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Serial: RA-16-0028 10 CFR 50.90
December 15, 2016

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

CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-413 AND 50-414
RENEWED LICENSE NOS. NPF-35 AND NPF-52

SUBJECT: APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT MULTIPLE TECHNICAL SPECIFICATION TASK FORCE TRAVELERS

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC, referred to henceforth as “Duke Energy”, is submitting a request for amendments to the Technical Specifications (TSs) for Catawba Nuclear Station, Units 1 and 2 (CNS).

The requested amendments will adopt a selection of previously NRC-approved Technical Specification Task Force (TSTF) Travelers. TSTF Travelers are generic changes to the Improved Standard Technical Specifications. A list of the Travelers proposed for adoption is included in Enclosure 1. The proposed amendments will bring the CNS TSs into closer alignment with NUREG-1431, “Standard Technical Specifications – Westinghouse Plants.”

Enclosure 1 provides the evaluation of the proposed changes, including the No Significant Hazards Consideration Determination and Environmental Consideration determinations. 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. Enclosure 2 provides the existing TS pages marked up to show the proposed changes. Enclosure 3 provides existing TS Bases pages marked up to show the proposed changes. Changes to the existing TS Bases will be implemented under the Technical Specification Bases Control Program. They are provided in Enclosure 3 for information only. The retyped TS pages will be provided to the NRC immediately prior to issuance of the approved amendment.

Duke Energy’s plans to submit this application were discussed in a pre-submittal meeting with the NRC on November 16, 2016.

Approval of the proposed amendment is requested by December 14, 2017. Once approved, the amendment shall be implemented within 120 days.

This submittal contains no new regulatory commitments. In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this license amendment request by transmitting a copy of this letter and enclosures to the designated State Official.

Should you have any questions concerning this letter, or require additional information, please contact Cecil Fletcher, Manager – Nuclear Regulatory Affairs, at 803-701-3622.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 15, 2016.

Sincerely,

Tom Simril

Tom Simril
Vice President – Catawba Nuclear Station

JLV

Enclosures:
1. Evaluation of the Proposed Changes
2. Proposed Technical Specification Changes (Mark-Up)
3. Proposed Technical Specification Bases Changes (Mark-Up) (For information only)

cc: C. Haney, USNRC Region II
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Enclosure 1

EVALUATION OF THE PROPOSED CHANGES

Subject: Application to Revise Technical Specifications to Adopt Multiple Technical Specification Task Force Travelers

1.0 Summary Description

The requested amendments will adopt a selection of NRC-approved Technical Specification Task Force (TSTF) Travelers. TSTF Travelers are generic changes to the Improved Standard Technical Specifications (ISTS).

The proposed amendments will bring the Catawba Nuclear Station (CNS) Technical Specifications (TSs) into closer alignment with NUREG-1431, “Standard Technical Specifications – Westinghouse Plants”. The Travelers included in the scope of this license amendment application are as follows:

- TSTF-142-A, Rev. 0, “Increase the Completion Time When the Core Reactivity Balance is Not Within Limit”
- TSTF-197-A, Rev. 2, “Require Containment Closure When Shutdown Cooling Requirements Are Not Met”
- TSTF-269-A, Rev. 2, “Allow Administrative Means of Position Verification for Locked or Sealed Valves”
- TSTF-283-A, Rev. 3, “Modify Section 3.8 Mode Restriction Notes”
- TSTF-285-A, Rev. 1, “Charging Pump Swap LTOP Allowance”
- TSTF-315-A, Rev. 0, “Reduce Plant Trips Due to Spurious Signals to the NIS During Physics Testing”
- TSTF-340-A, Rev. 3, “Allow 7 Day Completion Time for a Turbine-Driven AFW Pump Inoperable”
- TSTF-349-A, Rev. 1, “Add Note to LCO 3.9.5 Allowing Shutdown Cooling Loops Removal from Operation”
- TSTF-352-A, Rev. 1, “Provide Consistent Completion Time to Reach MODE 4”
- TSTF-361-A, Rev. 2, “Allow Standby SDC/RHR/DHR Loop to be Inoperable to Support Testing”
- TSTF-438-A, Rev. 0, “Clarify Exception Notes to be Consistent with the Requirement Being Excepted”

2.0 Proposed Changes, Justifications, and No Significant Hazards Consideration Determinations

The Travelers proposed for adoption are discussed in Sections 2.1 through 2.9 below. Each section contains the following topics:

Description of Proposed Changes

This topic describes the effect of adopting the subject Traveler on the CNS TSs.

Differences Between the Proposed Changes and the Approved Traveler

This topic describes differences between the changes proposed to the CNS TSs and the ISTS mark-ups provided in the approved Traveler.

Summary of the Approved Traveler Justification

This topic summarizes the justification provided in the Traveler.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

This topic describes any differences between the justification provided in the Traveler and the justification for adopting the Traveler in the CNS TSs.

NRC Approval

This topic discusses NRC approval of the Traveler.

Precedent

This topic provides precedents of adoption of the Traveler.

List of Affected Pages

This topic lists the CNS TS and TS 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.

No Significant Hazards Consideration Determination

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."

Given that CNS is a Westinghouse design, the term "ISTS" as used herein refers to NUREG-1431, "Standard Technical Specifications – Westinghouse Plants".

Existing CNS TS pages marked up to show the proposed changes are provided in Enclosure 2.

Existing CNS TS Bases pages marked up to show the proposed changes are provided in Enclosure 3. Changes to the existing TS Bases will be implemented under the Technical Specification Bases Control Program, and are provided in Enclosure 3 for information only.

2.1 TSTF-142-A, Rev. 0, “Increase the Completion Time When the Core Reactivity Balance is Not Within Limit”

Description of Proposed Changes

The proposed changes affect CNS TS 3.1.2, “Core Reactivity”. Condition A applies when the measured core reactivity is not within the specified limit of predicted values, and includes two required actions. Required Action A.1 requires re-evaluation of the core design and safety analysis to determine that the reactor core is acceptable for continued operation. Required Action A.2 requires establishment of appropriate operating restrictions and Surveillance Requirements (SRs). The proposed changes would extend the Completion Times for Required Actions A.1 and A.2 from 72 hours to 7 days, consistent with TSTF-142-A.

Differences Between the Proposed Changes and the Approved Traveler

The ISTS markup for TS 3.1.3, “Core Reactivity,” that was included in TSTF-142-A, was based on Revision 1 of NUREG-1431. CNS TS 3.1.2, “Core Reactivity,” corresponds to the ISTS markup included in TSTF-142-A.

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 a Design Basis Accident (DBA) or anticipated transient approaching the core design limits occurring during the 7 day period.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes.

NRC Approval

The NRC did not issue a letter approving this Traveler, but it was incorporated by the NRC into Revision 2 of the ISTS.

Precedent

An example of a plant-specific NRC approval of the changes in TSTF-142-A is Vogtle Electric Generating Plant, Units 1 and 2, Amendment Numbers 180/161, dated June 9, 2016 (ADAMS Accession No. ML15132A569).

List of Affected Pages

3.1.2-1
B 3.1.2-4

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 a Limiting Condition for Operation (LCO) is not met. The ISTS for Westinghouse Plants (NUREG-1431) provides 7 days to take action when the measured core reactivity is not within limit. The proposed Completion Time is consistent with the NUREG-1431 value.

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC ("Duke Energy") requests adoption of TSTF-142-A, Revision 0, "Increase the Completion Time When the Core Reactivity Balance is Not Within Limit," which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station, Units 1 and 2 Technical Specifications. The proposed changes revise Catawba Nuclear Station (CNS) Technical Specification (TS) 3.1.2, "Core Reactivity," to extend the allowable time to complete specified required actions, in the event measured core reactivity is not within limits, from 72 hours to 7 days.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed changes 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 extend 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

radiological 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 changes do 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 changes do 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 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 provide 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 changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-197-A, Rev. 2, “Require containment closure when shutdown cooling requirements are not met.”

Description of Proposed Changes

The proposed changes affect CNS TS 3.9.4, “Residual Heat Removal (RHR) and Coolant Circulation – High Water Level”, and CNS TS 3.9.5, “Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level”, and is consistent with TSTF-197-A.

Condition A of TS 3.9.4 applies when Residual Heat Removal (RHR) requirements are not met, and includes four required actions. Required Action A.4 requires, within four hours, the closure of all containment penetrations providing direct access from containment atmosphere to outside atmosphere. The proposed changes revise Required Action A.4 and add new Required Actions A.5, A.6.1, and A.6.2 to clarify that the intent of the required actions is to establish containment closure. Each of these required actions will have a completion time of four hours. The revised Required Action A.4 will require that the containment equipment hatch be closed and secured with four bolts. The new Required Action A.5 will require that one door in each air lock be closed. The new Required Action A.6 will require either: 1) closure of each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent (Required Action A.6.1); or 2) verification that each penetration is capable of being closed on a high containment radiation signal (Required Action A.6.2).

Condition B of TS 3.9.5 applies when no RHR loop is in operation, and includes three required actions. Required Action B.3 requires the closure of all containment penetrations providing direct access from containment atmosphere to outside atmosphere. The proposed changes are the same as the proposed changes to TS 3.9.4, consisting of a revision to Required Action B.3 and the addition of new Required Actions B.4, B.5.1, and B.5.2.

Differences Between the Proposed Changes and the Approved Traveler

CNS TS 3.9.4, “Residual Heat Removal (RHR) and Coolant Circulation – High Water Level,” is equivalent to TS 3.9.5, “RHR and Coolant Circulation – High Water Level,” in the ISTS markup included in TSTF-197-A. Also, CNS TS 3.9.5, “Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level,” is equivalent to TS 3.9.6, “RHR and Coolant Circulation – Low Water Level,” in ISTS markup included in TSTF-197-A.

The ISTS markups for TS 3.9.5 Required Action A.4 and TS 3.9.6 Required Action B.3, as included in TSTF-197-A for NUREG-1431, are proposed to read: “Close equipment hatch and secure with [four] bolts.” The CNS change for these associated actions propose to use more precise wording by specifying the “containment” equipment hatch in particular. This avoids potential confusion with other types of equipment hatches at CNS, such as the divider barrier equipment hatches utilized in the CNS ice condenser containment. Note that the use of four bolts to secure the containment equipment hatch is consistent with current CNS TS 3.9.3, “Containment Penetrations”.

The ISTS markups for TS 3.9.5 Required Action A.6.2 and TS 3.9.6 Required Action B.5.2, as included in TSTF-197-A for NUREG-1431, are proposed to read: “Verify each penetration is capable of being closed by an OPERABLE Containment Purge and

Exhaust Isolation System." CNS has a Containment Purge Exhaust System. At CNS, as described in Updated Safety Analysis Report (UFSAR) Sections 6.2.4 and 9.4.5, the containment purge isolation valves close on a Phase A Containment Isolation Signal (SI) or a high containment activity signal. Also note that per TS Table 3.3.2-1. Safety Injection (SI) is required to be operable only in Modes 1 through 4, whereas TS 3.9.4 and TS 3.9.5 are applicable in Mode 6. Therefore, the relevant containment purge valve closure signal in Mode 6 is the high containment activity signal. The containment high radiation monitor is addressed in Selected Licensee Commitment (SLC) 16.7.10, "Radiation Monitoring for Plant Operations", but not in TS. The term "OPERABLE" is intended to apply only to TS. Considering the above, the wording for CNS TS 3.9.4 Required Action A.6.2 and TS 3.9.5 Required Action B.5.2 is proposed to read: "Verify each penetration is capable of being closed on a high containment radiation signal."

<u>Summary of the Approved Traveler Justification</u>

The existing action requirements are vague and, in some cases, overly restrictive. There is no reason that the containment purge valves should be closed as long as they are operable (i.e., will close automatically on a Containment High Radiation Signal). The overly restrictive action requirements are replaced with a requirement to: a) close the equipment hatch and secure with [four] bolts; b) close one door in each air lock; and c) either close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent, or verify the penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

This change has several advantages. First, establishing containment closure meets the Bases description for the Action, which is to prevent fission products from being released from the containment during a loss of shutdown cooling event. Secondly, containment closure is a well understood and controlled condition which is used routinely during a refueling outage. The current action requirement to close all penetrations providing direct access from the containment atmosphere to the outside atmosphere is a rarely used arrangement. Utilizing containment closure instead of the current special actions gives greater confidence that the containment will be in the appropriate state.

Therefore, the analysis assumptions and Bases assumptions for the required actions are preserved while eliminating an unclear requirement, lessening the administrative burden on the plant, and increasing confidence that the containment will be in the proper status should an event occur.

<u>Differences Between the Plant-Specific Justification and the Approved Traveler Justification</u>

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes beyond those described above.

<u>NRC Approval</u>

The NRC documented their approval of TSTF-197-A, Revision 2, in a letter from William Beckner (NRC) to James Davis (NEI) dated July 26, 1999 (ADAMS Accession No. 9907300113).

Precedent

An example of a plant-specific NRC approval of the changes in TSTF-197-A is Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Amendment Numbers 268/244, dated September 13, 2004 (ADAMS Accession No. ML042530436).

List of Affected Pages

3.9.4-2
3.9.5-2
B 3.9.4-4
B 3.9.5-3

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 changes clarify the remedial actions and remove an operational restriction not needed for safety. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC ("Duke Energy") requests adoption of TSTF-197-A, Revision 2, "Require Containment Closure When Shutdown Cooling Requirements Are Not Met," which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station, Units 1 and 2 Technical Specifications. The proposed changes revise Catawba Nuclear Station (CNS) Technical Specification (TS) 3.9.4, "RHR and Coolant Circulation – High Water Level", and CNS TS 3.9.5, "RHR and Coolant Circulation – Low Water Level," to clarify that the intent of the required actions is to establish containment closure.

Duke Energy 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 revise the CNS TS to ensure that the appropriate actions are taken to establish containment closure in the event that Residual Heat Removal requirements are not met during refueling operations. Containment closure would be

appropriate for mitigation of a loss of shutdown cooling accident, but it does not affect the initiation of the accident. The containment purge system isolation valves will be capable of being closed automatically on a high containment radiation signal, such that there will be no significant increase in the radiological consequences of a loss of shutdown cooling. Therefore, the proposed changes do 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 changes do 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 containment purge system isolation valves will remain capable of being closed automatically on a high containment radiation signal. 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.

   Currently the Technical Specifications are vague and overly restrictive concerning the requirement for containment closure when shutdown cooling is lost. The proposed changes eliminate unclear requirements and provide a clear way to establish containment closure that meets the Bases description, which is to prevent radioactive gas from being released from the containment during a loss of shutdown cooling incident. The containment purge system isolation valves will remain capable of being closed automatically on a high containment radiation signal. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-269-A, Rev. 2, "Allow Administrative Means of Position Verification for Locked or Sealed Valves"

Description of Proposed Changes

The proposed changes affect CNS TS 3.6.3, "Containment Isolation Valves." This TS currently includes Required Actions to verify that the affected containment penetration flow path is isolated and periodically verified to be isolated. Consistent with TSTF-269-A, a Note is proposed to be added to TS 3.6.3 Required Actions A.2, C.2, and E.2, to allow isolation devices that are locked, sealed, or otherwise secured to be verified by use of administrative means.

Differences Between the Proposed Changes and the Approved Traveler

None.

Summary of the Approved Traveler Justification

The purpose of the periodic verification that a penetration with an inoperable isolation valve continues to be isolated is to detect and correct inadvertent repositioning of the isolation device. However, the function of locking, sealing, or securing an isolation device is to ensure that the device is not inadvertently repositioned. Therefore, it is sufficient to assume that the initial establishment of component status (e.g., isolation valves closed) was performed correctly, and subsequent periodic re-verification need only be a verification of the administrative control that ensures that the component remains in the required state. It is unnecessary and undesirable to remove the lock, seal, or other means of securing the component solely to perform an active verification of the required state, as it would increase the chance of mis-positioning due to the frequent manipulation.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes.

NRC Approval

The NRC documented their approval of TSTF-269-A, Revision 2, in a letter from William Beckner (NRC) to James Davis (NEI) dated July 26, 1999 (ADAMS Accession No. 9907300113).

Precedent

An example of a plant-specific NRC approval of the changes in TSTF-269-A is Edwin I. Hatch Nuclear Plant, Units 1 and 2, Amendment Numbers 279/223, dated September 29, 2016 (ADAMS Accession No. ML16231A041).

List of Affected Pages

3.6.3-2

3.6.3-3
3.6.3-4
B 3.6.3-7
B 3.6.3-8
B 3.6.3-10

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 changes remove an operational restriction not needed for safety. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC ("Duke Energy") requests adoption of TSTF-269-A, Rev. 2, "Allow Administrative Means of Position Verification for Locked or Sealed Valves", which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station, Units 1 and 2 Technical Specifications. The proposed changes revise Catawba Nuclear Station (CNS) Technical Specification (TS) 3.6.3, "Containment Isolation Valves", to reduce operational burden by adding provisions which allow isolation devices that are locked, sealed, or otherwise secured, to be verified by use of administrative means.

Duke Energy 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 modify CNS TS 3.6.3, "Containment Isolation Valves". This TS currently includes actions that require penetrations to be isolated and periodically verified to be isolated. A Note is proposed to be added to TS 3.6.3 Required Actions A.2, C.2, and E.2, to allow isolation devices that are locked, sealed, or otherwise secured to be verified by use of administrative means. The proposed changes do not affect any plant equipment, test methods, or plant operation, and is not an initiator of any analyzed accident sequence. The inoperable containment penetrations will continue to be isolated, and hence perform their isolation function. Operation in accordance with the proposed TSs will ensure that all analyzed accidents will continue to be mitigated as previously analyzed. Therefore, the

proposed changes do 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 changes do 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 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 will not affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. Affected containment penetrations will continue to be isolated as required by the existing TS. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-283-A, Rev. 3, “Modify Section 3.8 Mode Restriction Notes”

Description of Proposed Changes

The proposed changes affect CNS TS 3.8.1, “AC Sources – Operating”. Consistent with TSTF-283-A, Notes would be added to allow greater flexibility in performing Surveillance Requirements (SRs). The following SRs are affected:

- SR 3.8.1.11, which tests the response to a loss of offsite power signal, contains a Note prohibiting performance in Mode 1, 2, 3, or 4. The Note is proposed to be modified to state that performance is normally prohibited in Mode 1, 2, 3, or 4, but portions of the SR may be performed to reestablish operability provided an assessment determines the safety of the plant is maintained or enhanced. The modified Note would also allow credit to be taken for unplanned events that satisfy this SR.

- SR 3.8.1.16, which verifies the transfer from Diesel Generator (DG) to offsite power, contains a Note prohibiting performance in Mode 1, 2, 3, or 4. The Note is proposed to be modified to state that performance is normally prohibited in Mode 1, 2, 3, or 4, but may be performed to reestablish operability provided an assessment determines the safety of the plant is maintained or enhanced. The modified Note would also allow credit to be taken for unplanned events that satisfy this SR.

- SR 3.8.1.17, which verifies DG operation in test mode, contains a Note prohibiting performance in Mode 1, 2, 3, or 4. The Note is proposed to be modified to state that performance is normally prohibited in Mode 1, 2, 3, or 4, but portions of the SR may be performed to reestablish operability provided an assessment determines the safety of the plant is maintained or enhanced. The modified Note would also allow credit to be taken for unplanned events that satisfy this SR.

- SR 3.8.1.19, which verifies the response to a loss of offsite power signal and Engineered Safety Feature (ESF) actuation signal, contains a Note prohibiting performance in Mode 1, 2, 3, or 4. The Note is proposed to be modified to state that performance is normally prohibited in Mode 1, 2, 3, or 4, but portions of the SR may be performed to reestablish operability provided an assessment determines the safety of the plant is maintained or enhanced. The modified Note would also allow credit to be taken for unplanned events that satisfy this SR.

- SR 3.8.4.8 (ISTS SR 3.8.4.7), which is a test of the battery capacity and its ability to provide a high rate, short duration load, contains a Note prohibiting performance in Mode 1, 2, 3, or 4. The Note is proposed to be modified to state that performance is normally prohibited in Mode 1, 2, 3, or 4, but portions of the SR may be performed to reestablish operability provided an assessment determines the safety of the plant is maintained or enhanced. The modified Note would also allow credit to be taken for unplanned events that satisfy this SR.

- SR 3.8.4.9 (ISTS SR 3.8.4.8), which is a constant current capacity test to detect any change in the capacity determined by the acceptance test, contains a Note prohibiting performance in Mode 1, 2, 3, or 4. The Note is proposed to be

modified to state that performance is normally prohibited in Mode 1, 2, 3, or 4, but portions of the SR may be performed to reestablish operability provided an assessment determines the safety of the plant is maintained or enhanced. The modified Note would also allow credit to be taken for unplanned events that satisfy this SR.

<u>Differences Between the Proposed Changes and the Approved Traveler</u>

Each of the proposed changes to the CNS SRs include a sentence that allows credit to be taken for unplanned events that satisfy the SR. Although this change is not in the scope of TSTF-283-A, the allowance to take credit for unplanned events was included in NUREG-1431 Rev. 1 (which was the version of the ISTS which was marked up in TSTF-283-A), and is included in the current version of the ISTS (NUREG-1431 Rev. 4). The allowance to take credit for unplanned events is also consistent with ISTS Bases SR 3.0.1, which states:

> Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Several of the changes approved in TSTF-283-A are not applicable to the CNS TS. Specifically, CNS TS SRs 3.8.1.8, 3.8.1.9, 3.8.1.10, 3.8.1.12, 3.8.1.13, 3.8.1.14, 3.8.1.18, and 3.8.4.7 (ISTS SR 3.8.4.6) do not include Notes restricting the MODES in which the SR may be performed. As such, addition of the TSTF-283-A changes that provide exceptions to the Mode restrictions are not necessary, and are not adopted.

CNS SR 3.8.4.8 is equivalent to SR 3.8.4.7, in the ISTS markup included in TSTF-283-A. CNS SR 3.8.4.9 is equivalent to SR 3.8.4.8, in the ISTS markup included in TSTF-283-A.

The TSTF-283-A Bases changes associated with SRs 3.8.1.11, 3.8.1.16, 3.8.1.17, 3.8.4.8, 3.8.4.9, and 3.8.1.19 incorrectly state that the associated Notes restrict performance of the Surveillances in Mode 1 and 2. These Surveillances actually restrict performance of the Surveillances in Mode 1, 2, 3, or 4. This error is corrected in the CNS TS Bases markups included in Enclosure 3. In addition, the CNS TS Bases markups reflect the allowance being added to the CNS TS to take credit for unplanned events.

<u>Summary of the Approved Traveler Justification</u>

The proposed changes will reduce the potential for a plant shutdown should corrective maintenance (planned or unplanned) performed during power operation result in the need to perform any of the revised SRs to demonstrate operability.

The allowance to perform the SRs in currently prohibited Modes is restricted to only allow the SRs to be performed for the purpose of reestablishing operability (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated operability concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed SR, a successful SR, and a perturbation of the offsite or onsite system when they

are tied together or operated independently for the SR; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the SR is performed. Risk insights or deterministic methods may be used for this assessment.

Note that the Maintenance Rule provision contained in 10 CFR 50.65(a)(4) states that before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. This includes the performance of SRs to reestablish operability. Therefore, in addition to the assessment required by the SR Notes, an assessment of plant risk will also be performed.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes beyond those described above.

NRC Approval

The NRC did not issue a letter approving this Traveler, but it was incorporated by the NRC into Revision 2 of the ISTS.

Precedent

Two examples of plant-specific NRC approvals of the changes in TSTF-283-A are Edwin I. Hatch Nuclear Plant, Units 1 and 2, Amendment Numbers 279/223, dated September 29, 2016 (ADAMS Accession No. ML16231A041), and Joseph M. Farley Nuclear Plant, Units 1 and 2, Amendment Numbers 203/199, dated August 3, 2016 (ADAMS Accession No. ML15233A448).

List of Affected Pages

3.8.1-9
3.8.1-12
3.8.1-13
3.8.1-14
3.8.4-3
3.8.4-4
B 3.8.1-23
B 3.8.1-26
B 3.8.1-27
B 3.8.1-28
B 3.8.4-9
B 3.8.4-10

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 17, Electric Power Systems, states:

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.
>
> The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
>
> Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.
>
> Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18, Inspection and Testing of Electric Power Systems, states:

> Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Title 10 of the Code of Federal Regulations (10 CFR), 10CFR 50.36(c)(3), states:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulations do not specify details such as special Mode restrictions or allowing the crediting of unplanned events as satisfying a Surveillance Requirement. The proposed changes are consistent with NUREG-1431 and will continue to meet the intent of the General Design Criteria.

<u>No Significant Hazards Consideration Determination</u>

Duke Energy Carolinas, LLC ("Duke Energy") requests adoption of TSTF-283-A, Rev. 3, "Modify Section 3.8 Mode Restriction Notes", which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station, Units 1 and 2 Technical Specifications. The proposed changes revise Catawba Nuclear Station (CNS) Technical Specification (TS) 3.8.1, "AC Sources - Operating" and TS 3.8.4, "DC Sources - Operating", to allow certain Surveillance Requirements (SRs) to be performed, or portions thereof, in operating Modes which are currently restricted. The proposed changes reduce the potential for a plant shutdown should corrective maintenance (planned or unplanned) performed during power operation result in the need to perform certain Surveillance Requirements to demonstrate operability. The proposed changes also increase operational flexibility by including provisions to allow the crediting of unplanned events that satisfy the Surveillance Requirements.

Duke Energy 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 modify Mode restriction Notes in TS SRs 3.8.1.11, 3.8.1.16, 3.8.1.17, 3.8.1.19, 3.8.4.8, and 3.8.4.9 to allow performance of the Surveillance in whole or in part to reestablish Diesel Generator (DG) Operability, and to allow the crediting of unplanned events that satisfy the Surveillance Requirements. The emergency diesel generators and their associated emergency loads are accident mitigating features, and are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. To manage any increase in risk, the proposed changes require an assessment to verify that plant safety will be maintained or enhanced by performance of the Surveillance in the current prohibited Modes. The radiological consequences of an accident previously evaluated during the period that the DG is being tested to reestablish operability are no different from the radiological consequences of an accident previously evaluated while the DG is inoperable. As a result, the consequences of any accident previously evaluated are not increased. Therefore, the proposed changes do 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 changes do 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 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 purpose of Surveillances is to verify that equipment is capable of performing its assumed safety function. The proposed changes will only allow the performance of the Surveillances to reestablish operability, and the proposed changes may not be used to remove a DG from service. In addition, the proposed changes will potentially shorten the time that a DG is unavailable because testing to reestablish operability can be performed without a plant shutdown. The proposed changes also require an assessment to verify that plant safety will be maintained or enhanced by performance of the Surveillance in the current prohibited Modes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-285-A, Rev. 1, “Charging Pump Swap LTOP Allowance”

Description of Proposed Changes

The proposed changes affect CNS TS 3.4.12, “Low Temperature Overpressure Protection (LTOP) System”. Consistent with TSTF-285-A, the proposed changes increase the time allowed for swapping charging pumps to one hour, via addition of Note 1 to the Limiting Condition for Operation (LCO) section of the TS. Also, an existing Note in the Applicability section of the TS is reworded and relocated to the LCO section as Note 2.

Differences Between the Proposed Changes and the Approved Traveler

The Note to be deleted in Required Action B.1 in the TSTF-285-A markup of the ISTS (NUREG-1431) is not applicable to the CNS TS.

Summary of the Approved Traveler Justification

Inclusion of the existing Note in Required Action B.1, allowing 15 minutes to swap charging pumps is potentially confusing, given that the Completion Time for Condition B, “Two or more charging pumps capable of injecting into the Reactor Coolant System (RCS),” is “Immediately”. Furthermore, 15 minutes is insufficient time to prudently complete the operation of making the charging/makeup pump incapable of injection. Closing valves, or racking out the pump breaker requires appropriate administrative controls to be followed by Operations personnel. With proper diligence, these actions may not be safely accomplished in 15 minutes in all cases. One hour is reasonable considering the small likelihood of an event during this brief period and the other administrative controls available (e.g., operator action to stop any pump that inadvertently starts.) Therefore, the exception is reformatted as an LCO Note with a 1 hour allowance.

Additionally, the Applicability for LCO 3.4.12 is modified by a Note. This Note allows an exception to the LCO. Thus, it would be more appropriately located under the LCO. This Note was moved to the LCO, and renumbered as Note 2.

Pump swaps during LTOP conditions must take into account the restrictions of the LTOP analysis as well as the other required functions. In Mode 4, a charging pump is required to be operable to meet the ECCS requirements. The charging pump also is a part of charging and letdown to maintain RCS inventory and chemistry control. Further, securing charging for the purpose of not having more than the allowable pumps operable would also put thermal fatigue cycles on the piping and impact seal injection to the Reactor Coolant Pumps (RCPs), which has seal degradation potential. For these reasons it is desirable to have a provision to safely and deliberately swap pumps. In Modes 5 or 6, a charging pump is required for the necessary boration flowpath. This requirement has been relocated from the TS, but remains part of the Licensing Basis. While not as time critical as the Emergency Core Cooling System (ECCS) function, it is still required, and depending on plant status, the need for RCP seal injection may still be present. A time estimate for the charging pump swap performed by one Equipment Operator and one Reactor Operator was performed to confirm the requested time. This was an estimate starting with an open, racked out pump breaker on one pump and ending with an open, racked out, and properly surveilled pump breaker on the other

pump. The estimates clearly demonstrated 15 minutes to be inadequate to safely and deliberately complete the evolution. One hour is more appropriate, with the intent to minimize the actual time that more than [one] charging pump is physically capable of injection, which has been inserted into the Bases.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes beyond those described above. Note that the time estimate demonstration cited in TSTF-285-A (as described above) was not plant-specific to CNS, but is considered reasonable for CNS.

NRC Approval

The NRC documented their approval of TSTF-285-A, Revision 1, in a letter from William Beckner (NRC) to James Davis (NEI) dated May 12, 1999 (ADAMS Accession No. 9905180104).

Precedent

An example of a plant-specific NRC approval of the changes in TSTF-285-A is Watts Bar Nuclear Plant, Unit 1, Amendment Number 55, dated March 3, 2005 (ADAMS Accession No. ML050320368).

List of Affected Pages

3.4.12-1
B 3.4.12-6
B 3.4.12-8

Applicable Regulatory Requirements/Criteria

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G, “Fracture Toughness Requirements”.

The proposed changes are consistent with NUREG-1431 and will continue to meet 10 CFR 50 Appendix G.

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC (“Duke Energy”) requests adoption of TSTF-285-A, Rev. 1, “Charging Pump Swap LTOP Allowance”, which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station, Units 1 and 2 Technical Specifications (TS). The proposed changes revise Catawba Nuclear Station (CNS) Technical Specification (TS) 3.4.12, “Low Temperature Overpressure Protection (LTOP) System”, to allow two charging pumps the capability of injecting for one hour during pump swap operations, and makes several other associated administrative changes and clarifications to the TS.

Duke Energy 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 increase the time allowed for swapping charging pumps from 15 minutes to one hour, and make several other associated administrative changes and clarifications to the TS. These changes do not affect event initiators or precursors. Thus, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated. In addition, the proposed changes do not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the Updated Final Safety Analysis Report (UFSAR). As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of those SSCs in limiting the radiological consequences of previously evaluated accidents, and reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed changes do not involve a significant increase in the consequences of an accident 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 changes do not involve physical changes to analyzed SSCs or changes to the modes of plant operation defined in the technical specification. The proposed changes do not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor do they alter the design or operation of any plant systems. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new or different mode of failure has been created and no new or different equipment performance requirements are imposed for accident mitigation. As such, the proposed changes have no effect on previously evaluated accidents.

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

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

   Response: No.

   The proposed changes do not adversely affect any current plant safety margins or the reliability of the equipment assumed in the safety analysis. Therefore, there are no changes being made to any safety analysis assumptions, safety limits or limiting safety system settings that would adversely affect plant safety as a result of the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-315-A, Rev. 0, "Reduce Plant Trips Due to Spurious Signals to the NIS During Physics Testing"

Description of Proposed Changes

Consistent with TSTF-315-A, the proposed changes revise CNS TS 3.1.8, "PHYSICS TESTS Exceptions," to allow the number of channels required by LCO 3.3.1, "RTS Instrumentation," to be reduced from "4" to "3" to allow one nuclear instrumentation channel to be used as an input to the reactivity computer for physics testing without placing the nuclear instrumentation channel in a tripped condition.

Differences Between the Proposed Changes and the Approved Traveler

CNS TS 3.1.8, "PHYSICS TESTS Exceptions," is equivalent to TS 3.1.10, "PHYSICS TESTS Exceptions – MODE 2," in the ISTS markup included in TSTF-315-A.

TSTF-315-A, which is based on NUREG-1431 Rev. 1, identifies that a physics test exception is applicable to LCO 3.3.1 Functions 2, 3, 6, and 18.e. In NUREG-1431, Function 18.e of TS 3.3.1 Table 3.3.1-1, "Reactor Trip System Instrumentation," pertains to "Power Range Neutron Flux, P-10". Function 16.e is the corresponding function in TS Table 3.3.1-1 of the CNS TS. Therefore, the CNS TS is modified to apply to LCO 3.3.1 Functions 2, 3, 6, and 16.e.

Summary of the Approved Traveler Justification

During the performance of physics testing, one power range nuclear instrumentation channel is used to provide input to the reactivity computer. When this channel is used, the channel is usually placed in a tripped condition by removing the fuses to the electronics drawer. This effectively places the reactor trip logic in a one-out-of-three logic status. Any spurious signals received on one channel will result in a reactor trip. The proposed changes allow retaining the fuses in the nuclear instrumentation channel that is disconnected from the detector input. This would effectively place this channel in a bypass state and place the overall logic in a two-out-of-three logic status. This would preclude a spurious signal on one channel from causing a reactor trip.

During the performance of Physics Tests, the reactor power is restricted to less than 5 percent. In addition, the Nuclear Instrumentation System (NIS) trip setpoints are typically set to reduced values until after the core physics have been verified following a reload. Placing the NIS in a two-out-of-three logic versus a one-out-of-three logic reduces the risk of unnecessary plant trips during the performance of physics tests.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes.

NRC Approval

The NRC documented their approval of TSTF-315-A, Revision 0, in a letter from William D. Beckner (NRC) to James Davis (NEI), dated June 29,1999 (ADAMS Accession No. 9907060395).

Precedent

An example of a plant-specific NRC approval of the changes in TSTF-315-A is Joseph M. Farley Nuclear Plant, Units 1 and 2, Amendment Numbers 203/199, dated August 3, 2016 (ADAMS Accession No. ML15233A448).

List of Affected Pages

3.1.8-1
B 3.1.8-3

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 proposed changes do not affect the design of the plant instrumentation, but provides an operational allowance needed to safely perform required testing. There is no regulatory requirement that specifies what operational allowances should be included in the Technical Specifications. The allowance is consistent with the safety significance of the transitory condition. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC (“Duke Energy”) requests adoption of TSTF-315-A, Rev. 0, “Reduce Plant Trips Due to Spurious Signals to the NIS During Physics Testing”, which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station (CNS), Units 1 and 2 Technical Specifications (TS). The proposed changes revise CNS TS 3.1.8, “PHYSICS TESTS Exceptions” to allow the number of channels required by Limiting Condition for Operation (LCO) 3.3.1, “Reactor Trip System (RTS) Instrumentation,” to be reduced from “4” to “3”, to allow one nuclear instrumentation channel to be used as an input to the reactivity computer for physics testing without placing the nuclear instrumentation channel in a tripped condition. This change reduces the probability of a plant trip due to a spurious signal during PHYSICS TESTS.

Duke Energy 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 revise TS 3.1.8, “PHYSICS TESTS Exceptions,” to allow the number of channels required by LCO 3.3.1, “RTS Instrumentation,” to be reduced from “4” to “3”, to allow one nuclear instrumentation channel to be used as an input to the reactivity computer for physics testing without placing the nuclear instrumentation channel in a tripped condition. A reduction in the number of required nuclear instrumentation channels is not an initiator to any accident previously evaluated. With the nuclear instrumentation channel placed in bypass instead of in trip, reactor protection is still provided by the nuclear instrumentation system operating in a two-out-of-three channel logic. As a result, the ability to mitigate any accident previously evaluated is not significantly affected. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Therefore, the proposed changes do 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 changes do 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 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 reduce the probability of a spurious reactor trip during physics testing. The reactor trip system continues to be capable of protecting the reactor utilizing the power range neutron flux trips operating in a two-out-of-three trip logic. As a result, the reactor is protected and the probability of a spurious reactor trip is significantly reduced. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-340-A, Rev. 3, “Allow 7 Day Completion Time for a Turbine-Driven AFW Pump Inoperable”

Description of Proposed Changes

The proposed changes affect CNS TS 3.7.5, “Auxiliary Feedwater (AFW) System”. Condition A currently applies if one steam supply to the turbine driven AFW pump is inoperable, and allows a 7 day Completion Time to restore the steam supply. Under the proposed changes, consistent with TSTF-340-A, Condition A would be expanded to also include the situation of one turbine driven AFW pump inoperable in MODE 3, immediately following a refueling outage, if MODE 2 has not been entered. The same 7 day Completion Time would apply to restore the affected equipment.

Differences Between the Proposed Changes and the Approved Traveler

None

Summary of the Approved Traveler Justification

Present specifications have a 72 hour Completion Time for any inoperable AFW pump with an Action to be in MODE 4 within 18 hours if the 72 hour Completion Time is not met. The proposed changes 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.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes.

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 (ADAMS Accession No. ML003694199).

Precedent

Two examples of plant-specific NRC approvals of the changes in TSTF-340-A are Joseph M. Farley Nuclear Plant, Units 1 and 2, Amendment Numbers 203/199, dated August 3, 2016 (ADAMS Accession No. ML15233A448), and Vogtle Electric Generating Plant, Units 1 and 2, Amendment Numbers 180/161, dated June 9, 2016 (ADAMS Accession No. ML15132A569).

List of Affected Pages

3.7.5-1
B 3.7.5-5

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 changes make 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).

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC (“Duke Energy”) requests adoption of TSTF-340-A, Rev. 3, “Allow 7 Day Completion Time for a Turbine-Driven AFW Pump Inoperable”, which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station (CNS), Units 1 and 2 Technical Specifications. The proposed changes revise CNS Technical Specification (TS) 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. 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.

Duke Energy 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 revise TS 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. The proposed changes will not affect the source term, containment isolation, or radiological release

assumptions used in evaluating the radiological consequences of any accident previously evaluated. Therefore, the proposed changes do 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 changes do 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 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 TS 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 changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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-349-A, Rev. 1, "Add Note to LCO 3.9.5 Allowing Shutdown Cooling Loops Removal from Operation"[1]

TSTF-361-A, Rev. 2, "Allow Standby SDC/RHR/DHR Loop to be Inoperable to Support Testing"

TSTF-438-A, Rev. 0, "Clarify Exception Notes to be Consistent with the Requirement Being Excepted"

Description of Proposed Changes

The proposed changes affect CNS TS 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level". Consistent with TSTF-349-A and TSTF-438-A, Note 1 is proposed to be added to LCO 3.9.5 to allow the securing of the operating train of RHR for up to 15 minutes to support switching operating trains. The allowance is restricted to three conditions: a) the core outlet temperature is maintained greater than 10 degrees F below saturation temperature; b) no operations are permitted that would cause a introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1; and c) no draining operations to further reduce RCS water volume are permitted.

Consistent with TSTF-361-A, Note 2 is proposed to be added to LCO 3.9.5 to allow one required RHR loop to be inoperable for up to 2 hours for surveillance testing, provided that the other RHR loop is OPERABLE and in operation. LCO 3.9.5 currently requires two RHR loops to be OPERABLE, with one RHR loop in operation, and applies in Mode 6 with the water level less than 23 feet above the top of the reactor vessel flange.

Differences Between the Proposed Changes and the Approved Traveler

CNS TS 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level," is equivalent to TS 3.9.6, "RHR and Coolant Circulation – Low Water Level," in the ISTS markup included in TSTF-349-A, TSTF-361-A, and TSTF-438-A.

TSTF-349-A and TSTF-361-A include markups of NUREG-1431, Rev. 1. In this version of the NUREG there were no pre-existing notes applicable to LCO 3.9.5, and hence each new note is labeled as "Note" in the markup of the respective Traveler. The proposed changes to CNS TS 3.9.5 designate the new TSTF-349-A Note as "Note 1" and the new TSTF-361-A Note as "Note 2".

In the TSTF-349-A markup, the new Note reads: "All RHR pumps may be *de-energized*…" (Emphasis added). Consistent with TSTF-438-A, the wording of the new Note for CNS TS 3.9.5 is proposed to read: "All RHR pumps may be *removed from operation*…" (Emphasis added).

In the TSTF-349-A markup, the second condition of the new Note would read: "No operations are permitted that would cause a reduction of the Reactor Coolant System

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[1] The title of this Traveler refers to LCO 3.9.5. This Traveler applies to BWOG (NUREG-1430), WOG (NUREG-1431), and CEOG (NUREG-1432). For NUREG-1431, the affected LCO is LCO 3.9.6, not LCO 3.9.5.

boron concentration”. The wording of the new Note for CNS TS 3.9.5 is proposed to read: “No operations are permitted that would cause a introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1”. This proposed wording is consistent with the current version of the ISTS, NUREG-1431, Rev. 4, and is also consistent with the intent of TSTF-286-A, “Define ‘Operations Involving Positive Reactivity Additions’”.

The scope of TSTF-438-A addresses similar wording in several different ISTS LCOs. Based on a review of the corresponding CNS LCOs, only a change to LCO 3.9.5 is considered necessary.

Summary of the Approved Traveler Justification

TSTF-349-A, Rev. 1, inadvertently omitted a discussion of the justification for the Traveler. However, an earlier version, TSTF-349, Rev. 0, stated that securing of the operating train to support switching from one train to another should be permissible, as the allowed time frame is short, and limitations are in place to preclude RCS boron reduction and draining activities.

From TSTF-361-A, Rev 2: LCO 3.9.5 currently does not allow the non-operating shutdown cooling (SDC) loop to be made inoperable to support surveillance testing. LCOs 3.4.7 and 3.4.8 both allow the non-operating SDC loop to be inoperable for a period of up to 2 hours to perform surveillance testing, provided the other SDC loop is OPERABLE and operating. The Note is needed in LCO 3.9.5 to provide the flexibility to perform surveillance testing while ensuring that there is reasonable time for operators to respond to and mitigate any expected failures. Therefore, for consistency, and to support required outage activities and still maintain the plant in a safe condition, this Note should be added to LCO 3.9.5.

From TSTF-438-A, Rev 0: The wording of the Note added in TSTF-349-A is intended to convey that a temporary exception to the requirement to have a cooling loop in operation, is allowed. Re-phrasing the verbiage more clearly conveys the intent. The proposed change is editorial.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification

The justification presented in the three Travelers, as summarized above, is applicable to CNS. The Travelers are being adopted by CNS with no significant changes beyond those described above.

NRC Approval

The NRC did not issue a letter approving TSTF-349, but it was incorporated by the NRC into Revision 2 of the ISTS.

TSTF-361 was approved by letter from W. D. Beckner (NRC) to A. R. Pietrangelo (NEI) dated October 31, 2000 (ADAMS Accession No. ML003765449).

TSTF-438 was approved by letter from R. L. Dennig (NRC) to A. R. Pietrangelo (NEI), dated October 21, 2002 (ADAMS Accession No. ML022940574).

Precedent

Two examples of plant specific NRC approvals of the changes in TSTF-349-A are Joseph M. Farley Nuclear Plant, Units 1 and 2, Amendment Numbers 203/199, dated August 3, 2016 (ADAMS Accession No. ML15233A448), and Vogtle Electric Generating Plant, Units 1 and 2, Amendment Numbers 180/161, dated June 9, 2016 (ADAMS Accession No. ML15132A569).

An example of a plant specific NRC approval of the changes in TSTF-361-A is Vogtle Electric Generating Plant, Units 1 and 2, Amendment Numbers 163 and 145, dated November 9, 2011 (ADAMS Accession No. ML11256A181).

An example of a plant specific NRC approval of the changes in TSTF-438-A is Beaver Valley Power Station, Units 1 and 2, Amendment Numbers 265 and 146, dated March 11, 2005 (ADAMS Accession No. ML050600094).

List of Affected Pages

3.9.5-1
B 3.9.5-1

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 includes specifics on the content of the Limiting Condition for Operation (LCO), such as what operational allowances should be included. The proposed changes add operational allowances associated with operating RHR pumps, consistent with the safety significance of the transitory condition and consistent with similar LCOs in the CNS Technical Specifications. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

The proposed change associated with TSTF-438-A is an administrative clarification which does not affect any regulatory requirements or criteria.

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC (“Duke Energy”) requests adoption of three Technical Specification Task Force (TSTF) Travelers applicable to CNS Technical Specification (TS) 3.9.5, “Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level”. Each of the three Travelers is an approved change to the standard technical specifications (STS). In accordance with TSTF-349-A, Rev. 1, “Add Note to LCO 3.9.5

Allowing Shutdown Cooling Loops Removal from Operation", and TSTF-438-A, Rev. 0, "Clarify Exception Notes to be Consistent with the Requirement Being Excepted", a note is proposed to be added to CNS TS Limiting Condition for Operation (LCO) 3.9.5 to allow the securing of the operating train of RHR for up to 15 minutes to support switching operating trains. In accordance with TSTF-361-A, Rev. 2, "Allow Standby SDC/RHR/DHR Loop to be Inoperable to Support Testing", a second note is proposed to be added to CNS TS LCO 3.9.5 to allow one RHR loop to be inoperable for up to 2 hours for surveillance testing, provided the other RHR loop is operable and in operation.

Duke Energy 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 add two notes to CNS TS LCO 3.9.5. Note 1 would allow securing the operating train of Residual Heat Removal (RHR) for up to 15 minutes to support switching operating trains, subject to certain restrictions. Note 2 to would allow one RHR loop to be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is Operable and in operation. These provisions are operational allowances. Neither operational allowance is an initiator to any accident previously evaluated. In addition, the proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Therefore, the proposed changes do 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 changes do 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 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.

   An operational allowance is proposed which would allow securing the operating train of RHR for up to 15 minutes to support switching operating trains, subject to certain restrictions. Considering 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.

An operational allowance is also proposed which would allow one RHR loop to be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is operable and in operation. A similar allowance currently appears in CNS TS 3.4.7, “Reactor Coolant System (RCS) Loops – MODE 5, Loops Filled,” and CNS TS 3.4.8, “RCS Loops – MODE 5, Loops Not Filled,” and the conditions under which the operational allowance would be applied in TS 3.9.5 are not significantly different from those specifications. This operational allowance provides the flexibility to perform surveillance testing, while ensuring that there is reasonable time for operators to respond to and mitigate any expected failures. The purpose of the RHR System is to remove decay and sensible heat from the RCS, to provide mixing of borated coolant, and to prevent boron stratification. Removal of system components from service as described above, and with limitations in place to maintain the ability of the RHR System to perform its safety function, does not significantly impact the margin of safety. Operators will continue to have adequate time to respond to any off-normal events. Removing the system from service, for a limited period of time, with other operational restrictions, limits the consequences to those already assumed in the Updated Final Safety Analysis Report (UFSAR).

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above, Duke Energy 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-352-A, Rev. 1, “Provide Consistent Completion Time to Reach MODE 4”

Description of Proposed Changes

The proposed changes affect CNS TS 3.4.10, “Pressurizer Safety Valves,” CNS TS 3.7.4, “Steam Generator Power Operated Relief Valves (SG PORVs),” and CNS TS 3.7.6, “Condensate Storage System (CSS),” as follows:

- The Completion Time for Limiting Condition for Operation (LCO) 3.4.10 Required Action B.2, “Be in MODE 4 with any RCS cold leg temperatures ≤ 210 ºF”, is proposed to be changed from 12 hours to 24 hours.

- The Completion Time for LCO 3.7.4 Required Action C.2, “Be in MODE 4 without reliance upon steam generator for heat removal”, is proposed to be changed from 12 hours to 24 hours.

- The Completion time for LCO 3.7.6 Required Action B.2, “Be in MODE 4 without reliance upon steam generator for heat removal”, is proposed to be changed from 12 hours to 24 hours.

Differences Between the Proposed Changes and the Approved Traveler

CNS TS 3.7.4, “Steam Generator Power Operated Relief Valves (SG PORVs),” is equivalent to TS 3.7.4, “Atmospheric Dump Valves (ADVs),” in the ISTS. CNS TS 3.7.6, “Condensate Storage System (CSS),” is equivalent to TS 3.7.6, “Condensate Storage Tank (CST),” in the ISTS.

The change to ISTS TS 3.4.12, “Low Temperature Overpressure Protection (LTOP) System,” approved in TSTF-352-A, is not applicable to the CNS TS. The Completion Time for LCO 3.4.12 Required Action G.2, “Depressurize RCS and establish RCS vent of ≥ 4.5 square inches”, is already at 12 hours.

Summary of the Approved Traveler Justification

The following presents the allowed times for attaining various shutdown conditions from MODE 1 normal operating pressure and temperature (and assuming the required equipment is available):

MODE	Completion Time
1	…
3	6 hours
4	12 hours
5	36 hours

However, many Required Actions specify entry into conditions which take longer to achieve than MODE 4, but only allow the normal 12 hours to enter those conditions from MODE 3. This leaves insufficient time to enter MODE 4 and the required condition in a controlled manner. Therefore, for those conditions, the Completion Time is revised to 24 hours (half way between the MODE 4 and MODE 5 Completion Times) to provide a

consistent, adequate Completion Time. The affected Conditions are discussed individually below:

- Specification 3.4.10, Pressurizer Safety Valves, Condition B, requires the plant to be in MODE 4 and cooled down to the LTOP enable temperature within [12] hours. Assuming the LTOP enable temperature is below the MODE 4 entry conditions, additional time should be provided beyond the normal 12 hours allowed to reach MODE 4.

- Specification 3.7.4, Atmospheric Dump Valves, Condition C, requires the plant to be in MODE 4 without reliance on the SG for heat removal within [12] hours. This requires cooling to shutdown cooling entry conditions, which for many designs is below the MODE 4 entry temperature. In order to meet this Completion Time, many plants would have to start shutdown before the restoration period, allowed by conditions A or B, was up. Therefore, using the logic presented above, the Completion Time is revised to 24 hours for going from MODE 1 to MODE 4 without reliance on SG for heat removal. The additional time is needed for the cooldown and depressurization of the RCS to shutdown cooling (SDC) entry conditions.

- Specification 3.4.10, LTOP, Condition G, requires the plant to depressurize the RCS and establish a vent of [ ] inches within 8 hours. Eight hours is insufficient time to plan a MODE change, cool down (following the plant cooldown rate limits), plan and execute the maintenance activity of opening a vent, and cool the RCS sufficiently to safely open a vent. The proposed 12 hours is more appropriate.

<u>Differences Between the Plant-Specific Justification and the Approved Traveler Justification</u>

The justification presented in the approved Traveler is applicable to CNS. The Traveler is being adopted by CNS with no significant changes beyond those described above. As noted above, CNS TS 3.7.4, “Steam Generator Power Operated Relief Valves (SG PORVs),” is equivalent to TS 3.7.4, “Atmospheric Dump Valves (ADVs),” in the ISTS.

<u>NRC Approval</u>

The NRC did not issue a letter approving this Traveler, but it was incorporated by the NRC into Revision 2 of the ISTS.

<u>Precedent</u>

An example of a plant specific NRC approval of the changes in TSTF-352-A is Callaway Plant, Unit 1, Amendment No. 188, dated November 25, 2008 (ADAMS Accession No. ML082910895).

<u>List of Affected Pages</u>

3.4.10-1
3.7.4-1
3.7.6-1

B 3.4.10-4
B 3.7.4-3
B 3.7.6-3

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 remedial actions in the TSs are specified in terms of LCO Conditions, Required Actions, and Completion Times (CTs). When an LCO is not being met, the CTs specified in the TSs are the time allowed in the TSs for completing the specified required actions. The conditions and required actions specified in the TSs must be acceptable remedial actions for the LCO not being met, and the CTs must be a reasonable time for completing the required actions while maintaining the safe operation of the plant. The proposed changes allow a more reasonable time to plan and execute required actions. No systems or components are being changed by the proposed amendment; only the remedial actions for inoperable systems or components are being changed. The proposed changes are consistent with the ISTS for Westinghouse Plants (NUREG-1431).

No Significant Hazards Consideration Determination

Duke Energy Carolinas, LLC ("Duke Energy") requests adoption of TSTF-352-A, Rev. 1, "Provide Consistent Completion Time to Reach MODE 4", which is an approved change to the standard technical specifications (STS), into the Catawba Nuclear Station (CNS), Units 1 and 2 Technical Specifications (TSs). The proposed changes affect CNS TS 3.4.10, "Pressurizer Safety Valves," CNS TS 3.7.4, "Steam Generator Power Operated Relief Valves (SG PORVs)," and CNS TS 3.7.6, "Condensate Storage System (CSS)," as follows:

- The Completion Time for Limiting Condition for Operation (LCO) 3.4.10 Required Action B.2, "Be in MODE 4 with any RCS cold leg temperatures ≤ 210 ºF", is proposed to be changed from 12 hours to 24 hours.

- The Completion Time for LCO 3.7.4 Required Action C.2, "Be in MODE 4 without reliance upon steam generator for heat removal", is proposed to be changed from 12 hours to 24 hours.

- The Completion time for LCO 3.7.6 Required Action B.2, "Be in MODE 4 without reliance upon steam generator for heat removal", is proposed to be changed from 12 hours to 24 hours.

These changes will allow a more reasonable time to plan and safely execute the required actions.

Duke Energy 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 allow a more reasonable time to plan and execute required actions, and will not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not physically alter safety-related systems nor affect the way in which safety-related systems perform their functions. All accident analysis acceptance criteria will continue to be met with the proposed changes. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the CNS Updated Final Safety Analysis Report (UFSAR). The applicable radiological dose acceptance criteria will continue to be met.

   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.

   There are no proposed design changes nor are there any changes in the method by which any safety-related plant SSC performs its safety function. The proposed changes will not affect the normal method of plant operation or change any operating parameters. No equipment performance requirements will be affected. The proposed changes will not alter any assumptions made in the safety analyses.

   No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

   Therefore, the proposed changes do not create the possibility of a new or different accident from any accident previously evaluated.

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

   Response: No.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their intended functions. These barriers include the fuel cladding, the reactor coolant system pressure boundary, and the containment barriers. The proposed changes will not have any impact on these barriers. No accident mitigating equipment will be adversely impacted. Therefore, existing safety margins will be preserved. None of the proposed changes will involve a significant reduction in a margin of safety.

Based on the above, Duke Energy 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 Conclusions

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 changes will not be inimical to the common defense and security or to the health and safety of the public.

4.0 Environmental Consideration

The proposed changes would modify requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

Enclosure 2

Proposed Technical Specification Changes (Mark-up)

Index of Affected TS Pages vs. Traveler Number

Page	Traveler(s)
3.1.2-1	TSTF-142
3.1.8-1	TSTF-315
3.4.10-1	TSTF-352
3.4.12-1	TSTF-285
3.6.3-2	TSTF-269
3.6.3-3	TSTF-269
3.6.3-4	TSTF-269
3.7.4-1	TSTF-352

Page	Traveler(s)
3.7.5-1	TSTF-340
3.7.6-1	TSTF-352
3.8.1-9	TSTF-283
3.8.1-12	TSTF-283
3.8.1-13	TSTF-283
3.8.1-14	TSTF-283
3.8.4-3	TSTF-283
3.8.4-4	TSTF-283
3.9.4-2	TSTF-197
3.9.5-1	TSTF-349 TSTF-361 TSTF-438
3.9.5-2	TSTF-197

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 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation. AND A.2 Establish appropriate operating restrictions and SRs.	~~72 hours~~ 7 days ~~72 hours~~ 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. RCS lowest loop average temperature is ≥ 541°F; and

b. SDM is within the limit specified in the COLR.

and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels,

APPLICABILITY: MODE 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	AND	
	A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings $\geq$ 2435 psig and $\leq$ 2559 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 210°F.

--NOTE--
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperatures $\leq$ 210°F.	6 hours ~~12~~ 24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of two pumps (charging pumps, safety injection pumps, or charging and safety injection pumps) capable of injecting into the RCS, the accumulators isolated, reactor coolant pump operation limited as specified in Table 3.4.12-1 and either a, b, or c below:

a. Two power operated relief valves (PORVs) with nominal lift setting = 400 psig (as left calibrated), allowable value ≤ 425 psig (as found), with RCS cold leg temperature ≥ 70°F; or

b. Two residual heat removal (RHR) suction relief valves with lift settings ≥ 417 psig and ≤ 509 psig with an indicated RCS cold leg temperature ≥ 70°F; or

c. A combination of any one PORV and one RHR suction relief valve, each with lift settings as described above.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 210°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

--NOTES--

2. Accumulator may be unisolated ~~isolation is only required~~ when accumulator pressure is less than ~~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 Specification 3.4.3.

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1. Two charging pumps may be made capable of injecting for ≤ 1 hour for pump swap operations.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured 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. ------------NOTE------------ Only applicable to penetration flow paths with two containment isolation valves. ------------------------------- One or more penetration flow paths with two containment isolation valves inoperable except for purge valve or reactor building bypass 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. ------------NOTE------------ Only applicable to penetration flow paths with only one containment isolation valve and a closed system. -------------------------------	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.	72 hours
	AND	
One or more penetration flow paths with one containment isolation valve inoperable.	C.2 ------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means ----------------------------------	
	Verify the affected penetration flow path is isolated.	Once per 31 days
D. Reactor building bypass leakage not within limit.	D.1 Restore leakage within limit.	4 hours
E. One or more penetration flow paths with one or more containment purge, hydrogen purge, or containment air release and addition valves not within leakage limits.	E.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	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.2 -------------NOTES------------- 1. 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
	<u>AND</u>	
	E.3 Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action E.1.	Once per 92 days
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	F.2 Be in MODE 5.	36 hours

2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

3.7 PLANT SYSTEMS

3.7.4 Steam Generator Power Operated Relief Valves (SG PORVs)

LCO 3.7.4 Four SG PORV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SG PORV line inoperable.	A.1 Restore SG PORV line to OPERABLE status.	7 days
B. Two or more SG PORV lines inoperable.	B.1 Restore all but one SG PORV line to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours ~~12~~ 24 hours

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

-----NOTE-----
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore ~~steam supply~~ affected equipment to OPERABLE status.	7 days __AND__ 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours __AND__ 10 days from discovery of failure to meet the LCO

(continued)

Insert TS 3.7.5 Condition A

<u>INSERT TS 3.7.5 Condition A</u>

<u>OR</u>

--------------------- NOTE --------------------
Only applicable if MODE 2 has not
been entered following refueling.

One turbine driven AFW pump inoperable
in MODE 3 following refueling.

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage System (CSS)

LCO 3.7.6 The CSS inventory shall be $\geq$ 225,000 gal.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CSS inventory not within limit.	A.1 Verify by administrative means OPERABILITY of assured water supply.	4 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore CSS inventory to within limit.	7 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, without reliance on steam generator for heat removal.	~~12~~ hours 24

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. -------------------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes the emergency bus in $\leq$ 11 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage $\geq$ 3950 V and $\leq$ 4580 V, 4. maintains steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies auto-connected shutdown loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	-----------------------------NOTES----------------------------- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 1 hour loaded ≥ 5600 kW and ≤ 5750 kW or until operating temperature is stabilized. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. --	
	Verify each DG starts and achieves, in ≤ 11 seconds, voltage ≥ 3950 V, and frequency ≥ 57 Hz and maintains steady state voltage ≥ 3950 V and ≤ 4580 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.16	-----------------------------NOTE----------------------------- This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. --	
	Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to standby operation.	In accordance with the Surveillance Frequency Control Program

(continued)

However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	-------------------------------NOTE------------------------------- This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to standby operation; and b. Automatically energizing the emergency load from offsite power.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.18	Verify interval between each sequenced load block is within the design interval for each automatic load sequencer.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	-------------------------------NOTES------------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. -- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes the emergency bus in $\leq$ 11 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage $\geq$ 3950 V and $\leq$ 4580 V, 4. achieves steady state frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz, and 5. supplies auto-connected emergency loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.4	Verify DC channel and DG battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Remove visible terminal corrosion, verify DC channel and DG battery cell to cell and terminal connections are clean and tight, and are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	Verify all DC channel and DG battery connection resistance values meet Table 3.8.4-1 limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.7	Verify each DC channel battery charger supplies ≥ 200 amps and the DG battery charger supplies ≥ 75 amps with each charger at ≥ 125 V for ≥ 8 hours.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.8	-------------------------------NOTES------------------------------ 1. The modified performance discharge test in SR 3.8.4.9 may be performed in lieu of the service test in SR 3.8.4.8. normally 2. This Surveillance shall not be performed for the DG batteries in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. -- Verify DC channel and DG battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	-----------------------------NOTE---------------------------- This Surveillance shall not normally be performed for the DG batteries in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. -- Verify DC channel and DG battery capacity is $\geq$ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program AND 18 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating AND --------NOTE-------- Not applicable to DG batteries ----------------------- 24 months when battery has reached 85% of the expected life with capacity $\geq$ 100% of manufacturer's rating

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 ~~Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.~~	~~4 hours~~

INSERT T.S. 3.9.4 Condition A.4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1000 gpm and RCS temperature is ≤ 140°F.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.2	Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

INSERT TS 3.9.4 Condition A.4

REQUIRED ACTION		COMPLETION TIME
A.4	Close the containment equipment hatch and secure with four bolts.	4 hours
AND		
A.5	Close one door in each air lock.	4 hours
AND		
A.6.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
OR		
A.6.2	Verify each penetration is capable of being closed on a high containment radiation signal.	4 hours

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level

LCO 3.9.5 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

INSERT TS 3.9.5 Notes

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
	<u>OR</u>	
	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 that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
		(continued)

INSERT TS 3.9.5 Notes

-- NOTES --

1. All RHR pumps may be removed from operation 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 introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1, and

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one RHR loop to operation. AND ~~B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.~~	Immediately ~~4 hours~~

INSERT TS 3.9.5 Condition B.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1000 gpm and RCS temperature is ≤ 140°F.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.3	Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

INSERT TS 3.9.5 Condition B.3

REQUIRED ACTION		COMPLETION TIME
B.3	Close the containment equipment hatch and secure with four bolts.	4 hours
AND		
B.4	Close one door in each air lock.	4 hours
AND		
B.5.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
OR		
B.5.2	Verify each penetration is capable of being closed on a high containment radiation signal.	4 hours

Enclosure 3

Proposed Technical Specification Bases Changes (Mark-up)
(For Information Only)

Index of Affected TS Bases Pages vs. Traveler Number

Page	Traveler(s)
B 3.1.2-4	TSTF-142
B 3.1.8-3	TSTF-315
B 3.4.10-4	TSTF-352
B 3.4.12-6	TSTF-285
B 3.4.12-8	TSTF-285
B 3.6.3-7	TSTF-269
B 3.6.3-8	TSTF-269
B 3.6.3-10	TSTF-269
B 3.7.4-3	TSTF-352

Page	Traveler(s)
B 3.7.5-5	TSTF-340
B 3.7.6-3	TSTF-352
B 3.8.1-23	TSTF-283
B 3.8.1-26	TSTF-283
B 3.8.1-27	TSTF-283
B 3.8.1-28	TSTF-283
B 3.8.4-9	TSTF-283
B 3.8.4-10	TSTF-283
B 3.9.4-4	TSTF-197
B 3.9.5-1	TSTF-349 TSTF-361 TSTF-438
B 3.9.5-3	TSTF-197

APPLICABILITY (continued)

and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated 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.

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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (Ref.6).

Reference 7 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from "4" to "3".

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 541°F; and

b. SDM is within limit specified in the COLR.

, and the number of required channels for LCO 3.3.1, "RTS Instrumentation", Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels,

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

BASES

ACTIONS (continued)

B.1 and B.2

24

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ 210°F 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. With any RCS cold leg temperatures at or below 210°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3% and -2% of the nominal setpoint of 2485 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Chapter 15.
3. UFSAR, Section 5.2.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 417 psig and 509 psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ 10% of the rated lift setpoint.

Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR system does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.

The RHR suction relief valves are considered to be active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO permits a maximum of two pumps (charging and/or safety injection) capable of injecting into the RCS and requires all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3. The LCO also limits RCP operation based on existing RCS cold leg temperature as required by the LTOP analysis.

INSERT B 3.4.12 (new paragraph)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

INSERT B 3.4.12

The LCO is modified by two Notes. Note 1 allows two charging pumps to be made capable of injecting for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and surveillance requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

BASES

APPLICABILITY (continued)

operator action to mitigate the event.

~~The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.~~

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP system. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, 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

With more than two pumps (charging and/or safety injection) capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1

With RCP operation not limited in accordance with Table 3.4.12-1, RCS overpressurization is possible.

To immediately initiate action to limit pump operation reflects the urgency of removing the RCS from this condition.

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS

BASES

ACTIONS (continued)

isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

two Notes. Note 1

Required Action A.2 is modified by ~~a Note that~~ applies to isolation devices 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. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

INSERT B 3.6.3 Action A.2

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

BASES

ACTIONS (continued)

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 4. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

two Notes. Note 1

Required Action C.2 is modified by ~~a Note that~~ 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. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

INSERT B 3.6.3 Action C.2

BASES

ACTIONS (continued)

For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

INSERT B 3.6.3 ACTION E.2 (new paragraph)

For the valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.3.6 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, 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 5 within 36 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each containment purge supply and exhaust isolation valve for the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System is required to be verified sealed closed. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of these valves to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses has not been performed. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness.

INSERT B 3.6.3 ACTION A.2

Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

INSERT B 3.6.3 ACTION C.2

Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

INSERT B 3.6.3 ACTION E.2

Required Action E.2 is modified by two Notes. Note 1 applies to isolation devices 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. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

BASES

LCO (continued)

An SG PORV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand using the nitrogen gas supply.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the SG PORVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one SG PORV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE SG PORV lines, a nonsafety grade backup in the Steam Dump System, and MSSVs.

B.1

With two or more SG PORV lines inoperable, action must be taken to restore all but one SG PORV line to OPERABLE status. Since the block valve can be closed to isolate an SG PORV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable SG PORV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the SG PORV lines.

C.1 and C.2

If the SG PORV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within ~~12~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

24

BASES

ACTIONS (continued)

be applied in this circumstance.

A.1

INSERT B 3.7.5 Action A.1 (Part 1)

~~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:~~

~~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.~~

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

INSERT B 3.7.5 Action A.1 (Part 2) (new paragraph)

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 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 AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

INSERT B 3.7.5 Action A.1 (Part 1)

If one of the two steam supplies to the turbine driven AFW train is inoperable, or if a turbine driven pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

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.

INSERT B 3.7.5 Action A.1 (Part 2)

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 the turbine-driven AFW pump 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

ACTIONS (continued)

MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within ~~12~~ 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSS contains the required inventory of cooling water. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 10.4.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.

SURVEILLANCE REQUIREMENTS (continued)

The requirement to verify the connection and power supply of the emergency bus and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

INSERT B 3.8.1 SR 3.8.1.11

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (11 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d ensures that the emergency bus remains energized from the offsite electrical power system on an ESF signal without loss of offsite power.

SURVEILLANCE REQUIREMENTS (continued)

avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least an hour at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 10), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to standby operation when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in standby operation when the DG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

INSERT B 3.8.1 SR 3.8.1.16

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to standby operation if a LOCA actuation signal is received during operation in the test mode. Standby operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by Regulatory Guide 1.9 (Ref. 3).

SURVEILLANCE REQUIREMENTS (continued)

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

INSERT B 3.8.1 SR 3.8.1.17

SR 3.8.1.18

Under accident and loss of offsite power conditions loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The load sequence time interval tolerance in Table 8-6 of Reference 2 ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Table 8-6 of Reference 2 provides a summary of the automatic loading of ESF buses.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

INSERT B 3.8.1 SR 3.8.1.19

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a performance discharge test that is augmented to include the high-rate, short duration discharge loads (during the first minute and 11-to-12 minute discharge periods) of the service test. The duty cycle of the modified performance test must fully envelope the duty cycle of the service test if the modified performance discharge test is to be used in lieu of the service test. Since the ampere-hours removed by the high-rate, short duration discharge periods of the service test represents a very small portion of the battery capacity, the test rate can be changed to that for the modified performance discharge test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rates of the duty cycle). This will often confirm the battery's ability to meet the critical periods of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

INSERT B 3.8.4 SR 3.8.4.8

SR 3.8.4.9

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.8. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.9; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.9 while satisfying the requirements of SR 3.8.4.8 at the same time.

SURVEILLANCE REQUIREMENTS (continued)

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9). This reference recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 18 months. However (for DC vital batteries only), if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its average capacity on the previous performance tests or when it is ≥ 10% below the manufacturer's rating. This SR is modified by a Note which is applicable to the DG batteries only. The reason for the Note is that performing the Surveillance would perturb the associated electrical distribution system and challenge safety systems.

INSERT B 3.8.4 SR 3.8.4.9

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1971 and 1974.
4. UFSAR, Chapter 8.
5. IEEE-485-1983, June 1983.
6. UFSAR, Chapter 6.
7. UFSAR, Chapter 15.
8. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
9. IEEE-450-1975 and/or 1980.
10. Regulatory Guide 1.32, February 1977.
11. IEEE-450-1995.

INSERT B 3.8.1 SR 3.8.1.16

This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, at a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

INSERT B 3.8.1 SR 3.8.1.11
INSERT B 3.8.1 SR 3.8.1.17
INSERT B 3.8.1 SR 3.8.1.19
INSERT B 3.8.4 SR 3.8.4.8
INSERT B 3.8.4 SR 3.8.4.9

This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, at a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

BASES

ACTIONS (continued)

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

, A.5, A.6.1, and A.6.2

A.4

INSERT B3.9.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The RCS temperature is determined to ensure the appropriate decay heat removal is maintained. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 5.5.7.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management."

INSERT B 3.9.4

When RHR loop requirements are not met, the following actions must be taken:

a. The containment equipment hatch must be closed and secured with four bolts,
b. One door in each air lock must be closed, and
c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed on a high containment radiation signal.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant and component cooling water through the RHR heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE.

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.

INSERT B 3.9.5 LCO (new paragraph)

INSERT B 3.9.5 LCO

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation 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 maintained > 10 degrees F below saturation temperature. Note 1 also prohibits boron dilution or draining operations when RHR forced flow is stopped. Note 2 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.

BASES

ACTIONS (continued)

concentration greater than that which would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2

INSERT B 3.9.5

~~If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded. The Completion Time of 4 hours is appropriate for the majority of time during refueling operations, based on time to coolant boiling, since water level is not routinely maintained at low levels.~~

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability, prevent vortexing in the suction of the RHR pumps, and to prevent thermal and boron stratification in the core. The RCS temperature is determined to ensure the appropriate decay heat removal is maintained. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

INSERT B 3.9.5

If no RHR is in operation, the following actions must be taken:

a. The containment equipment hatch must be closed and secured with four bolts,
b. One door in each air lock must be closed, and
c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed on a high containment radiation signal.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.