies the need for a confirmatory FHA dose calculation that has been accepted by the NRC staff, and that indicates acceptable radiological consequences. NRC acceptance of the Vogtle FHA dose calculation was documented during review of a prior license amendment request affecting LCO 3.9.4 that allowed the personnel airlock to remain open during core alterations or movement of irradiated fuel assemblies within the containment (see LCO 3.9.4.b). NRC acceptance of this change was based on doses for a 2 hour release and a licensee commitment for a designated and available person to close the airlock door. This acceptance is documented in a letter from Louis Wheeler (NRC) to C. K. McCoy (SNC), Amendments 92/70, dated November 30, 1995 (ACN ML012350007).

The Reviewer's Note also identifies the need for a licensee commitment to implement administrative procedures that ensure the open containment airlock can be promptly closed in the event of an FHA following personnel evacuation, and that open penetration flow path(s) can be promptly closed. The Reviewer's Note identifies that the time to close such penetrations, or combinations of penetrations, will be included in the confirmatory dose calculations.

SNC will establish administrative controls to ensure: 1) appropriate personnel are aware of the open status of the penetration flow path(s) during core alterations or movement of irradiated fuel assemblies within the containment, and 2) specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment. SNC will also include the time needed to close open containment penetrations in the confirmatory dose calculation for FHAs.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None.

Licensee Commitments Required to Adopt this Change
1. Administrative controls will be established to ensure appropriate personnel are aware of the open status of the penetration flow path(s) during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment.
2. Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.
3. The time needed to close open containment penetration(s) will be incorporated into the confirmatory dose calculation for FHAs.

NRC Approval
TSTF-312-A, Revision 1, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated August 16, 1999 (ACN ML9908250220). TSTF-312-A, Revision 1, has been adopted by many plants as part of complete conversion to the ISTS, such as North Anna Power Station (ACN ML0212110540). An example of a plant-specific NRC approval of

the changes in TSTF-312-A, Revision 1, is Arkansas Nuclear One, Unit 2, Amendment Number 245 dated August 10, 2011 (ACN ML111940085).

List of Affected Pages
3.9.4-1
B3.9.4-5

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 56—Primary Containment Isolation, states:

> Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
>
> (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
>
> (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
>
> Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

The proposed change to LCO 3.9.4 will allow containment penetration flow path(s) to be open during refueling operations under administrative control. This change does not significantly change how the plant would mitigate an accident previously evaluated, and is bounded by the existing FHA accident analysis.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change would allow containment penetrations to be unisolated under administrative controls during core alterations or movement of irradiated fuel assemblies within containment. The status of containment penetration flow paths (i.e., open or closed) is not an initiator for any design basis accident or event, and therefore the proposed change does not increase the probability of any accident previously evaluated. The proposed change does not affect the design of the primary containment, or alter plant operating practices such that the probability of an accident previously evaluated would be significantly increased. The proposed change does not significantly change how the plant would mitigate an accident previously evaluated, and is bounded by the fuel handling accident (FHA) accident analysis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Allowing penetration flow paths to be open is not an initiator for any accident. The proposed change to allow open penetration flow paths will not affect plant safety functions or plant operating practices such that a new or different accident could be created. There are no design changes associated with the proposed changes, and the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The change does not alter assumptions made in the safety analysis, and is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   TS 3.9.4 provides measures to ensure that the dose consequences of a postulated FHA inside containment are minimized. The proposed change to LCO 3.9.4 will allow penetration flow path(s) to be open during refueling operations under administrative control. These administrative controls will

provide assurance that prompt closure of open penetrations flow paths can and will be achieved in the event of an FHA inside containment, and will minimize dose consequences. The proposed change is bounded by the existing FHA analysis. The proposed change does not affect the safety analysis acceptance criteria for any analyzed event, nor is there a change to any safety analysis limit. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined, nor is there any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change will not result in plant operation in a configuration outside the design basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.20 TSTF-314-A, Revision 0, "Require Static and Transient F_Q Measurement"

Description of Proposed Change
The proposed change revises the Required Actions of Specification 3.1.4, "Rod Group Alignment Limits," and Specification 3.2.4, "Quadrant Power Tilt Ratio," to require measurement of both the steady state and transient portions of the Heat Flux Hot Channel Factor, $F_Q(Z)$.

Differences Between the Proposed Change and the Approved Traveler
The Vogtle Section 3.1 specification numbers are different from the ISTS Section 3.1 specification numbers. Vogtle Specification 3.1.4, "Rod Group Alignment Limits" is equivalent to Specification 3.1.5 in the ISTS. This has no effect on the requested change.

The Bases changes associated with TSTF-314-A are adopted with exception. TSTF-314-A modifies the Bases 3.1.5 discussion for Actions B.2, B.3, B.4, B.5 and B.6, and the Bases 3.2.4 discussion for Actions A.3 and A.6, to explicitly state that verification of $F_Q(Z)$ is within limits requires verification of both the steady state and transient portions of $F_Q(Z)$. The Vogtle Specifications do not use the same symbols as the ISTS for the steady state and transient portion of $F_Q(Z)$. The Bases are revised to reflect the Vogtle terminology that is in use.

Summary of the Approved Traveler Justification
$F_Q(Z)$ is approximated by both a steady state and transient component of F_Q. When Actions require that $F_Q(Z)$ be verified to be within limits, both the steady state and transient portions of $F_Q(Z)$ should be confirmed to be within their limits. Currently, the Rod Group Alignment Limits and Quadrant Power Tilt Specifications only require measurement of the steady state $F_Q(Z)$, as determined by SR 3.2.1.1. Both specifications are revised to also require measurement of the transient $F_Q(Z)$, as determined by SR 3.2.1.2. This change will ensure that the hot channel factors are within their limits when the rod alignment limits or quadrant power tilt ratio are not within their limits.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC documented their approval of TSTF-314-A, Revision 0 in a letter from William D. Beckner (NRC) to James Davis (NEI) dated January 13, 1999 (ACN ML9901210038). TSTF-314-A has been adopted by many plants as part of complete conversion to the ISTS, such as Donald C. Cook Nuclear Plant Amendment Numbers 287/269, dated June 1, 2005 (ACN ML050620034).

List of Affected Pages
3.1.4-2
3.2.4-1
3.2.4-3

B3.1.4-8
B3.2.4-3
B3.2.4-6

Applicable Regulatory Requirements/Criteria

Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

There is no regulatory requirement that specifies what remedial actions are to be taken when an LCO is not met. The proposed change makes the remedial actions taken when the Heat Flux Hot Channel Factor, $F_Q(Z)$, is not within its limit consistent with the LCO. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the Required Actions of Specification 3.1.4, "Rod Group Alignment Limits," and Specification 3.2.4, "Quadrant Power Tilt Ratio," to require measurement of both the steady state and transient portions of the Heat Flux Hot Channel Factor, $F_Q(Z)$. This change will ensure that the hot channel factors are within their limits when the rod alignment limits or quadrant power tilt ratio are not within their limits. The verification of hot channel factors is not an initiator of any accident previously evaluated. The verification that both the steady state and transient portion of $F_Q(Z)$ are within their limits will ensure this initial assumption of the accident analysis is met should a previously evaluated accident occur. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change revises the Required Actions in the Specifications for Rod Group Alignment Limits and Quadrant Power Tilt Ratio to require measurement of both the steady state and transient portions of the Heat Flux Hot Channel Factor, $F_Q(Z)$. This change is a correction that ensures that the plant conditions are as assumed in the accident analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.21 TSTF-340-A, Revision 3, "Allow 7 Day Completion Time for a Turbine-Driven AFW Pump Inoperable"

Description of Proposed Change
The proposed change revises Specification 3.7.5, "Auxiliary Feedwater System," to allow a 7 day Completion Time to restore an inoperable turbine-driven AFW pump in Mode 3 immediately following a refueling outage if Mode 2 has not been entered.

Differences Between the Proposed Change and the Approved Traveler
None

Summary of the Approved Traveler Justification
Present specifications have a 72 hour Completion Time for any inoperable Auxiliary Feedwater (AFW) pump with an Action to be in Mode 4 within 18 hours if the 72 hour Completion Time is not met. The proposed change would allow a 7 day Completion Time for the turbine-driven AFW pump if the inoperability occurs in Mode 3, immediately following a refueling outage, if Mode 2 has not been entered. This change will reduce the number of unnecessary Mode changes by providing added flexibility in Mode 3 to repair and test the turbine-driven AFW pump following a refueling outage. In the proposed condition, there is minimal decay heat due to the decay of the irradiated fuel during the refueling outage and the replacement of irradiated fuel with unirradiated fuel. The change is reasonable given the redundant capabilities afforded by the AFW system, the time needed to perform repairs and testing of the turbine-driven pump, and the low probability of an accident occurring during this time period that would require the operation of the turbine driven pump. In addition, there are alternate methods, such as feed and bleed, available to remove decay heat if necessary.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC documented their approval of TSTF-340-A, Revision 3, in a letter from William D. Beckner (NRC) to James Davis (NEI), dated March 16, 2000 (ACN ML003694199). An example of a plant-specific NRC approval of the changes in TSTF-340-A is Palo Verde Units 1, 2, and 3 Amendment Numbers 134/134/134 dated March 29, 2001 (ACN ML010930242).

List of Affected Pages
3.7.5-1
B3.7.5-5

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a

nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

There is no regulatory requirement that specifies what remedial actions are to be taken when an LCO is not met. The proposed change makes the remedial actions consistent with safety significance of the condition when a turbine-driven Auxiliary Feedwater pump is inoperable and the reactor core is in a low decay-heat state following a refueling outage. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431), Revision 3.1.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises Specification 3.7.5, "Auxiliary Feedwater (AFW) System," to allow a 7 day Completion Time to restore an inoperable AFW turbine-driven pump in Mode 3 immediately following a refueling outage, if Mode 2 has not been entered. An inoperable AFW turbine-driven pump is not an initiator of any accident previously evaluated. The ability of the plant to mitigate an accident is no different while in the extended Completion Time than during the existing Completion Time. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change revises Specification 3.7.5, "Auxiliary Feedwater (AFW) System," to allow a 7 day Completion Time to restore an inoperable turbine-driven AFW pump in Mode 3 immediately following a refueling outage if Mode 2 has not been entered. In Mode 3 immediately following a refueling outage, core decay heat is low and the need for AFW is also diminished. The two operable motor driven AFW pumps are available and there are alternate means of decay heat removal if needed. As a result, the risk presented by the extended Completion Time is minimal. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.22 TSTF-343, Revision 1, "Containment Structural Integrity"

Description of Proposed Change
The proposed change revises the Containment Leakage Rate Testing Program in TS Section 5.5, for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix).

The Containment Leakage Rate Testing Program description will be revised to add the following exception to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Testing Program,"

> The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

Differences Between the Proposed Change and the Approved Traveler
The Administrative Controls numbering in Vogtle TS Section 5.5 differs from the ISTS Administrative Controls numbering. Vogtle TS 5.5.17, "Containment Leakage Rate Testing Program," is equivalent to TS 5.5.16 in the ISTS. This has no effect on the requested change.

The changes identified in TSTF-343-A, Revision 1, for TS Section 5.5.6, "Containment Tendon Surveillance Program," and conforming changes to the Bases for SR 3.6.1.2 and the TS 3.6.1 Bases References, are not adopted. The changes in TSTF-343-A that affect this program are already reflected in the Vogtle Technical Specifications and Bases, and are therefore not necessary. Similarly changes to the Bases References for TS 3.6.1 and to SR 3.6.1.1 that are related to visual inspection of the steel liner plate are not adopted because they are already reflected in the Vogtle Bases.

Summary of the Approved Traveler Justification
On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code (the Code). The final rule, Subpart 50.55a(g)(6)(ii)(B) of Title 10 of the Code of Federal Regulations (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

The Vogtle containment consists of a prestressed, post-tensioned reinforced concrete structure with cylindrical walls, hemispherical roof and a reinforced concrete foundation basemat. The cylindrical portion of the containment is prestressed by a post-tensioning system composed of horizontal and vertical

tendons. A 1/4-in.-thick welded steel liner is attached to the inside face of the concrete.

TS Section 5.5.17 requirements for the Containment Leakage Rate Testing Program specify that the program shall be in accordance with the guidelines contained in RG 1.163. Regulatory Position C.3 of this regulatory guide states:

> Section 9.2.1, "Pretest Inspection and Test Methodology," of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval, for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration.

There are no specific requirements in NEI 94-01 for the visual examination except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.

In addition to the requirements of RG 1.163 and NEI 94-01, the steel liner plate inside containment must be visually examined in accordance with ASME Section XI Code, Subsection IWE. The frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year period. The steel liner plate visual examinations performed pursuant to Subsection IWE are performed during refueling outages since this is the only time that the liner plate is fully accessible.

The steel liner plate visual examinations performed pursuant to Subsection IWE are more rigorous than those performed pursuant to RG 1.163 and NEI 94-01. For example, Subarticle IWE-2320 requires the general visual examination to be the responsibility of an individual who is knowledgeable in the requirements for design, inservice inspection, and testing of Class MC and metallic liners of Class CC components. Subsection IWE, Subarticle-2330 requires the examination to be performed either directly or remotely, by an examiner with visual acuity sufficient to detect evidence of degradation. Furthermore, visual examinations of the liner plate must be reviewed by an Inspector employed by a State or municipality of the United States or an Inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance, in accordance with IWA 2110 and IWA 2120.

The combination of the Code requirements for the rigor of the visual examinations plus the third party review more than offsets the fact that fewer visual examinations of the concrete will be performed during a 10-year interval.

<u>Differences Between the Plant-Specific Justification and the Approved Traveler Justification</u>
None

<u>Licensee Commitments Required to Adopt this Change</u>
None

NRC Approval
The NRC documented their approval of TSTF-343-A, Revision 1, in a letter from Thomas H. Boyce (NRC) to the Technical Specification Task Force, dated December 6, 2005 (ACN ML053460302). An example of a plant-specific NRC approval of the changes in TSTF-340-A is Diablo Canyon, Units 1 and 2, Amendment Numbers 197/198, dated June 26, 2007 (ACN ML071370731).

List of Affected Pages
5.5-16

Applicable Regulatory Requirements/Criteria
Appendix A to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "General Design Criteria for Nuclear Power Plants," contains the following pertinent criterion:

Criterion 16, Containment Design, states:

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Appendix J to 10 CFR, Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," contains the following pertinent criterion:

Option B, Performance Based Requirements, states:

> A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leaktightness.

When implementing Option B of Appendix J to 10 CFR Part 50, "Performance-Based Leakage-Test Requirements," TS 5.5.17 states that the licensee's program shall be in accordance with the guidelines in RG 1.163, "Performance-Based Containment Leak-Test Program." Regulatory Position C.3 of RG 1.163 discusses visual examinations of accessible interior and exterior surfaces of the containment system. Specifically, Regulatory Position C.3 states, "examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years. . ."

In addition, Section 50.55a of 10 CFR Part 50 requires licensees to perform their containment ISI requirements in accordance with Subsection IWE of Section XI, Division I of the ASME Code. Paragraph 50.55a(g)(4) of 10 CFR requires licensees to update their containment ISI requirements in accordance with subsection IWE of Section XI, Division I, of the ASME Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(vii) and 10 CFR 50.55a(b)(2)(ix).

The proposed TS changes revise the containment leakage rate testing program for consistency with the requirements of 10 CFR 50.55a(g)(4) for components

classified as ASME Code CC. Specifically, TS Section 5.5.17, "Containment Leakage Rate Testing Program," is revised to allow the performance of visual examinations of the containment steel liner plate pursuant to ASME Code, Section XI, Subsection IWE, in lieu of the visual examinations performed pursuant to RG 1.163.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Significant Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the Technical Specifications (TS) Administrative Controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class CC. The proposed changes affect the frequency of visual examinations that will be performed for the steel containment liner plate for the purpose of the Containment Leakage Rate Testing Program.

   The frequency of visual examinations of the containment and the mode of operation during which those examinations are performed does not affect the initiation of any accident previously evaluated. The use of NRC approved methods and frequencies for performing the inspections will ensure the containment continues to perform the mitigating function assumed for accidents previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change revises the TS Administrative Controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class CC. The proposed change affects the frequency of visual examinations that will be performed for the steel containment liner plate for the purpose of the Containment Leakage Rate Testing Program.

The proposed changes do not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes revise the Technical Specifications (TS) Administrative Controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class CC. The proposed change affects the frequency of visual examinations that will be performed for the steel containment liner plate for the purpose of the Containment Leakage Rate Testing Program. The safety function of the containment as a fission product barrier will be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

2.23 TSTF-349-A, Revision 1, "Add Note to LCO 3.9.5 Allowing Shutdown Cooling Loops Removal from Operation"

Description of Proposed Change
The proposed change adds an LCO Note to LCO 3.9.6, "RHR and Coolant Circulation — Low Water Level," to allow the securing of the operating train of RHR for up to 15 minutes to support switching operating trains.

Differences Between the Proposed Change and the Approved Traveler
None.

Summary of the Approved Traveler Justification
The proposed change adds an LCO Note to LCO 3.9.6, "RHR and Coolant Circulation — Low Water Level," to allow the securing of the operating train of RHR to support switching operating trains. The allowance is acceptable because the allowed time frame is short and limitations are in place to ensure the RCS boron concentration is not reduced and to preclude draining activities. The proposed Note is consistent with the allowance LCO 3.4.8, "RCS Loops - MODE 5, Loops not filled." With the plant in MODE 6 with less than 23 feet of water above the Reactor Vessel flange, the reactor coolant system (RCS) is in an inventory status similar to LCO 3.4.8. Therefore, the allowances should also be consistent.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
None

Licensee Commitments Required to Adopt this Change
None

NRC Approval
The NRC did not issue a letter approving TSTF-349-A, Revision 1, however it was incorporated by the NRC into Revision 2 of the ISTS. An example of a plant-specific NRC approval of the changes in TSTF-349-A is Calvert Cliffs Units 1 and 2 Amendment Numbers 256/233 dated February 25, 2003 (ACN ML030560015).

List of Affected Pages
3.9.6-1
B3.9.6-2

Applicable Regulatory Requirements/Criteria
Title 10 of the Code of Federal Regulations (10 CFR), 10 CFR 50.36(c)(2), states:

> Limiting conditions for operation. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

There is no regulatory requirement that specifies what operational allowances should be included in the Technical Specifications. The proposed change makes

an operational allowance required to shift operating pumps. The allowance is consistent with the safety significance of the transitory condition and is consistent with similar LCOs in the Vogtle Technical Specifications. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

Signification Hazards Consideration

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change adds an LCO Note to LCO 3.9.6, "RHR and Coolant Circulation — Low Water Level," to allow securing the operating train of Residual Heat Removal (RHR) for up to 15 minutes to support switching operating trains. The allowance is restricted to conditions in which core outlet temperature is maintained at least 10 degrees F below the saturation temperature, when there are no draining operations, and when operations that could reduce the reactor coolant system (RCS) boron concentration are prohibited. Securing an RHR train to facilitate the changing of the operating train is not an initiator to any accident previously evaluated. The restrictions on the use of the allowance ensure that an RHR train will not be needed during the 15 minute period to mitigate any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change adds an LCO Note to LCO 3.9.6, "RHR and Coolant Circulation — Low Water Level," to allow securing the operating train of RHR to support switching operating trains. The allowance is restricted to conditions in which core outlet temperature is maintained at least 10 degrees F below the saturation temperature, when there are no draining operations, and when operations that could reduce the reactor coolant system (RCS) boron concentration are prohibited. With these restrictions, combined with the short time frame allowed to swap operating RHR trains and the ability to start an operating RHR train if needed, the occurrence of an event that would require immediate operation of an RHR train is extremely remote. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.0 Environmental Considerations

SNC has reviewed the proposed changes pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed Technical Specifications changes have no significant effect on the human environment and satisfy the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment.

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Adoption of Previously NRC-Approved Generic Technical Specification Changes

Enclosure 2

Marked-Up Technical Specifications Pages

Index of Affected Technical Specification Pages vs. Traveler Number

Page	Traveler(s)
1.1-6	TSTF-248-A
1.4-1	TSTF-284-A
1.4-4	TSTF-284-A
3.1.2-1	TSTF-142-A
3.1.4-2	TSTF-314-A
3.1.4-3	TSTF-110-A
3.1.6-3	TSTF-110-A
3.1.7-1	TSTF-234-A
3.1.7-2	TSTF-234-A
3.2.1-1	TSTF-95-A
3.2.1-2	TSTF-99-A
3.2.2-1	TSTF-95-A
3.2.3-1	TSTF-110-A
3.2.4-1	TSTF-314-A
3.2.4-3	TSTF-314-A
3.2.4-4	TSTF-110-A
3.3.4-1	TSTF-266-A
3.3.4-3	TSTF-266-A
3.4.2-1	TSTF-27-A
3.4.5-2	TSTF-87-A
3.4.9-1	TSTF-87-A
3.4.11-1	TSTF-247-A
3.4.11-2	TSTF-247-A
3.4.11-3	TSTF-247-A, TSTF-284-A
3.4.12-4	TSTF-284-A
3.4.16-1	TSTF-28-A
3.6.3-4	TSTF-45-A
3.6.3-5	TSTF-45-A, TSTF-46-A
3.7.5-1	TSTF-340-A
3.7.5-3	TSTF-245-A
3.8.3-3	TSTF-2-A
3.8.3-4	TSTF-2-A
3.9.1-1	TSTF-272-A
3.9.4-1	TSTF-312-A
3.9.4-2	TSTF-284-A
3.9.6-1	TSTF-349-A
5.5-3	TSTF-308-A
5.5-15	TSTF-273-A
5.5-16	TSTF-343-A

TSTF-248

However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation.

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

~~Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.~~

TSTF-284

INSERT - TS 1.4 Description

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

INSERT – TS 1.4 Description

TSTF-284

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.
Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3 FREQUENCY BASED ON A SPECIFIED CONDITION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency and the provisions of SR 3.0.3 would apply.

INSERT TS 1.4 Example 1.4-4

INSERT TS 1.4 Example 1.4-5

INSERT TS 1.4 Example 1.4-6

TSTF-284

Insert – TS 1.4 Example 1.4-4

TSTF-284

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be met in MODE 1. ----------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

Insert – TS 1.4 Example 1.4-5

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EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be performed in MODE 1. ----------	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

Insert – TS 1.4 Example 1.4-6

TSTF-284

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be met in MODE 3. ----------	
Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times. As described in Example 1.4-1. However. the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore. if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2). and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore. no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Measured core reactivity not within limit.	A.1	Reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	~~72 hours~~ 7 days TSTF-142
	AND		
	A.2	Establish appropriate operating restrictions and SRs.	~~72 hours~~ 7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND		
	B.3	Verify SDM is ≥ the limit specified in the COLR.	Once per 12 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	AND		
	B.5	Perform SR 3.2.2.1.	72 hours
	AND		
	B.6	Reevaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

TSTF-314

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	In accordance with the Surveillance Frequency Control Program ~~AND~~ ~~Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable~~ TSTF-110

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program ~~AND~~ ~~Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable~~ TSTF-110
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

Separate Condition entry is allowed for each inoperable rod position indicator and each inoperable demand position indicator.

ACTIONS

----------NOTE----------

~~Separate Condition entry is allowed for each group with no more than one inoperable rod position indicator in the group and for each bank with no more than one inoperable demand position indicator in the bank.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
~~B.~~ C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	~~B.1~~ C.1 Verify the position of the rods with inoperable DRPIs indirectly by using movable incore detectors.	8 hours
	OR	
		(continued)

Insert TS 3.1.7 - Condition B

TSTF-234

TS 3.1.7 – Condition B Insert

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Monitor and Record RCS T_{avg}.	Once per 1 hour
	AND		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors.	Once per 8 hours
	AND		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ~~B.~~ (continued)	C.2 ~~B.2~~ Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. ~~C.~~ One demand position indicator per bank inoperable for one or more banks.	D.1.1 ~~C.1.1~~ Verify by administrative means all DRPIs for the affected banks are OPERABLE.	Once per 8 hours
	AND	
	D.1.2 ~~C.1.2~~ Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours TSTF-234
	OR	
	D.2 ~~C.2~~ Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
E. ~~D.~~ Required Action and associated Completion Time not met.	E.1 ~~D.1~~ Be in MODE 3.	6 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

LCO 3.2.1 $F_Q(Z)$ shall be within the steady state and transient limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q(Z)$ not within steady state limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q(Z)$ exceeds steady state limit.	15 minutes
	AND		
	A.2	Reduce Power Range Neutron Flux — High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds steady state limit.	~~8~~ 72 hours
	AND		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds steady state limit.	72 hours
	AND		
	A.4	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

TSTF-095

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limits.	A.1.1 Restore $F^N_{\Delta H}$ to within limits.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce Power Range Neutron Flux — High trip setpoints to ≤ 55% RTP.	~~8~~ 72 hours
	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

TSTF-095

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD shall be maintained within the limits specified in the COLR.

----------NOTE----------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	In accordance with the Surveillance Frequency Control Program ~~AND~~ ~~Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable~~ TSTF-110

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.6 must be completed whenever Required Action A.5 is implemented. ---------- QPTR not within limit.	A.1	Limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00.	2 hours
	AND		
	A.2.1	Perform SR 3.2.4.1.	Once per 12 hours
	AND		
	A.2.2	Limit THERMAL POWER to ≥ 3% below RTP for each 1% QPTR > 1.00.	----------NOTE---------- For performances of Required Action A.2.2 the Completion Time is measured from the completion of SR 3.2.4.1. ---------- 2 hours
	AND		
	A.3	Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Actions A.1 and A.2.2

TSTF-314

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6 ----------NOTE---------- Perform Required Action A.6 only after Required Action A.5 is completed. ---------- Perform SR 3.2.1.1 and SR 3.2.2.1. , SR 3.2.1.2,	----------NOTE--------- Only one of the following Completion Times, whichever becomes applicable first, must be met. ---------- Within 24 hours after reaching RTP OR Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

TSTF-314

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- With one power range channel inoperable, the remaining three power range channels can be used for calculating QPTR. ----------	
	Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program ~~AND~~ ~~Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable~~ TSTF-110
SR 3.2.4.2	----------NOTE---------- Only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER ≥ 75% RTP. ----------	
	Confirm that the normalized symmetric power distribution is consistent with QPTR.	Once within 12 hours AND In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Functions ~~in Table 3.3.4-1~~ shall be OPERABLE. TSTF-266

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

Removed from TS and placed in Bases as Table B 3.3.4-1

Table 3.3.4-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF CHANNELS
MONITORING INSTRUMENTATION	
1. Source Range Neutron Flux	1
2. Extended Range Neutron Flux	1
3. RCS Cold Leg Temperature	1/loop
4. RCS Hot Leg Temperature	2
5. Core Exit Thermocouples	2
6. RCS Wide Range Pressure	2
7. Steam Generator Level Wide Range	1/loop
8. Pressurizer Level	2
9. RWST Level	1[a]
10. BAST level	1[a]
11. CST Level	1/tank[a] [c]
12. Auxiliary Feedwater Flow	1/loop
13. Steam Generator Pressure	1/loop
TRANSFER AND CONTROL CIRCUITS	
1. Reactivity Control	(b)
2. RCS Pressure Control	(b)
3. Decay Heat Removal	
a. Auxiliary Feedwater	(b)
b. Steam Generator Atmospheric Relief Valve[d]	(b)
4. RCS Inventory/Charging System	(b)
5. Safety support systems required for the above functions	(b)

(a) Alternate local level indication may be established to fulfill the required number of channels.

(b) The required channels include the transfer switches and control circuits necessary to place and maintain the unit in a safe shutdown condition using safety grade components.

(c) Only required for the OPERABLE tank.

(d) Refer also to LCO 3.7.4.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ 551°F.	~~Once within 30 minutes and every 30 minutes thereafter when the T_{avg} - T_{ref} deviation alarm is not reset and any RCS loop T_{avg} < 561°F~~ In accordance with the Surveillance Frequency Control Program

TSTF-027

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One required RCS loop not in operation, ~~and reactor trip breakers closed and~~ Rod Control System capable of rod withdrawal. with	C.1	Restore required RCS loop to operation.	1 hour
	OR		Place the Rod Control System in a condition incapable of rod withdrawal.
	C.2	~~De-energize all control rod drive mechanisms (CRDMs).~~	1 hour TSTF-087
D. Two required RCS loops inoperable.	D.1	~~De-energize all CRDMs.~~	Immediately
	AND		
OR	D.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
No RCS loop in operation.	AND		
	D.3	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 92%; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 ~~with reactor trip breakers open~~.	6 hours
TS 3.4.9 - Condition A Insert	AND	
A.4	~~A.2~~ Be in MODE 4.	12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4.	12 hours

TSTF-087

TS 3.4.9 – Condition A Insert

TSTF-087

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
	.		
	.		
	.		
	A.2	Fully insert all rods.	6 hours
	AND		
	A.3	Place Rod Control System in a condition incapable of rod withdrawal.	6 hours
	AND		
	.		
	.		
	.		

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

TSTF-247

---NOTE---
Separate Condition entry is allowed for each PORV and each block valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve.	1 hour
	AND	
	B.2 Remove power from associated block valve.	1 hour
	AND	
	B.3 Restore PORV to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	C.1	Place associated PORV in manual control.	1 hour
	AND		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable and not capable of being manually cycled.	E.1	Close associated block valves.	1 hour
	AND		
	E.2	Remove power from associated block valves.	1 hour
	AND		
	E.3	Be in MODE 3.	6 hours
	AND		
	E.4	Be in MODE 4.	12 hours
F. ~~More than one~~ Two block ~~valve~~ valves inoperable.	~~F.1~~	~~Place associated PORVs in manual control.~~	~~1 hour~~
	~~AND~~		

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(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.1 ~~F.2~~ Restore one block valve to OPERABLE status.	2 hours
	~~AND~~	
	~~F.3 Restore remaining block valve to OPERABLE status.~~	~~72 hours~~
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3.	6 hours
	AND	
	G.2 Be in MODE 4.	12 hours

TSTF-247

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	~~NOTE~~ Not required to be performed with block valve closed in accordance with the Required ~~Action~~ of ~~Conditions A, B, or E~~ of this LCO. 2. Only required to be performed in MODES 1 and 2. Perform a complete cycle of each block valve. NOTE Only required to be performed in MODES 1 and 2.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program

1. NOTES Actions

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify both safety injection pumps are incapable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify each accumulator is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	Verify RHR suction valves are open for each required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	----------NOTE---------- met Only required to be ~~performed~~ when complying with LCO 3.4.12.b. ---------- Verify RCS vent size within specified limits.	TSTF-284 In accordance with the Surveillance Frequency Control Program

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

----------NOTE----------
LCO 3.0.4c is applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	~~B.1 Perform SR 3.4.16.2.~~	~~4 hours~~
	~~AND~~	
	~~B.2~~ B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

TSTF-028

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 24 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition C of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each 14 inch purge valve is closed, except when the associated penetration(s) is (are) permitted to be open for purge or venting operations and purge system surveillance and maintenance testing under administrative control.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program

TSTF-045

and not locked, sealed, or otherwise secured

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	----------NOTES---------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. The fuel transfer tube blind flange is only required to be verified closed once after refueling prior to entering MODE 4 from MODE 5. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days and not locked, sealed, or otherwise secured TSTF-045
SR 3.6.3.5	Verify the isolation time of each ~~power operated and each~~ automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program TSTF-046
SR 3.6.3.6	Perform leakage rate testing for containment purge valves with resilient seals.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
LCO 3.0.4b is not applicable.
---

affected equipment

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore ~~steam supply~~ to OPERABLE status.	7 days
	INSERT - TS 3.7.5 Condition A	
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours

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(continued)

Insert - TS 3.7.5 Condition A

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OR ----------NOTE---------- Only applicable if MODE 2 has not been entered following refueling. ---------- One turbine driven AFW pump inoperable in MODE 3 following refueling.		

SURVEILLANCE REQUIREMENTS

Insert TS 3.7.5 - SR Note 1

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.3	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	~~----------NOTE----------~~ ~~Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator.~~ ~~----------~~ Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

Insert TS 3.7.5 - SR Note 1

Insert TS 3.7.5 - SR Note 2

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(continued)

TS 3.7.5 – SR Note Inserts

TSTF-245

SR Note 1

---------- NOTE ----------
AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.

SR Note 2

---------- NOTES ----------
1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator.

2. AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 68,000 gal of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify lube oil inventory is ≥ 336 gal.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG has one air start receiver with a pressure ≥ 210 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.6	Verify each DG ventilation supply fan starts and the necessary dampers actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program

~~(continued)~~ TSTF-002

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~SR 3.8.3.7~~ ~~NOTE~~ ~~Not required to be performed when DG is required OPERABLE in accordance with Specification 3.8.2.~~ ~~For each fuel oil storage tank:~~ ~~a. Drain the fuel oil;~~ ~~b. Remove the sediment; and~~ ~~c. Clean the tank.~~	~~In accordance with the Surveillance Frequency Control Program~~

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR. TSTF-272

APPLICABILITY: MODE 6.

---------- NOTE ----------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
	AND		
	A.3	Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

INSERT - LCO 3.9.4 Note

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APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

INSERT – LCO 3.9.4 Note

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---------- NOTE ----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.2	----------NOTE---------- ~~Only required for unisolated penetrations.~~ ---------- Verify at least two containment ventilation valves in each open containment ventilation penetration providing direct access from the containment atmosphere to the outside atmosphere are capable of being closed from the control room.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.3	----------NOTE---------- Only required for an open equipment hatch. ---------- Verify the capability to install the equipment hatch.	In accordance with the Surveillance Frequency Control Program

Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.4.c.1.

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3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

NOTES

1. ~~NOTE~~

INSERT - TS 3.9.6 Note 2

One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

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APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1	Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	OR		
	A.2	Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
			(continued)

INSERT – TS 3.9.6 Note 2

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2. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS water volume are permitted.

5.5 Programs and Manuals

5.5.4 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:

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

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentrations stated in 10 CFR 20, Appendix B (to paragraphs 20.1001-20.2401), Table 2, Column 2;

c. 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;

d. 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;

e. ~~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~~; TSTF-308

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that

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.

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

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b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s),

When a loss of safety function is caused by inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

(continued)

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5.5 Programs and Manuals (continued)

5.5.16 MS and FW Piping Inspection Program

This program shall provide for the inspection of the four Main Steam and Feedwater lines from the containment penetration flued head outboard welds, up to the first five-way restraint. The extent of the inservice examinations completed during each inspection interval (ASME Code Section XI) shall provide 100% volumetric examination of circumferential and longitudinal welds to the extent practical. This augmented inservice inspection is consistent with the requirements of NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," November 1975 and Section 6.6 of the FSAR.

5.5.17 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, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995, as modified by the following exceptions:

1. Leakage rate testing for containment purge valves with resilient seals is performed once per 18 months in accordance with LCO 3.6.3, SR 3.6.3.6 and SR 3.0.2.

2. Containment personnel air lock door seals will be tested prior to reestablishing containment integrity when the air lock has been used for containment entry. When containment integrity is required and the air lock has been used for containment entry, door seals will be tested at least once per 30 days during the period that containment entry(ies) is (are) being made.

3. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief or alternative has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

INSERT - Section 5.5.17

(continued)

INSERT - Section 5.5.17

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4. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Adoption of Previously NRC-Approved Generic Technical Specification Changes

Enclosure 3

Example Marked-Up Technical Specifications Bases Pages

Index of Affected Technical Specification Bases Pages vs. Traveler Number

Page	Traveler(s)
B3.0-9	TSTF-273-A
B3.1.2-5	TSTF-142-A
B3.1.4-8	TSTF-314-A
B3.1.6-6	TSTF-110-A
B3.1.7-4	TSTF-234-A
B3.1.7-5	TSTF-234-A
B3.1.7-6	TSTF-234-A
B3.2.1-5	TSTF-95-A
B3.2.1-6	TSTF-99-A
B3.2.2-5	TSTF-95-A
B3.2.3-1	TSTF-110-A
B3.2.3-4	TSTF-110-A
B3.2.4-3	TSTF-314-A
B3.2.4-6	TSTF-314-A
B3.2.4-7	TSTF-110-A
B3.3.4-2	TSTF-266-A
B3.3.4-3	TSTF-266-A
B3.3.4-5	TSTF-266-A
B3.3.4-6	TSTF-266-A
B3.3.4-7	TSTF-266-A
B3.4.2-3	TSTF-27-A
B3.4.5-1	TSTF-87-A
B3.4.5-2	TSTF-87-A
B3.4.5-3	TSTF-87-A
B3.4.5-4	TSTF-87-A
B3.4.5-5	TSTF-87-A
B3.4.9-3	TSTF-87-A
B3.4.11-4	TSTF-247-A
B3.4.11-6	TSTF-247-A
B3.4.11-7	TSTF-284-A
B3.4.12-13	TSTF-284-A
B3.4.16-4	TSTF-28-A
B3.6.3-10	TSTF-45-A
B3.6.3-11	TSTF-45-A
B3.6.3-11	TSTF-46-A
B3.7.5-5	TSTF-340-A
B3.7.5-7	TSTF-245-A
B3.7.5-7	TSTF-245-A
B3.7.5-8	TSTF-245-A
B3.8.3-13	TSTF-2-A
B3.8.3-14	TSTF-2-A
B3.9.1-3	TSTF-272-A
B3.9.1-4	TSTF-272-A
B3.9.4-5	TSTF-312-A

Page	Traveler(s)
B3.9.4-7	TSTF-284-A,
B3.9.6-2	TSTF-349-A

BASES

LCO 3.0.6 (continued)

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

INSERT - LCO 3.0.6 Bases

TSTF-273

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to

(continued)

INSERT – LCO 3.0.6 Bases

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

BASES

ACTIONS

A.1 and A.2 (continued)

to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of ~~72 hours~~ 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined. TSTF-142

The required Completion Time of ~~72 hours~~ 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by

(continued)

BASES

ACTIONS

B.2, B.3, B.4, B.5, and B.6 (continued)

A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

INSERT - Bases 3.1.4 Action

TSTF-314

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

The following accident analyses require reevaluation for continued operation with a misaligned rod.

RCCA Insertion Characteristics
RCCA Misalignment
Decrease in Reactor Coolant Inventory

- Inadvertent Opening of a Pressurizer Safety or Relief Valve
- Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary That Penetrates Containment
- Loss-of-Coolant-Accidents

Increase in Heat Removal by the Secondary System (Steam System Piping Rupture) Spectrum of RCCA Ejection Accidents.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least

(continued)

INSERT – BASES 3.1.4 Action

, as approximated by the steady state and transient $F_Q(Z)$,

TSTF-314

BASES

ACTIONS

C.1 (continued)

full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

Among the factors that impact the estimated critical position (ECP) is Xenon concentration, which varies with time, either increasing or decreasing depending on the amount of time since the trip occurred. The 4 hour limit within which the ECP must be verified within the insertion limits ensures that changes in Xenon concentration will be limited and, hence, it ensures that criticality will not occur with control rods outside of the insertion limits due to Xenon decay.

SR 3.1.6.2

the specified

TSTF-110

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency ~~of 4 hours~~ is sufficient to detect control banks that may be approaching the insertion limits.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. This surveillance is accomplished from the control room by verifying via the

(continued)

BASES

APPLICABILITY (continued)

in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each ~~group with no more than one~~ inoperable rod position indicator ~~in the group~~ and ~~for~~ each ~~bank with no more than one~~ inoperable demand position indicator ~~in the bank~~. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, F_{deltaH} satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the non-indicating rods have not been moved.

TSTF-234

A.1

When one DRPI channel per group fails, the position of the rod may still be determined indirectly by use of the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of ~~B.1 or B.2~~ C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

Insert Bases 3.1.7 - Condition B

(continued)

Insert Bases 3.1.7 - Condition B

B.1, B.2, B.3 and B.4

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion will not occur. Together with the indirect position determination available via movable incore detectors will minimize the potential for rod misalignment.

The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

The position of the rods may be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that $F_Q(Z)$ satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the non-indicating rods have not been moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication (Ref. 4).

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

BASES

ACTIONS (continued)

C.1 and C.2

~~B.1 and B.2~~

These Required Actions ensure that when one or more rods with inoperable digital rod position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, prompt action is taken to begin verifying that these rods are still properly positioned, relative to their group positions.

Either the rod positions must be determined within 8 hours, or THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions using the moveable incore detectors.

D.1.1 and D.1.2

~~C.1.1 and C.1.2~~

TSTF-234

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This verification can be an examination of logs, administrative controls, or other information that all DRPIs in the affected bank are OPERABLE.

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position will not cause core peaking to approach core peaking factor limits. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions ~~C.1.1 and C.1.2~~ or reduce power to ≤ 50% RTP.

E.1

~~D.1~~

D.1.1 and D.1.2

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve

(continued)

TSTF-234

BASES

ACTIONS

~~D.1~~ E.1 (continued)

this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
2. FSAR, Chapter 15.

BASES

LCO (continued)

operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q(Z)$ exceeds its steady state limit, maintains an acceptable absolute power density. $F_Q(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

A reduction of the Power Range Neutron Flux—High trip setpoints by ≥ 1% of RTP for each 1% by which $F_Q(Z)$ exceeds its steady state limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. TSTF-095

A.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by ≥ 1% (in %RTP) for each 1% by which $F_Q(Z)$ exceeds its limit, is a conservative

(continued)

BASES

ACTIONS

A.1.1 (continued)

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux — High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive. TSTF-095

72

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.2

Once corrective action has been taken in accordance with Required Action A.1.1 or A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

TSTF-110

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

(continued)

BASES

ACTIONS

A.1 (continued)

the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

~~The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.~~

TSTF-110

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits ~~and is consistent with the status of the AFD monitor alarm~~. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored.~~

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10216(NP), June 1983.

BASES (continued)

ACTIONS

A.1

With the QPTR exceeding its limit, limiting THERMAL POWER to ≥ 3% below RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2.1 and A.2.2

Because the QPTR alarm is already in its alarmed state, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly within the following 2 hours. A Note clarifies that the Completion Time of Required Action A.2.2 begins after Required Action A.2.1 is complete. These Completion Times are sufficient because any additional change in QPTR would be relatively slow.

A.3

INSERT - Bases 3.2.4 Action

TSTF-314

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Action A.1 or A.2.2 ensures that these primary indicators of power distribution are within their respective limits. The above Completion Time takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and

(continued)

INSERT – BASES 3.2.4 Action

, as approximated by the steady state and transient $F_Q(Z)$,	TSTF-314

BASES

ACTIONS

A.6 (continued)

INSERT - Bases 3.2.4 Action

TSTF-314

power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1 and A.2.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show QPTR = 1.00 (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show QPTR = 1.00 and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by a Note that allows QPTR to be calculated with three power range channels if one power range channel is inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Valid inputs to the detector current comparator from the upper and lower sections from 3 or 4 power range channels are required for the QPTR alarm to be OPERABLE.

~~When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for~~ Those causes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt. TSTF-110

For

SR 3.2.4.2

This Surveillance is modified by a Note, which states that the surveillance is only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER ≥75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

When one power range channel is inoperable, the incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Remote Shutdown System is considered an important contributor to the reduction of unit risk to accidents and as such it satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

LCO

The Remote Shutdown System LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table B 3.3.4-1. ~~Table 3.3.4-1 in the accompanying LCO.~~

The controls, instrumentation, and transfer switches are required for:

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves or an SG ARV on at least one SG; TSTF-266
- RCS inventory control via charging flow; and
- Safety support systems for the above Functions.

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the Remote Shutdown System Function are OPERABLE. In some cases, Table B 3.3.4-1 ~~Table 3.3.4-1~~ may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long as one channel of any of the alternate information or control sources is OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

(continued)

BASES (continued)

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

A Remote Shutdown System division is inoperable when each function is not accomplished by at least one designated Remote Shutdown System channel that satisfies the OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each ~~Function listed on Table 3.3.4-1.~~ Function. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

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A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System are inoperable. This includes ~~any Function listed in Table 3.3.4-1, as well as the transfer switches and control circuits.~~ the control and transfer switches for any required function. A required Function is considered to be inoperable if one or more of its required channels is inoperable.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2

SR 3.3.4.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The surveillance may be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Any change in the scope or frequency of this SR requires reevaluation of STI Evaluation number 417332, in accordance with the Surveillance Frequency Control Program.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. STI Evaluation 417332.

INSERT - TABLE B 3.3.4-1 →

TSTF-266

(continued)

INSERT – TABLE B 3.3.4-1

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Table B 3.3.4-1 (page 1 of 1)

Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF CHANNELS
MONITORING INSTRUMENTATION	
1. Source Range Neutron Flux	1
2. Extended Range Neutron Flux	1
3. RCS Cold Leg Temperature	1/loop
4. RCS Hot Leg Temperature	2
5. Core Exit Thermocouples	2
6. RCS Wide Range Pressure	2
7. Steam Generator Level Wide Range	1/loop
8. Pressurizer Level	2
9. RWST Level	1[(a)]
10. BAST level	1[(a)]
11. CST Level	1/tank[(a) (c)]
12. Auxiliary Feedwater Flow	1/loop
13. Steam Generator Pressure	1/loop
TRANSFER AND CONTROL CIRCUITS	
1. Reactivity Control	(b)
2. RCS Pressure Control	(b)
3. Decay Heat Removal	
a. Auxiliary Feedwater	(b)
b. Steam Generator Atmospheric Relief Valve[(d)]	(b)
4. RCS Inventory/Charging System	(b)
5. Safety support systems required for the above functions	(b)

(a) Alternate local level indication may be established to fulfill the required number of channels.

(b) The required channels include the transfer switches and control circuits necessary to place and maintain the unit in a safe shutdown condition using safety grade components.

(c) Only required for the OPERABLE tank.

(d) Refer also to LCO 3.7.4.

BASES

APPLICABILITY (continued)

it is necessary to allow RCS loop average temperatures to fall below the HZP temperature, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

INSERT - SR 3.4.2.1 Bases

~~RCS loop average temperature is required to be verified at or above 551°F when the T_{avg} - T_{ref} deviation alarm (TI-0412, TI-0422, TI-0432, TI-0442) is not reset and any RCS loop T_{avg} < 561°F. When these conditions are present, RCS loop average temperatures could fall below the LCO requirement without additional warning. The frequency of 30 minutes is sufficient to prevent the inadvertent violation of the LCO. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

TSTF-027

REFERENCES

1. FSAR, Section 4.3 and Subsections 15.0.3 and 15.4.8.

INSERT - SR 3.4.2.1 Bases

TSTF-027

RCS loop average temperature is required to be periodically verified at or above 551°F. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops — MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with ~~RTBs in the closed position and~~ the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are

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(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops — MODE 3 satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the ~~RTBs in the closed position and~~ Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the ~~RTBs closed and~~ Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents. TSTF-087

~~With the RTBs in the open position, or the CRDMs de-energized,~~ When the Rod Control System is not capable of rod withdrawal ~~therefore,~~ only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure adequate decay heat removal capability.

The Note permits all RCPs to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a

(continued)

BASES

LCO (continued)

change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with ~~RTBs in the closed position.~~ the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the ~~RTBs open.~~ Rod Control System not capable of rod withdrawal.

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Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.6, "RCS Loops — MODE 4";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

(continued)

BASES (continued)

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

is

)

place the Rod Control System in a condition incapable of rod withdrawal (e.g.,

If the required RCS loop is not in operation, and the ~~RTBs are closed~~ ~~and~~ Rod Control System capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets. When the ~~RTBs are in the closed position~~ ~~and~~ Rod Control System capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the ~~RTBs must be opened.~~ The Completion Times of 1 hour to restore the required RCS loop to operation or ~~de-energize all~~ ~~CRDMs~~ is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

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is

Rod Control System must be rendered incapable of rod withdrawal.

defeat the Rod Control System

(continued)

BASES

ACTIONS (continued)

D.1, D.2, and D.3

place the Rod Control System in a condition incapable of rod withdrawal (e.g.,

If two required RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

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SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification that the required loops are in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water level (LI-0501, LI-0502, LI-0503, LI-0504) for the required RCS loops is above the highest point of the steam generator U-tubes for each required loop. To assure that the steam generator is capable of functioning as a heat sink for the removal of decay heat, the U-tubes must be completely submerged. Plant procedures provide the minimum indicated levels for the range of the steam generator operating conditions required to satisfy this SR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

LCO (continued)

groups of pressurizer heaters onto the non-Class 1E emergency buses. These non-Class 1E emergency buses are in turn fed from the Class 1E 4160-V buses which can in turn be supplied from the emergency diesel generators or offsite power sources. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1, A.2, A.3 and A.4

~~A.1 and A.2~~

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level — High Trip.

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within 6 hours

all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought

If the pressurizer water level is not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with ~~the reactor trip breakers open, within 6 hours and~~ to MODE 4 within 12 hours. This takes the unit out of the applicable MODES

(continued)

BASES

APPLICABILITY (continued)

requirements in MODES 4, 5, and 6 with the reactor vessel head in place.

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ACTIONS

A Note has been added to clarify that all pressurizer PORVs and block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis).

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage, instrumentation problems, or other causes that do not create a possibility for a small break LOCA). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. The PORVs may be considered OPERABLE in either the manual or automatic mode. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to

(continued)

BASES

ACTIONS (continued)

E.1, E.2, E.3, and E.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two PORVs inoperable. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, 5, and 6, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

F.1, F.2, and F.3

two block valves are

If ~~more than one block valve is~~ inoperable, it is necessary to ~~either~~ restore ~~the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and~~ restore at least one block valve within 2 hours ~~and restore the remaining block valve within 72 hours~~. The Completion ~~Times are~~ reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

Time is

TSTF-247

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, 5, and 6, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Note modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Actions of Conditions A, B, or E.~~

INSERT - Bases SR 3.4.11.1

SR 3.4.11.2

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SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

INSERT - Bases SR 3.4.11.2

REFERENCES

1. Regulatory Guide 1.32, February 1977.

INSERT - Bases 3.4.11 Reference

INSERT – Bases SR 3.4.11.1

The SR has two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Actions of this LCO. Note 2 this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 2, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

INSERT – Bases SR 3.4.11.2

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 2, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

Insert – Bases 3.4.11 Reference

2. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f),"June 25, 1990.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be ~~performed~~ met if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12 b. TSTF-284

SR 3.4.12.5

The PORV block valve must be verified open to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ the COPS arming temperature specified in the PTLR on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ the COPS arming temperature specified in the PTLR. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES (continued)

ACTIONS

A Note permits the use of the provisions of LCO 3.0.4c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at or proceeds to power operation.

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is acceptable because of the low probability of an SGTR accident occurring during this period.

B.1 ~~and B.2~~

the unit must be placed in a MODE in which the requirement does not apply.

With the gross specific activity in excess of the allowed limit, ~~an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.~~

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The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

and not locked, sealed, or otherwise secured

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those Containment Isolation valves outside containment and capable of being mispositioned are in the correct position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The SR specifies that Containment Isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these Containment Isolation valves, once they have been verified to be in the proper position, is small.

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SR 3.6.3.4

and not locked, sealed, or otherwise secured

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For Containment Isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these Containment Isolation valves are operated under administrative controls and the probability of their

(continued)

This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

misalignment is low. The SR specifies that valves that are open under administrative controls are not required to meet the SR during the time they are open.

Note 1 allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these Containment Isolation valves, once they have been verified to be in their proper position, is small.

Note 2 modifies the requirement to verify the blind flange on the fuel transfer canal. This blind flange is only required to be verified closed after the completion of refueling activities when the flange has been replaced for MODE 4 entry and no more fuel transfers between the fuel handling building and containment will occur. The flange is only removed to support refueling operations and once replaced is not removed again until the next refueling. Since the removal of this flange is limited to refueling operations, and access to it is restricted during MODES 1, 2, 3, and 4, the probability of it being mispositioned between refuelings is small. Therefore, it is reasonable that it be verified once upon completion of refueling activities prior to entering MODE 4 from MODE 5. TSTF-046

SR 3.6.3.5

power operated

Verifying that the isolation time of each ~~power operated and~~ automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program. Any change in the scope or frequency of this SR requires reevaluation of STI Evaluation number 417332, in accordance with the Surveillance Frequency Control Program.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost.

In MODE 4 the AFW System may be used for heat removal via the steam generators, but is not required since the RHR System is available in this MODE.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4b to an inoperable AFW train. There is an increased risk associated with an AFW train inoperable and the provisions of LCO 3.0.4b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

or if turbine driven pump is inoperable while in MODE 3 immediately following refueling,

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

TSTF-340

the inoperable equipment to an

INSERT - TS 3.7.5 Bases Action A

~~a. The redundant OPERABLE steam supply to the turbine driven AFW pump;~~

~~b. The availability of redundant OPERABLE motor driven AFW pumps; and~~

~~c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.~~

B.1

With one of the required AFW trains (pump or flow path) inoperable for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the

(continued)

Insert - TS 3.7.5 Bases Action A

TSTF-340

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion time is reasonable since there is a redundant steam supply line for the turbine driven pump.

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling, the 7 day Completion time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps; and due to the low probability of an event requiring the use of the turbine driven AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. The correct position is the position of the valves necessary to support the operational needs of the plant at that time, including during low power operation and surveillance testing, provided that the requirements of the Technical Specification safety analysis are met. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

TSTF-245

INSERT: Bases SR 3.7.5.1 NOTE

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.5.2

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 Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. 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. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The 31 day frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Ref. 2.

In addition to the acceptance criteria of the Inservice Testing Program, performance of this SR also verifies that pump performance is greater than or equal to the performance assumed in the safety analysis.

(continued)

INSERT: Bases SR 3.7.5.1 Note

TSTF-245

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.2 (continued)

This SR is modified by a Note allowing the SR to be deferred until suitable test conditions are established. This deferral may be required because there may be insufficient steam pressure to perform the test.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. However, for the turbine driven AFW train this SR may be performed in conjunction with ASME Section XI full flow check valve testing which must be performed when steam is available to run the turbine driven AFW pump.

INSERT: Bases SR 3.7.5.3 NOTE

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. However, for the turbine driven AFW train this SR must be performed when steam is available to run the pump.

TSTF-245

This SR is modified by ~~a Note allowing~~ two Notes. The first Note allows the SR to be deferred until suitable test conditions are established. This deferral may be required because there may be insufficient steam pressure to perform the test.

INSERT: Bases SR 3.7.5.4 NOTE

(continued)

INSERT: Bases SR 3.7.5.3 Note

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

INSERT: Bases SR 3.7.5.4 Note

The second Note states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.5 (continued)

regarding the watertight integrity of the fuel oil system. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.8.3.6

This surveillance demonstrates that each DG ventilation supply fan starts automatically and the necessary dampers actuate to the correct position on a simulated or actual actuation signal. The two fans in each DG building and associated dampers start and actuate on different signals. Fans 1/2-1566-B7-001 (train A) and 1/2-1566-B7-002 (train B) start automatically and the necessary intake and discharge dampers actuate to the correct position on a train associated DG running signal and fans 1/2-1566-B7-003 and 1/2-1566-B7-004 start automatically and the necessary intake and discharge dampers actuate to the correct position on high DG building temperature signal coincident with a DG running signal. TSTF-002

~~SR 3.8.3.7~~

~~Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required by Regulatory Guide 1.137 (Ref. 2), paragraph 2.f. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR, provided that accumulated sediment is removed during performance of the Surveillance.~~

~~While this SR is being performed, the requirement for sufficient fuel oil to support ≥ 7 days of operation may be met by alternate means as discussed in FSAR section 9.5.4.2.2.~~

(continued)

BASES

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.8.3.7 (continued)~~

~~The SR is modified by a Note that excepts the performance of this SR when the associated DG is required OPERABLE by LCO 3.8.2. This exception is consistent with the SR performance exceptions in LCO 3.8.2 for SRs that might impact the OPERABILITY of the DGs.~~

TSTF-002

REFERENCES

1. FSAR, Paragraph 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.
6. ASTM Standards: D4057-06; D1298-06; D4176-04; D2709-96; D1552-07; D2622-07; D4294-08a; D5452-08.
7. ASTM Standards, D975-07.
8. Southern Company Services Calculation number X4C2403V08, Standby Diesel Generator Fuel Oil Consumption and Storage Tank Capacity.
9. Southern Company Services Calculation numbers X4C2403V11 and X4C2403V12, Emergency Diesel Generator Lube Oil Inventory Technical Specification Values.
10. Southern Company Services Calculation number X4C2403V09, Emergency Diesel Generator Starting Air Pressure Technical Specification Value.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in all filled portions of the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. In MODES 1 and 2, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensure an adequate amount of negative reactivity is available to shut down the reactor. In MODES 3, 4, and 5, LCO 3.1.1, "SHUTDOWN MARGIN" ensures an adequate amount of negative reactivity is available to shut down the reactor.

TSTF-272

ACTIONS

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the Reactor Coolant System. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the filled portions of the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

(continued)

BASES

ACTIONS (continued)

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

There are no safety analysis assumptions of boration flow rate and concentration that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in all filled portions of the RCS, the refueling canal and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Subsection 15.4.6.

BASES

LCO (continued)

action to close containment penetrations to minimize potential offsite doses. The LCO requirements for penetration closure may also be met by the automatic isolation capability of the CVI system. Temporary non-1E power may be supplied to the air operated and/or solenoid operated CVI valves. The temporary non-1E power must be connected in such a way that it cannot affect the capability of the valves to close either automatically or manually from the control room handswitch.

INSERT - Bases LCO 3.9.4 Note

TSTF-312

Item b of this LCO includes requirements for both the emergency air lock and the personnel air lock. The personnel and emergency air locks are required by Item b of this LCO to be isolable by at least one air lock door in each air lock. Both containment personnel and emergency air lock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided at least one air lock door is isolable in each air lock. An air lock is isolable when the following criteria are satisfied:

1. one air lock door is OPERABLE,

2. at least 23 feet of water shall be maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated individual is available to close the door.

OPERABILITY of a containment air lock door requires that the door seal protectors are easily removed, that no cables or hoses are being run through the air lock, and that the air lock door is capable of being quickly closed.

The equipment hatch is considered isolable when the following criteria are satisfied:

1. the necessary equipment required to close the hatch is available.

2. at least 23 feet of water is maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated trained hatch closure crew is available.

Similar to the air locks, the equipment hatch opening must be capable of being cleared of any obstruction so that closure can be achieved as soon as possible.

(continued)

INSERT – Bases LCO 3.9.4 Note

TSTF-312

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note stating that this surveillance is not required to be met for valves in isolated penetrations. LCO 3.9.4.c.1 provides the option to close penetrations in lieu of requiring automatic actuation capability.

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve in each open containment ventilation penetration actuates to its isolation position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

TSTF-284

SR 3.9.4.3

The equipment hatch is provided with a set of hardware, tools, and equipment for moving the hatch from its storage location and installing it in the opening. The required set of hardware, tools, and equipment shall be inspected to ensure that they can perform the required functions.

The 7 day frequency is adequate considering that the hardware, tools, and equipment are dedicated to the equipment hatch and not used for any other functions.

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in its opening, the availability of the means to install the hatch is not required.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. FSAR, Subsection 15.7.4.

3. Regulatory Guide 1.195, May 2003.

BASES

LCO (continued)

Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

This LCO is modified by ~~a Note that~~ two Notes. The first Note 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 that the core time to boil is short, 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. TSTF-349

The second Note permits the RHR pumps to be de-energized for <= 15 minutes when switching from one train to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short (and the core outlet temperature is limited to > 10 degrees F below saturation temperature). The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is

(continued)

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Adoption of Previously NRC-Approved Generic Technical Specification Changes

Enclosure 4

Clean-Typed Technical Specifications Pages

1.0 USE AND APPLICATION

1.1 Definitions

----------NOTE----------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known inputs. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, and trip functions. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

(continued)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

(continued)

1.1 Definitions (continued)

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 14 minutes, making up at least 95% of the total noniodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

(continued)

1.1 Definitions

LEAKAGE (continued)	b. Unidentified LEAKAGE All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

(continued)

1.1 Definitions

PHYSICS TESTS (continued)	a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, Cold Overpressure Protection System (COPS) arming temperature and the nominal PORV setpoints for the COPS, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these operating limits is addressed in individual specifications.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3625.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

(continued)

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

(continued)

1.4 Frequency

DESCRIPTION (continued)

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1 SINGLE FREQUENCY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.04 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2 MULTIPLE FREQUENCIES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3 FREQUENCY BASED ON A SPECIFIED CONDITION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency and the provisions of SR 3.0.3 would apply.

(continued)

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be met in MODE 1. ----------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Only required to be performed in MODE 1. ----------	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be met in MODE 3. ----------	
Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times. As described in Example 1.4-1. However. the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore. if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2). and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore. no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Measured core reactivity not within limit.	A.1	Reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7days
	AND		
	A.2	Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	----------NOTE---------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ----------	
	Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable.	A.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND		
	B.3	Verify SDM is ≥ the limit specified in the COLR.	Once per 12 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	AND		
	B.5	Perform SR 3.2.2.1.	72 hours
	AND		
	B.6	Reevaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify rod freedom of movement by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.3	Verify rod drop time of each rod, from the physical fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 551°F; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1	Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Restore control bank(s) to within limits.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	OR	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2 Restore control bank sequence and overlap to within limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable rod position indicator and each inoperable demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Monitor and Record RCS T_{avg}.	Once per 1 hour
	AND		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors.	Once per 8 hours
	AND		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable DRPIs indirectly by using movable incore detectors.	8 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1 Verify by administrative means all DRPIs for the affected banks are OPERABLE.	Once per 8 hours
	AND	
	D.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours
	OR	
	D.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

LCO 3.2.1 $F_Q(Z)$ shall be within the steady state and transient limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q(Z)$ not within steady state limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q(Z)$ exceeds steady state limit.	15 minutes
	AND		
	A.2	Reduce Power Range Neutron Flux — High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds steady state limit.	72 hours
	AND		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds steady state limit.	72 hours
	AND		
	A.4	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_Q(Z)$ not within transient limit.	B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q(Z)$ exceeds transient limit and control AFD within reduced limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q(Z)$ is within steady state limit.	Once after each refueling after achieving equilibrium conditions at any power level exceeding 50% RTP AND Once after achieving equilibrium conditions after exceeding, by ≥ 20% RTP, the THERMAL POWER at which $F_Q(Z)$ was last verified AND In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- If measurements indicate maximum over Z $\left[\frac{F_Q(Z)}{K(Z)}\right]$ has increased since the previous evaluation of $F_Q(Z)$: a. Increase $F_Q(Z)$ by an appropriate penalty factor specified in the COLR and verify this value is within the transient limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate maximum over Z $\left[\frac{F_Q(Z)}{K(Z)}\right]$ has not increased. ----------	
	Verify $F_Q(Z)$ is within transient limit.	Once after each refueling after achieving equilibrium conditions at any power level exceeding 50% RTP AND

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once after achieving equilibrium conditions after exceeding, by ≥ 20% RTP, the THERMAL POWER at which $F_Q(Z)$ was last verified AND In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limits.	A.1.1	Restore $F^N_{\Delta H}$ to within limits.	4 hours
	OR		
	A.1.2.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND		
	A.1.2.2	Reduce Power Range Neutron Flux — High trip setpoints to ≤ 55% RTP.	72 hours
	AND		
	A.2	Perform SR 3.2.2.1.	24 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	----------NOTE---------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- Perform SR 3.2.2.1.	 Prior to THERMAL POWER exceeding 50% RTP <u>AND</u> Prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD shall be maintained within the limits specified in the COLR.

----------NOTE----------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.6 must be completed whenever Required Action A.5 is implemented. ---------- QPTR not within limit.	A.1	Limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00.	2 hours
	AND		
	A.2.1	Perform SR 3.2.4.1.	Once per 12 hours
	AND		
	A.2.2	Limit THERMAL POWER to ≥ 3% below RTP for each 1% QPTR > 1.00.	----------NOTE---------- For performances of Required Action A.2.2 the Completion Time is measured from the completion of SR 3.2.4.1. ---------- 2 hours
	AND		
	A.3	Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Actions A.1 and A.2

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)			AND Once per 7 days thereafter
	AND		
	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
	AND		
	A.5	----------NOTE---------- Perform Required Action A.5 only after Required Action A.4 is completed. ----------	
		Calibrate excore detectors to show QPTR = 1.00.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	----------NOTE---------- Perform Required Action A.6 only after Required Action A.5 is completed. ---------- Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	 ----------NOTE---------- Only one of the following Completion Times, whichever becomes applicable first, must be met. ---------- Within 24 hours after reaching RTP <u>OR</u> Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- With one power range channel inoperable, the remaining three power range channels can be used for calculating QPTR. ---------- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program
SR 3.2.4.2	----------NOTE---------- Only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER ≥ 75% RTP. ---------- Confirm that the normalized symmetric power distribution is consistent with QPTR.	Once within 12 hours <u>AND</u> In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required Functions inoperable.	A.1	Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL CHECK for each required monitoring instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.3	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION for each required monitoring instrumentation channel.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 551°F.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

----------NOTE----------
All reactor coolant pumps may be de-energized for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause reduction of the RCS boron concentration; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One required RCS loop not in operation, with Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation. OR C.2 Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour 1 hour
D. Two required RCS loops inoperable. OR No RCS loop in operation.	D.1 Place the Rod Control System in a condition incapable of rod withdrawal. AND D.2 Suspend all operations involving a reduction of RCS boron concentration. AND D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify steam generator secondary side water levels are above the highest point of the steam generator U-tubes for required RCS loops.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 92%; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Pressurizer water level not within limit.	A.1	Be in MODE 3.	6 hours
	AND		
	A.2	Fully insert all rods.	6 hours
	AND		
	A.3	Place Rod Control System in a condition incapable of rod withdrawal.	6 hours
	AND		
	A.4	Be in MODE 4.	12 hours
B. One required group of pressurizer heaters inoperable.	B.1	Restore required group of pressurizer heaters to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 92%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each PORV and each block valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve.	1 hour
	AND	
	B.2 Remove power from associated block valve.	1 hour
	AND	
	B.3 Restore PORV to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	C.1	Place associated PORV in manual control.	1 hour
	AND		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable and not capable of being manually cycled.	E.1	Close associated block valves.	1 hour
	AND		
	E.2	Remove power from associated block valves.	1 hour
	AND		
	E.3	Be in MODE 3.	6 hours
	AND		
	E.4	Be in MODE 4.	12 hours
F. Two block valves inoperable.	F.1	Restore one block valve to OPERABLE status.	2 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTES---------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. ---------- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Cold Overpressure Protection Systems (COPS)

LCO 3.4.12 A COPS shall be OPERABLE with all safety injection pumps incapable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two RCS relief valves, as follows:

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 440 psig and ≤ 460 psig, or

3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint within specified limits.

b. The RCS depressurized and an RCS vent of ≥ 1.5 square inches (based on an equivalent length of 10 feet of pipe).

APPLICABILITY: MODE 4 with any RCS cold leg temperature ≤ the COPS arming temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

----------NOTE----------
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

ACTIONS

----------NOTE----------
LCO 3.0.4b is not applicable for entry into MODE 4, entry into MODE 6 with reactor vessel head on from MODE 6, and entry into MODE 5 from MODE 6 with the reactor vessel head on.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more safety injection pumps capable of injecting into the RCS.	A.1 Render all safety injection pumps incapable of injecting into the RCS.	4 hours
B. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	B.1 Isolate affected accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature to > the COPS arming temperature specified in the PTLR.	12 hours
	OR	
	C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
D. One required RCS relief valve inoperable in MODE 4 with any RCS cold leg temperature ≤ the COPS arming temperature specified in the PTLR.	D.1 Restore required RCS relief valve to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One required RCS relief valve inoperable in MODE 5 or 6.	E.1 Restore required RCS relief valve to OPERABLE status.	24 hours
F. Two required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, C, D, or E not met. OR COPS inoperable for any reason other than Condition A, B, D, or E.	F.1 Depressurize RCS and establish RCS vent size within specified limits.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify both safety injection pumps are incapable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify each accumulator is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	Verify RHR suction valves are open for each required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	----------NOTE---------- Only required to be met when complying with LCO 3.4.12.b. ---------- Verify RCS vent size within specified limits.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.6	----------NOTE---------- Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ the COPS arming temperature specified in the PTLR. ---------- Perform a COT on each required PORV, excluding actuation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.7	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

----------NOTE----------
LCO 3.0.4c is applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/$\bar{E}$ μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	----------NOTE---------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---------- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	In accordance with the Surveillance Frequency Control Program

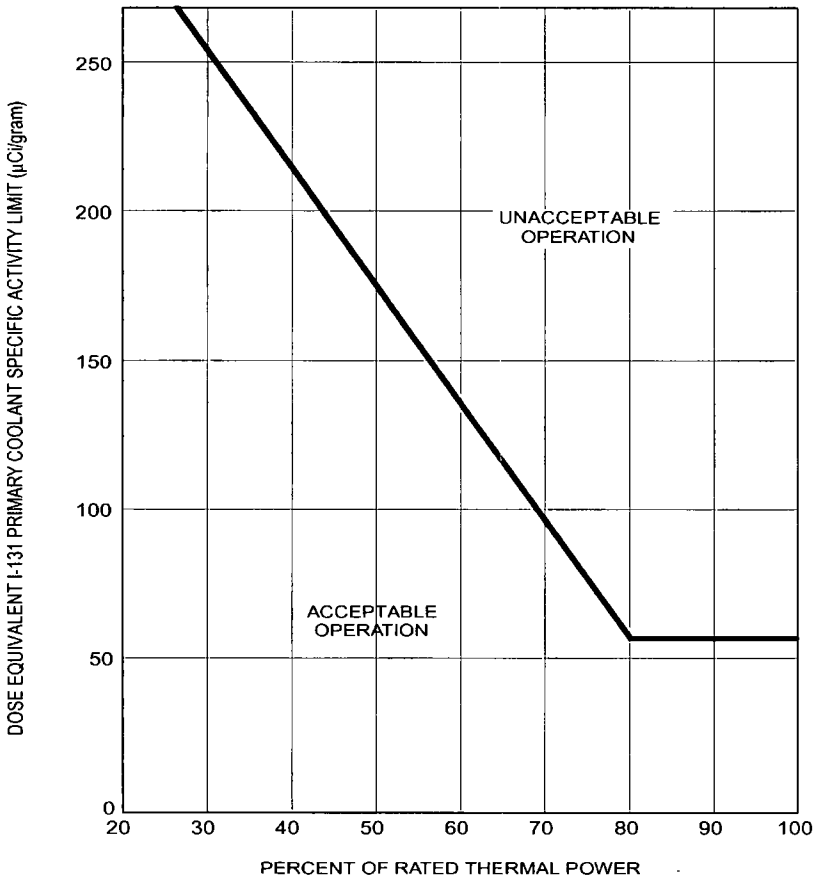

FIGURE 3.4.16-1
REACTOR COOLANT DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACITVITY >1 mCi/gram DOSE EQUIVALENT I-131

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

----------NOTES----------

1. Penetration flow path(s) (except for 24 inch purge valves) may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one containment isolation valve inoperable except for purge valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. One or more penetration flow paths with two containment isolation valves inoperable except for purge valve leakage not within limit.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
	AND		
	C.2	----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ----------	
		Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 24 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition C of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each 14 inch purge valve is closed, except when the associated penetration(s) is (are) permitted to be open for purge or venting operations and purge system surveillance and maintenance testing under administrative control.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	 In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	----------NOTES---------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. The fuel transfer tube blind flange is only required to be verified closed once after refueling prior to entering MODE 4 from MODE 5. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for containment purge valves with resilient seals.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
LCO 3.0.4b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable. OR ----------NOTE---------- Only applicable if MODE 2 has not be entered following refueling. ---------- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met. OR Two AFW trains inoperable.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	6 hours 12 hours
D. Three AFW trains inoperable.	D.1	----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------- Initiate action to restore one AFW train to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	----------NOTE---------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	----------NOTE---------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. 2. AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.5	Verify that each AFW pumphouse ESF supply fan starts and associated dampers actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.6	Verify that the ESF outside air intake and exhaust dampers for the turbine-driven AFW pump actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, Starting Air, and Ventilation

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits and ventilation supply fans OPERABLE for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel level < 68,000 gal and > 52,000 gal in storage tank.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more DGs with lube oil inventory < 336 gal and > 288 gal.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more DGs with stored fuel oil total particulates not within limit.	C.1 Restore fuel oil total particulates within limit.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with both starting air receiver pressures < 210 psig and ≥ 175 psig.	E.1 Restore one starting air receiver pressure per DG to ≥ 210 psig.	48 hours
F. One or more DGs with one ventilation supply fan inoperable per DG.	F.1 Restore ventilation supply fan to OPERABLE status.	14 days
G. Required Action and associated Completion Time not met. OR One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E. OR One or more DGs with two ventilation supply fans inoperable per DG.	G.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 68,000 gal of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify lube oil inventory is ≥ 336 gal.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG has one air start receiver with a pressure ≥ 210 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.6	Verify each DG ventilation supply fan starts and the necessary dampers actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

----------NOTE----------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
	AND		
	A.3	Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control

SURVEILLANCE	FREQUENCY
	Program

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

---------- NOTE ----------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. AND	Immediately

A.	(continued)	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.2	----------NOTE---------- Not required to be met for containment purge and exhaust valves(s) in penetrations closed to comply with LCO 3.9.4.c.1. ---------- Verify at least two containment ventilation valves in each open containment ventilation penetration providing direct access from the containment atmosphere to the outside atmosphere are capable of being closed from the control room.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.3	----------NOTE---------- Only required for an open equipment hatch. ---------- Verify the capability to install the equipment hatch.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

----------NOTES----------

1. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

2. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System (RCS) boron concentration; and

   c. No draining operations to further reduce RCS water volume are permitted.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status. <u>OR</u>	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one RHR loop to operation.	Immediately
	AND		
	B.3	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	In accordance with the Surveillance Frequency Control Program

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Vice President - Vogtle; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page

(continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include:

1) Residual Heat Removal System;
2) Containment Spray System;
3) Safety Injection (excluding Boron Injection and Accumulators);
4) Chemical and Volume Control System (Letdown and Charging Systems);
5) Post Accident Processing System (until such time as a modification eliminates the Post Accident Processing System as a potential leakage path);
6) Gaseous Waste Processing System; and
7) Nuclear Sampling System (Pressurizer steam and liquid sampling lines, Reactor Coolant sample lines, RHR sample lines, CVCS Demineralizer and Letdown Heat Exchanger sample lines only).

The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Leak test requirements for each system at least once per 18 months. The provisions of SR 3.0.2 are applicable

5.5.3 Not Used.

(continued)

5.5 Programs and Manuals

5.5.4 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:

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

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentrations stated in 10 CFR 20, Appendix B (to paragraphs 20.1001-20.2401), Table 2, Column 2;

c. 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;

d. 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;

e. 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;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at and beyond the site boundary as follows:

1. For noble gases: dose rates of ≤ 500 mrem/yr to the whole body and 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives > 8 days: a dose rate of ≤ 1500 mrem/yr to any organ;

h. 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;

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

j. 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.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the cyclic and transient occurrences to ensure that components are maintained within the design limits. The component cyclic or transient limits are provided in FSAR, Section 3.9.

(continued)

5.5 Programs and Manuals

5.5.6 Prestressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Boiler and Pressure Vessel Code Section XI, Subsection IWL and applicable addenda as required by 10 CFR 50.55a except where an exemption, relief, or alternative has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel at least once per 10 years by conducting either:

a. An in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius; or

b. A surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

The provisions of SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program.

(continued)

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

Tubes with service-induced flaws located greater than 15.2 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.2 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. Portions of the tube below 15.2 inches below the top of the tubesheet are excluded from this requirement.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period.

   b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

   c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

3. If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

(continued)

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry Program (continued)

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1980:

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flow rate specified below ± 10%.

ESF Ventilation System	Flow Rate
Control Room Emergency Filtration System (CREFS)	19,000 CFM
Piping Penetration Area Filtration and Exhaust (PPAFES)	15,500 CFM

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flow rate specified below ± 10%.

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flow Rate
CREFS	19,000 CFM
PPAFES	15,500 CFM

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
CREFS	.2%	70%
PPAFES	10%	95%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the charcoal adsorbers, and CREFS cooling coils is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flow rate specified below ± 10%.

ESF Ventilation System	Delta P	Flow Rate
CREFS	7.1 in. water gauge	19,000 CFM
PPAFES	6 in. water gauge	15,500 CFM

e. Demonstrate that the heaters for the CREFS dissipate ≥ 95 kW when corrected to 460 V when tested in accordance with ASME N510-1989.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System, the quantity of radioactivity contained in each Gas Decay Tank, and the quantity of radioactivity contained in

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be limited to 10 curies per outdoor tank in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is limited to ≤ 10 curies per tank, excluding tritium and dissolved or entrained noble gases. This surveillance program provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

(continued)

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

1. an API gravity or an absolute specific gravity within limits, or an API gravity or specific gravity within limits when compared to the supplier's certificate;

2. a flash point within limits for ASTM 2D fuel oil, and, if gravity was not determined by comparison with supplier's certification, a kinematic viscosity within limits for ASTM 2D fuel oil; and

3. a clear and bright appearance with proper color.

b. Other properties for ASTM 2D fuel oil are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

5.5.14 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. a 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.

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

e. Proposed changes that meet the criteria of (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).

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 MS and FW Piping Inspection Program

This program shall provide for the inspection of the four Main Steam and Feedwater lines from the containment penetration flued head outboard welds, up to the first five-way restraint. The extent of the inservice examinations completed during each inspection interval (ASME Code Section XI) shall provide 100% volumetric examination of circumferential and longitudinal welds to the extent practical. This augmented inservice inspection is consistent with the requirements of NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," November 1975 and Section 6.6 of the FSAR.

5.5.17 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, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995, as modified by the following exceptions:

1. Leakage rate testing for containment purge valves with resilient seals is performed once per 18 months in accordance with LCO 3.6.3, SR 3.6.3.6 and SR 3.0.2.

2. Containment personnel air lock door seals will be tested prior to reestablishing containment integrity when the air lock has been used for containment entry. When containment integrity is required and the air lock has been used for containment entry, door seals will be tested at least once per 30 days during the period that containment entry(ies) is (are) being made.

3. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief or alternative has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

4. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria are $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are` $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.18 Configuration Risk Management Program

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:

(continued)

5.5 Programs and Manuals

5.5.18 Configuration Risk Management Program (continued)

a. Provisions for the control and implementation of a Level 1 at power internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the LCO Condition for preplanned activities.

c. Provisions for performing an assessment after entering the LCO Condition for unplanned entry into the LCO Condition.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Condition.

e. Provisions for considering other applicable risk significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

5.5.19 Battery Monitoring and Maintenance Program

This program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," of the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.

(continued)

5.5 Programs and Manuals (continued)

5.5.20 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

(continued)

5.5 Programs and Manuals

5.5.20 Control Room Envelope Habitability Program (continued)

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.21 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Vogtle Electric Generating Plant
Request for Technical Specifications Amendment
Adoption of Previously NRC-Approved Generic Technical Specification Changes

Enclosure 5

Summary of Regulatory Commitments

Summary of Regulatory Commitments

The following table identifies the regulatory commitments in this document. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENTS	DUE DATE / EVENT
1. Administrative methods will be established to control performance of the 10 year diesel fuel oil storage tank cleaning activities that are currently described in SR 3.8.3.7.	90 days from NRC approval of LAR
2. Administrative controls will be established to ensure appropriate personnel are aware of the open status of the penetration flow path(s) during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment.	90 days from NRC approval of LAR
3. Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.	90 days from NRC approval of LAR
4. The time needed to close open containment penetration(s) will be incorporated into the confirmatory dose calculation for FHAs.	90 days from NRC approval of LAR