J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672

o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

RA-19-0039 10 CFR 50.90

April 13, 2020

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

Duke Energy Carolinas, LLC
Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55

Subject: License Amendment Request to Revise Technical Specifications to Adopt Two NRC-Approved Generic Technical Specification Changes

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend the Technical Specifications (TS) for Oconee Nuclear Station (ONS) Units 1, 2, and 3. The proposed changes are associated with the following two NRC-approved Technical Specification Task Force (TSTF) Travelers:

- TSTF-272-A, Rev. 1, Refueling Boron Concentration Clarification; and
- TSTF-421-A, Rev. 0, Revision to RCP Flywheel Inspection Program (WCAP-15666).

The Enclosure to this letter provides the basis for the proposed TS changes, a No Significant Hazards Consideration and Environmental Consideration. Attachments 1 and 2 to the Enclosure provide marked-up TS and TS Bases pages, respectively. Attachment 3 provides retyped (clean) TS pages. The marked-up TS Bases is provided for information only.

In accordance with Duke Energy administrative procedures that implement the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the On-site Review Committee. A copy of this LAR is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Upon NRC approval, the amendment shall be implemented within 120 days.

There are no regulatory commitments being made as a result of the proposed change.

Inquiries on this proposed amendment request should be directed to Mr. Arthur Zaremba, Fleet Nuclear Licensing Manager, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 13, 2020.

Sincerely,

J. Ed Burchfield, Jr.

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Enclosure: Evaluation of Proposed Changes

Attachments:

1 Marked-Up Technical Specifications Pages
2 Marked-Up Technical Specifications Bases Pages
3 Retyped Technical Specifications Pages

cc w/enclosure and attachments:

Ms. Laura Dudes, Administrator, Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Mr. Michael Mahoney, Project Manager (by electronic mail only)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

Mr. Jared Nadel (by electronic mail only)
NRC Senior Resident Inspector
Oconee Nuclear Station

Ms. Anuradha Nair-Gimmi (by electronic mail only: naira@dhec.sc.gov)
Department of Health & Environmental Control
Bureau of Environmental Health Services
2600 Bull Street
Columbia, SC 29201

ENCLOSURE
EVALUATION OF PROPOSED CHANGE

Subject: License Amendment Request to Revise Technical Specifications to Adopt Two NRC-Approved Generic Technical Specification Changes

1 SUMMARY DESCRIPTION

This License Amendment Request (LAR) proposes to amend the Technical Specifications (TS) for Oconee Nuclear Station (ONS) Units 1, 2, and 3 to adopt the following two NRC-approved Technical Specification Task Force (TSTF) Travelers:

- TSTF-272-A, Rev. 1, Refueling Boron Concentration Clarification; and
- TSTF-421-A, Rev. 0, Revision to RCP Flywheel Inspection Program (WCAP-15666).

It is recognized that NRC approval of TSTF-421-A is applicable to Westinghouse designed plants. In the NRC approval of this TSTF published in the Federal Register, NRC staff recognized that the model safety evaluation and some of the supporting material may be applicable to some B&W units since some of the data was provided by B&W units. ONS was one of those B&W units that provided data. This LAR demonstrates that sufficient basis exists for NRC approval of a TS change identical to that described in TSTF-421-A.

2 DETAILED DESCRIPTION, TECHNICAL EVALUATION, REGULATORY EVALUATION

Each Traveler is discussed in an individual analysis provided in Sections 2.1 and 2.2. Each of these sections contains the following topics:

Description of Proposed Change - This topic describes the effect of adopting the TS changes of the subject Traveler in the ONS Technical Specifications.

Differences Between the Proposed Change and the Approved Traveler - This topic describes differences between the changes proposed to the ONS Technical Specifications and the Standard Technical Specification (STS) mark-ups provided in the approved Traveler.

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

Differences Between the Plant-Specific Justification and the Approved Traveler Justification - This topic describes any differences between the Traveler justification utilized by the NRC when approving the Traveler and the justification for adopting the Traveler in the ONS TS.

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

NRC Approval - This topic references the NRC letter, if any, approving the Traveler. It also provides at least one example of an NRC approval of a plant-specific request to adopt the Traveler.

List of Affected Pages - This topic lists the ONS 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 - 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."

Marked-up TS pages are provided in Attachment 1.
Retyped TS pages are provided in Attachment 3.
Mark-ups of affected TS Bases pages are included in Attachment 2. The TS Bases mark-ups are provided for information only and will be revised under the Technical Specification Bases Control Program following NRC approval of the proposed TS changes.

2.1 TSTF-272-A, Revision 1, Refueling Boron Concentration Clarification

Description of Proposed Change
The proposed change adds an Applicability Note to TS 3.9.1, "Boron Concentration." The note clarifies that the TS Limiting Condition for Operation (LCO) only applies to the refueling canal when that volume is connected to the reactor coolant system (RCS).

Differences Between the Proposed Change and the Approved Traveler
ONS converted to the NUREG-1430 Standard Technical Specifications (STS) via NRC-approved license amendment dated 12/16/1998. This conversion included the following deviations from the STS 3.9.1 LCO wording:

1. The ONS TS 3.9.1 LCO wording does not include the term "refueling cavity." This deviation was taken because the ONS design does not identify a refueling cavity; the term "refueling canal" includes the area typically referred to as the "refueling cavity."

2. The ONS TS 3.9.1 LCO wording does not include STS LCO 3.9.1.b. The basis for this deviation was that the NRC had approved TSTF-214-A which deletes this part of the LCO.

Because of deviation #1 above, the wording of Inserts 1, 2 and 3 as provided in TSTF-272-A is modified for ONS to exclude mention of the refueling cavity. Deviation #2 above has no bearing on the adoption of TSTF-272-A and is only discussed herein for completeness. This variance on the TSTF-272-A wording has no adverse impact on the intent, justification or use of the proposed TS change.

Summary of the Approved Traveler Justification
TS 3.9.1, Boron Concentration, is revised to clarify that the boron concentration limits do not apply to the refueling canal and refueling cavity when these areas are not connected to the RCS. This TS limits the boron concentrations of the RCS, the refueling canal and refueling cavity during refueling to ensure the reactor remains subcritical during MODE 6. However, when the refueling canal and refueling cavity are isolated from the RCS, no potential for dilution exists. In this condition it is not necessary to place a limit on the boron concentration of water in the refueling cavity and the refueling canal. This change is consistent with the intent of the TS and eliminates restrictions that have no effect on safety.

Differences Between the Plant-Specific Justification and the Approved Traveler Justification
The only difference is with respect to the use of the term "refueling cavity" in the NRC-approved justification. As noted previously, the term "refueling canal" at ONS includes the area typically referred to as the "refueling cavity." There are no substantive differences in the justifications.

(TSTF-272-A, continued)

Regulatory Commitments Required to Adopt this Change
None

NRC Approval
NRC approval of TSTF-272-A, Revision 1, is documented in a letter from William D. Beckner (NRC) to James Davis (NEI), dated December 12, 1999, ADAMS Accession No. ML993630256.

An example of a plant-specific NRC approval of the changes in TSTF-272-A, Revision 1, is Farley Units 1 and 2, Amendment Numbers 203 and 199, respectively, documented in a letter from S.A. Williams (NRC) to C.R. Pierce (Southern Nuclear Operating Company), Joseph M. Farley Nuclear Plant, Units 1 and 2 - Issuance of Amendments Adopting 21 Previously NRC-approved TSTF Travelers and One Request not Associated with TSTF Travelers, dated August 3, 2016, ADAMS Accession No. ML15233A448.

List of Affected Pages
TS 3.9.1-1
TS Bases 3.9.1-2
TS Bases 3.9.1-3

Applicable Regulatory Requirements/Criteria
The ONS licensing basis predates the General Design Criteria of 10 CFR 50, Appendix A; however, UFSAR Section 3.1 describes how the ONS principle design criteria were developed in consideration of the seventy General Design Criteria for Nuclear Power Plant Construction Permits proposed by the AEC in a proposed rule-making published for 10 CFR Part 50 in the Federal Register of July 11, 1967. The ONS principal design criteria include the following items relevant to this LAR:

Criterion 13 - Fission Process Monitors and Controls

> Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.
>
> Discussion
> This criterion is met by reactivity control means and control room display. Reactivity control is by movable control rods and by chemical neutron absorber (in the form of boric acid) dissolved in the reactor coolant. The position of each control rod will be displayed in the control room. Changes in the reactivity status due to soluble boron will be indicated by changes in the position of the control rods. Actual boron concentration in the reactor coolant is determined periodically by sampling and analysis.

Criterion 27 - Redundancy of Reactivity Control

> At least two independent Reactivity Control Systems, preferably of different principles, shall be provided.
>
> Discussion
> This criterion is met by movable control rods Section 4.3.2, Section 7.6.1.1 and soluble boron poison.

(TSTF-272-A, continued)

Criterion 66 - Prevention of Fuel Storage Criticality

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Discussion

Criticality of new or spent fuel is prevented by limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies. Fuel assemblies cannot be placed in other than the prescribed locations.

The proposed change clarifies that the limits on RCS boron concentration are only applicable to those portions of the RCS that are in communication with the reactor core and can, therefore, affect core reactivity. There is no adverse impact on the ability to meet the above regulatory requirements as a result of this change.

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 the proposed revision to TS 3.9.1 and operation of the unit in the proposed manner, (2) the proposed revision will be implemented in a manner consistent with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

No Significant Hazards Consideration

Duke Energy has evaluated 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," as discussed below:

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

   No. The proposed change modifies the Applicability of Technical Specification 3.9.1, "Boron Concentration," to clarify that the boron concentration limits are only applicable to the refueling canal when that volume is connected to the reactor coolant system (RCS). The boron concentration of water volumes not connected to the RCS is not an initiator of an accident previously evaluated. The ability to mitigate any accident previously evaluated is not affected by the boron concentration of water volumes not connected to the RCS. Therefore, the proposed TS change does not significantly increase 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?

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

(TSTF-272-A, continued)

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

No. The proposed change modifies the Applicability of Technical Specification 3.9.1, "Boron Concentration," to clarify that the boron concentration limits are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. Technical Specification Surveillance Requirement 3.0.4 requires that Surveillances be met prior to entering the Applicability of a Specification. As a result, the boron concentration of the refueling cavity or the refueling canal must be verified to satisfy the LCO prior to connecting those volumes to the RCS. The margin of safety provided by the refueling boron concentration is not affected by this change as the RCS boron concentration will continue to satisfy the LCO. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, 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-421-A, Rev. 0, Revision to RCP Flywheel Inspection Program (WCAP-15666)

Description of Proposed Change

Consistent with the NRC-approved TSTF-421, the proposed change revises TS 5.5.8 to read: "This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20-year intervals.

Differences Between the Proposed Change and the Approved Traveler

There is no difference between the proposed ONS TS 5.5.8 wording and final wording of TS 5.5.7 as provided in TSTF-421-A.

In addition to adopting the change made by TSTF-421-A, the proposed change to the ONS TS includes adoption of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, with the noted exceptions to Positions C.4.b(1) and C.4.b(2). This ONS request is similar to the TSTF-421-A change approved by the NRC for the Shearon Harris Nuclear Power Plant in a Safety Evaluation issued on June 21, 2005, ADAMS Accession No. ML051610409.

Summary of the Approved Traveler Justification

Duke Energy is proposing to adopt previously accepted changes to the RCP flywheel inspection methods that define an allowable alternative to the inspections described in Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1. The inspections are defined as constituting an in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or an alternative surface examination (magnetic particle testing [MT] and/or liquid penetrant testing [PT]) of exposed surfaces of removed flywheels. Prior to TSTF-421, the NRC staff accepted an allowable interval for these alternate inspections of approximately 10 years. Although ONS did not adopt previously accepted generic changes (i.e., alternate inspections to RG 1.14 and inspection intervals of 10 years), the technical basis for the change, as described in the NRC-approved topical report WCAP-15666, is valid to incorporate the allowable alternative to the inspections described in RG 1.14 and to adopt the 20-year inspection interval.

The justification for the proposed change is provided in WCAP-15666, which the NRC staff accepted for referencing in license applications by a letter and Safety Evaluation dated May 5, 2003. The WCAP-15666 topical report addresses the three critical speeds defined in RG 1.14: (a) the critical speed for ductile failure, (b) the critical speed for non-ductile failure and (c) the critical speed for excessive deformation of the flywheel. The NRC found that the topical report adequately addressed these RG 1.14 issues and demonstrated that acceptance criteria, for normal and accident conditions defined in RG 1.14, would continue to be met for all domestic Westinghouse plants. Although ONS is a B&W plant, its flywheels are of the same material type (i.e., SA533B) as the flywheels evaluated in the topical report and are appropriately bounded by WCAP-15666, as discussed below. The WCAP-15666 topical report also provides a risk assessment for extending the RCP flywheel inspection interval and the NRC staff's review, documented in the SE for the topical report, determined that the analysis methods and risk estimates are acceptable when compared to the guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific

Changes to the Licensing Basis." This risk assessment in WCAP-15666 is also applicable to ONS, as discussed below.

Duke Energy used the following WCAP-15666 risk evaluation assumptions along with ONS plant-specific data to confirm that the WCAP applies to ONS and is appropriately bounding:

1. The conditional core damage probability given an RCP motor flywheel failure is assumed to be 1.0 in WCAP-15666. This is a conservative and bounding assumption acknowledged by the NRC staff in the Crystal River Unit 3 precedent.

2. The conditional probability of loss of offsite power (LOOP) and consequential loss of power to the RCP given a loss-of-coolant accident (LOCA) and startup of the emergency core cooling system (ECCS) is estimated in NUREG/CR-6538 as 1.4E-2. The same value is conservatively used for a LOOP following a general reactor trip (a general reactor trip places less demand on the electrical systems than the startup of the ECCS, and NUREG/CR-6538 estimates the conditional probability of a LOOP given a general transient reactor trip as about a factor of 10 lower). This generic conditional probability of 1.4E-2 for a conditional LOOP is used in WCAP-15666. ONS has not experienced a LOOP due to a reactor trip. Therefore, the use of the generic value for a conditional LOOP in a PWR obtained from NUREG/CR-6538 is reasonable and consistent with ONS operating experience.

3. The generic frequency for a general transient reactor trip is estimated as one (1.0) event per year in WCAP-15666. The current ONS probabilistic risk assessment model frequency for a reactor trip is slightly lower (i.e., 2.33E-01 per year). Therefore, on its own, applying the ONS plant-specific reactor trip frequency instead of the WCAP-15666 estimate would result in a smaller increase in risk than the change in risk values documented in WCAP-15666.

4. The mean value for the frequency of a large LOCA that is used in WCAP-15666 is 2E-06 per year. The plant-specific ONS large LOCA initiating event frequency estimate is 3.43E-07 per year. Therefore, applying the ONS plant-specific large LOCA initiating event frequency instead of the WCAP-15666 estimate would result in a smaller increase in risk than the change in risk values documented in WCAP-15666.

5. The material used for ONS flywheels (i.e., SA-533B / ASTM A-533 Grade B Class 1) is the same material analyzed in WCAP-15666.

6. The sensitivity study described in Section 3.3 of WCAP-15666 demonstrates that the flaw detection probability, which is a measure of how well the RCP flywheel inspections are performed, has essentially no effect on flywheel failure probability. The sensitivity study results documented in Table 3-9 of WCAP-15666 match the sensitivity study results documented in Table 5-6 of WCAP-14535A, the latter of which was based on WCAP-14535A Group 10 flywheels. The ONS flywheels are Group 10.

7. From a review of WCAP-14535A, “Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,” the following conclusions are noteworthy to justify extension of the RCP flywheel inspection interval to an interval not to exceed 20 years.

   - The ductile failure limiting speeds for the ONS flywheels from WCAP-14535A are comparable to the ductile limiting speeds for the flywheels in WCAP-15666.

   - The ONS maximum flywheel deformation for the flywheel overspeed condition from WCAP-14535A (i.e., 0.006 in.) is the same as that of flywheels analyzed in WCAP-15666.

   - Fatigue crack growth in ONS flywheels from WCAP-14535A assuming 6000 startups and shutdowns (i.e., 0.07 in.) is less than that of the flywheels analyzed in WCAP-15666.

   - The ONS critical crack length for flywheel overspeed to 1500 RPM that includes shrink fit stresses was provided in a WCAP-14535A RAI response dated June 17, 1996. The ONS critical crack length (Group 10) including shrink fit is similar to that of the RCP flywheels analyzed in WCAP-15666 and the comparison between the critical crack length values indicates similar flaw tolerance between the ONS flywheels and the flywheels evaluated in WCAP-15666. As noted in the NRC Safety Evaluation for WCAP-15666, critical crack size values of 3.1 inches and 3.6 inches for Group 1 and Group 2 flywheels having an assumed RT_{NDT} of 60°F are quite large, even when considering higher values of RT_{NDT} and a lower than expected operating temperature of 70°F. The ONS flywheel critical crack size is 3.3 inches at an RT_{NDT} of 60°F.

Based on the above evaluation of the ONS RCP motor flywheels, Duke Energy has confirmed that WCAP-15666 is applicable to ONS. Consistent with the conclusions of WCAP-15666, the change in risk from extending the inspection interval for the ONS RCP motor flywheels to an interval not to exceed 20 years is significantly below the RG 1.174 acceptance criteria.

<u>Differences Between the Plant-Specific Justification and the Approved Traveler Justification</u>
TSTF-421-A is identified as being applicable to NUREG-1431, which is the Standard Technical Specifications for Westinghouse plants. ONS is a Babcock and Wilcox (B&W) plant; however, in a Westinghouse Owner's Group (WOG) letter to the NRC dated September 8, 2003 (ADAMS Accession No. ML032830622), the WOG indicated that WCAP-15666 may be applied to the B&W plants listed in Table 2-3 of the WCAP, provided that the licensees confirm the applicability of the WCAP to their plants. Oconee Units 1, 2 and 3 are listed in Table 2-3. In the Notice of Availability for TSTF-421 (68 FR 60422), the NRC stated "The NRC staff acknowledges that some of the supporting material for TSTF-421 may also help to support plant-specific applications for the B&W units included in portions of WCAP-15666. The NRC staff will work with licensees for the applicable B&W units to ensure that our processes work as efficiently as possible for those applying for license amendments similar to that described in TSTF-421."

(TSTF-421-A, continued)

The justification provided herein confirms the applicability of WCAP-15666 to ONS and is consistent with the information provided in a previous NRC-approved license amendment for another B&W plant as discussed below.

Regulatory Commitments Required to Adopt this Change
None

NRC Approval
TSTF-421, Revision 0 was approved by the NRC as documented in the NRC Notice of Availability published on October 22, 2003 (68 FR 60422), Notice of Availability of Model Application Concerning Technical Specification Improvement Regarding Extension of Reactor Coolant Pump Motor Flywheel Examination for Westinghouse Plants Using the Consolidated Line Item Improvement Process.

The proposed change is consistent with an NRC-approved license amendment issued to Florida Power Corporation on July 27, 2005 for Crystal River Unit 3 (ADAMS Accession No. ML051890348). Like ONS, Crystal River Unit 3 is a B&W plant. The justification provided above is largely based on the content and level of detail documented by the NRC staff in its Safety Evaluation for Crystal River Unit 3.

List of Affected Pages
TS 5.0-12
There are no TS Bases changes associated with the proposed change.

Applicable Regulatory Requirements/Criteria
The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on October 22, 2003 (68 FR 60422), NRC Notice for Comment published on June 24, 2003 (68 FR 37590), TSTF-421, WCAP-15666 and the NRC safety evaluation for the WCAP.

No Significant Hazards Consideration
Duke Energy has reviewed the proposed no significant hazards consideration determination published on June 24, 2004 (68 FR 37590). Duke Energy has concluded that the proposed determination presented in the notice is applicable to Oconee Nuclear Station, Units 1, 2 and 3 and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

3 ENVIRONMENTAL CONSIDERATION

Duke Energy Carolinas, LLC (Duke Energy) has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke Energy has determined that this license amendment request meets the criteria for a categorical exclusion as set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

No significant hazards (NSH) considerations for the proposed adoption of TSTF-272-A and TSTF-421-A are documented in Sections 2.1 and 2.2 of this document, respectively. These NSH considerations both conclude that the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed changes will not change the types or amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not increase the individual or cumulative occupational radiation exposure.

Therefore, neither an environmental impact statement nor environmental assessment are required pursuant to 10 CFR 51.22(b).

ATTACHMENT 1

MARKED-UP TECHNICAL SPECIFICATIONS PAGES

[2 pages follow this cover page]

NOTE: This attachment contains marked-up TS Pages 3.9.1-1 &. 5.0-12.

TSTF-272-A

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

5.5 Programs and Manuals

5.5.7 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.8 Reactor Coolant Pump Flywheel Inspection Program

~~This program shall provide for inspection of each reactor coolant pump flywheel. At approximately three year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an inplace, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed if the interval measured from the previous such inspection is greater than 6 2/3 years. The interval may be extended up to one year to permit inspections to coincide with a planned outage.~~

5.5.9 Inservice Testing Program (Deleted)

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

ATTACHMENT 2

MARKED-UP TECHNICAL SPECIFICATION BASES PAGES

[2 pages follow this cover page]

NOTE: This attachment contains marked-up TS Bases Pages B 3.9.1-2 & 3.

BASES (continued)

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations with CONTROL RODS and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

Violation of the LCO results in uncertainty with respect to the degree of subcriticality and could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical.

Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.5, "Safety Rod Position Limits," and LCO 3.2.1, "Regulating Rod Position Limits," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal when that volume is connected to the Reactor Coolant System. When the refueling canal is isolated from the RCS, no potential path for boron dilution exists.

ACTIONS

A.1 and A.2

BASES

ACTIONS

A.1 and A.2 (continued)

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, action to restore the concentration must be initiated immediately.

One means of complying with the action is to initiate boration of the affected volume. In determining the required combination of boration flow rate and concentration, there is no unique Design Basis Event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

connected portions of

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined by chemical analysis.

required

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 3.1
2. 10 CFR 50.36.

Prior to re-connecting portions of the refueling canal to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the canal was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

ATTACHMENT 3

RETYPED TECHNICAL SPECIFICATIONS PAGES

[2 pages follow this cover page]

NOTE: This attachment contains retyped TS Pages 3.9.1-1 & 5.0-12

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

5.5 Programs and Manuals

5.5.7 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.8 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

5.5.9 Inservice Testing Program (Deleted)

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.