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June 04, 2026

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

Serial No.	25-139
NRA/CL:	R0
Docket No.	50-395
License No.	NPF-12

DOMINION ENERGY SOUTH CAROLINA, INC.
VIRGIL C. SUMMER NUCLEAR STATION UNIT 1
LICENSE AMENDMENT REQUEST - PROPOSED REVISION TO TECHNICAL SPECIFICATIONS TO RELOCATE CERTAIN CYCLE-SPECIFIC CORE OPERATING LIMITS TO THE CORE OPERATING LIMITS REPORT AND INCORPORATION OF IMPROVED F_Q METHODOLOGY

Pursuant to 10 CFR 50.90, Dominion Energy South Carolina, Inc. (DESC) is hereby submitting a License Amendment Request (LAR) for Virgil C. Summer Nuclear Station (VCSNS) Unit 1.

The proposed amendment would revise the Technical Specifications (TS) for VCSNS to relocate certain cycle-specific core operating limits to the Core Operating Limits Report (COLR). The proposed amendment will also revise VCSNS TS to address non-conservatisms in the Heat Flux Hot Channel Factor (F_Q) identified by Westinghouse Nuclear Safety Advisory Letter (NSAL) 09-5, Revision 1, and incorporate the NRC-approved Improved F_Q Methodology in WCAP-17661-P-A as well as other improvements to VCSNS F_Q TS.

Attachment 1 provides DESC's description and assessment of the proposed relocation of the cycle-specific core operating limits to the COLR. Attachment 2 provides DESC's description and assessment of the proposed incorporation of the NRC-approved Improved F_Q TS. Attachment 3 provides the marked-up VCSNS TS pages to reflect the proposed amendment. Attachment 4 provides the revised (clean) VCSNS TS pages. Attachment 5 provides the associated TS Bases changes for information only.

The proposed amendment does not involve a Significant Hazards Consideration under the standards set forth in 10 CFR 50.92. The basis for this determination is included in Attachment 1 and Attachment 2. DESC has also determined the proposed amendment is eligible for exclusion from an environmental assessment or environmental impact statement as set forth in 10 CFR 51.22.

The proposed amendment has been reviewed and approved by the station's Facility Safety Review Committee.

DESC requests approval of this license amendment request by May 31, 2027, with a 90-day implementation period. This supports implementation during VCSNS Unit 1 Refueling Outage 30.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of South Carolina.

If you have any questions or require additional information, please contact Cailyn Ludwig at (804) 273-5131.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 06/04/26.

Respectfully,

James Holloway (Jun 4, 2026 06:42:59 EDT)

James E. Holloway
Vice President – Nuclear Engineering and Fleet Support

Attachments:
1. Description and Assessment of Proposed Change – Cycle-Specific Core Operating Limits
2. Description and Assessment of Proposed Change – Improved F_Q Technical Specification
3. Marked-up Technical Specification Pages
4. Revised (Clean) VCSNS TS Pages
5. Associated Technical Specifications Bases Changes

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
Region II

Mr. G. Edward Miller
NRC Senior Project Manager
U.S. Nuclear Regulatory Commission

NRC Senior Resident Inspector
V. C. Summer Nuclear Station

Director, Radiation Protection Program
Bureau of Land and Waste Management

Mr. G. J. Lindamood
Santee Cooper – Nuclear Coordinator

Attachment 1

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGE – CYCLE-SPECIFIC CORE OPERATING LIMITS

**Dominion Energy South Carolina, Inc.
Virgil C. Summer Nuclear Station Unit 1**

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1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Energy South Carolina, Inc. (DESC) requests an amendment to the Renewed Facility Operating License, NPF-12, for Virgil. C. Summer Nuclear Station (VCSNS), Unit 1 to revise the Technical Specifications (TS) to relocate certain cycle-specific core operating limits from the VCSNS TS to the VCSNS Core-Operating Limits Report (COLR). DESC is proposing a change to the following TS and affiliated TS and Surveillance Requirement (SR) sections and actions:

- TS 2.1.1: Safety Limits, Reactor Core
    - TS Figure 2.1-1
- TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints
- TS 3/4.1.1: Reactivity Control Systems, Boration Control/Movable Control Assemblies and Shutdown Margin
    - Modes 1 and 2
        - TS 3.1.1.1
        - SR 4.1.1.1.1
        - TS 3.1.3.1
        - TS 3.10.1
        - SR 4.10.1.2
    - Modes 3, 4, and 5
        - TS 3.1.1.2
        - TS 3.1.2.2
        - TS 3.1.2.6
        - TS Table 3.3-1
        - TS Figure 3.1-3
- TS 3/4.2.3: RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor
- TS 3/4.2.5: DNB Parameters
    - TS Table 3.2-1
    - SR 4.2.5
- TS 3/4.9.1: Refueling Operations, Boron Concentration
- TS 6.9.1.11: Reporting Requirements, Core Operating Limits Report

The proposed TS changes relocate certain cycle-specific core operating limits from the VCSNS TS to the COLR to align with the rest of the Dominion Energy fleet and provide flexibility to use available plant operating margin without the need for cycle-specific LARs. The proposed changes implement aspects of NRC-approved documents, including Technical Specification Task Force Travelers (TSTF) TSTF-339-A, TSTF-9-A, NUREG-1431, and Generic Letter (GL) 88-16. Since VCSNS TS has not converted to the NUREG-1431 TS format, NUREG-0452, “Standard Technical Specifications for Westinghouse Pressurized Water Reactors”, was reviewed to ensure the proposed changes align with NUREG-0452 wording. The cycle-specific core operating limits and the methodologies used to calculate the cycle-specific core operating limits are not being altered.

2.0 DETAILED DESCRIPTIONS AND TECHNICAL EVALUATIONS

2.1 Justification of Proposed Change to TS 2.1.1, "Safety Limits, Reactor Core"

System Design and Operation

The TS figure for the Reactor Core Safety Limits (RCSL) (TS Figure 2.1-1) presents the limiting Reactor Coolant System (RCS) average temperature conditions (T_{avg}) as a function of pressurizer pressure and fractional rated thermal power. The figure was included in the Standard TS to satisfy the requirements of Section 50.36 of Title 10 of the Code of Federal Regulations (10 CFR 50.36), which states, in part:

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity."

The RCSL figure is intended to provide the relationship between the process variables that are available to the operators, (i.e., T_{avg}, pressurizer pressure, rated thermal power level as indicated by the excore detectors) and the departure from nucleate boiling (DNB) design basis. If a Condition I or II event were to occur, the safety limits figure could be used by the licensee to determine whether or not the DNB design basis was met. For Condition III and IV events, the RCSL figure is not applicable.

Current Technical Specification Requirement

TS 2.1.1 defines the allowed operational combination of thermal power, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}). TS 2.1.1 is applicable in Modes 1 and 2 and requires the plant to be in Hot Standby within one hour and comply with the requirements of TS 6.7.1 if operability is not restored.

Reason for the Proposed Change

TSTF-339-A [Reference 1] permits the relocation of cycle-specific parameters from the TS to the COLR. These parameters had been evaluated for COLR inclusion by WCAP-14483-A [Reference 3].

Description of Proposed Changes

The proposed changes will relocate Figure 2.1-1, "Reactor Core Safety Limits-Three Loop Operation" to the COLR, redirect the reference for Figure 2.1-1 in TS 2.1.1 to the COLR, and add two safety limits: 1) departure from nucleate boiling ratio (DNBR) and 2) peak fuel centerline temperature (as new TS 2.1.1.a and TS 2.1.1.b, respectively) to TS 2.1.1. The affiliated TS 2.1.1 Action is reworded to reference the COLR.

VCSNS Unit 1 TS 2.1.1, “Reactor Core,” currently states:

> “2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation.
>
> APPLICABILTY: MODES 1 and 2
>
> ACTION:
> Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.”

The proposed change (grey highlight) revises the VCSNS Unit 1 TS 2.1.1 to read as follows:

> “2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:
>
> a. The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.17 for the WRB-2 DNB correlation.
> b. The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.
>
> APPLICABILTY: MODES 1 and 2
>
> ACTION:
>
> Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the limits specified in the COLR, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.”

TS Figure 2.1-1, “Reactor Core Safety Limits-Three Loop Operation,” will be deleted and replaced with the wording: “THIS PAGE INTENTIONALLY LEFT BLANK.”

Markups of the proposed TS changes are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

The guidance from TSTF-339-A, Rev. 2, and WCAP-14483-A has been used to relocate the cycle-specific Reactor Core Safety Limits in TS 2.1.1 and TS Figure 2.1-1 from the TS to the COLR. The proposed change also adds TS 2.1.1.a and TS 2.1.1.b that state the Safety Limits that must be met to ensure that the DNB and fuel melt design bases, respectively, will not be exceeded.

Insert 1 of TSTF-339-A, Rev. 2, specifies that the DNBR shall be maintained greater than or equal to 1.17 for WRB-1/WRB-2 DNB correlations. The Westinghouse WRB-2 CHF correlation is used to evaluate the DNB performance of the Westinghouse VANTAGE 5 fuel, which is currently employed at VCSNS. The WRB-1 CHF correlation is not applicable to VANTAGE 5 fuel design [Reference 10] and therefore is not reflected in the new TS 2.1.1.a. The NRC-approved DNBR limit for the WRB-2 CHF correlation is 1.17 per WCAP-10444-P-A [Reference 11].

TSTF-339-A, Rev. 2, specifies that the peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup. VCSNS Final Safety Analysis Report (FSAR) Section 4.2.1.1.1 identifies the fuel melt limit employed as 5080°F decreasing by 9°F per 10,000 MWD/MTU of burnup, based upon the use of the PAD5 fuel performance code (WCAP-17642-P-A) [Reference 9]; therefore, a decrease in the fuel melt limit of 9°F per 10,000 MWD/MTU of burnup is proposed for new TS 2.1.1.b.

The NRC-approved methodology used to derive the parameters in the Reactor Core Safety Limits figure that is being relocated to the COLR is WCAP-9272-P-A. WCAP-9272-P-A is already included in TS 6.9.1.11 as an NRC-approved methodology for determining core operating limits at VCSNS. The reference to this methodology in TS 6.9.1.11 will be revised to add TS 2.1.1 as an applicable section.

2.2 Justification of Proposed Change to TS Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints”

System Design and Operation

The Overtemperature ΔT (OTDT) and Overpower ΔT (OPDT) reactor trip functions ensure operation within the DNB and fuel temperature design bases, respectively. Both limits are functions of reactor coolant temperature and pressure as well as the core thermal power, which are important reload design parameters. The OTDT and OPDT setpoints are calculated using the RCSLs, which are being moved to the COLR as described in Section 2.1, and the Axial Offset Limits, which are currently specified in the VCSNS COLR. These two reactor trips ensure that the operating regime defined by the RCSLs and Axial Offset Limits is not exceeded, thereby ensuring that the fuel temperature and DNB design bases are met.

Current Technical Specification Requirement

VCSNS Unit 1 TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," defines the reactor trip system instrumentation and interlock setpoint values. Should the instrumentation and interlock setpoints be non-conservative relative to their respective value in the Trip Setpoint column of TS Table 2.2-1, the setpoint is required to be adjusted consistent with the allowable values(s). If the instrumentation or interlock setpoint is less conservative than the value shown in the Allowable Values column of TS Table 2.2-1, the channel is declared inoperable and the applicable action statement requirement of TS 3.3.1 is applied.

Reason for the Proposed Change

The revision of VCSNS Unit 1 TS Table 2.2-1 adopts TSTF-339-A, Revision 2, which permits the relocation of cycle-specific parameters from the TS to the COLR, consistent with WOG Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report" [References 1, 3].

Description of Proposed Changes

The proposed changes will relocate the OTDT and OPDT setpoint parameters from TS Table 2.2-1 to the COLR. Specifically, the K constant values, time constants utilized in lead-lag controllers for T_{avg}, indicated T_{avg} at rated thermal power, breakpoint and slope values for the $f_1(\Delta I)$ penalty function, and nominal RCS operating pressure parameters in NOTE 1 and NOTE 3 are being relocated. Additionally, the removal of the lower bounds on T' and T" is proposed.

VCSNS Unit 1 TS Table 2.2-1, Note 1 and Note 3 currently state:

"NOTE 1: Overtemperature ΔT

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

Where:

ΔT = Measured ΔT by RTD Instrumentation

ΔT_0 $\leq$ Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.23
K_2 $\geq$ 0.0292/°F
$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, τ_2 = Time constants utilized in lead-lag controller for T_{avg}
$\tau_1 \geq 28\ secs, \tau_2\ \leq 4\ secs.$
T = Average temperature, °F
T' $\leq$ Indicated T_{avg} at RATED THERMAL POWER, 572.0°F $\leq T' \leq$ 587.4°F
K_3 $\geq$ 0.00161/psi
P = Pressurizer pressure, psig
P' $\geq$ 2235 psig, Nominal RCS operating pressure
S = Laplace transform operator, sec^{-1}

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -35 percent and +6 percent $f_1(\Delta I)$ = 0 where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.
(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 2.46 percent of its value at RATED THERMAL POWER.
(iii) for each percent that the magnitude of $q_t - q_b$ exceeds + 6 percent, the ΔT trip setpoint shall be automatically reduced by 3.29 percent of its value at RATED THERMAL POWER.

NOTE 3: Overpower ΔT

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

Where:
ΔT = as defined in Note 1
ΔT_0 = as defined in Note 1
K_4 $\leq$ 1.078
K_5 $\geq$ 0.02/°F for increasing average temperature and 0 for decreasing average temperature
$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation
τ_3 = Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \geq$ 10 secs.
K_6 $\geq$ 0.00198/°F for T > T'', and K_6 = 0 for T $\leq$ T''
T = as defined in Note 1
T'' $\leq$ Indicated T_{avg} at RATED THERMAL POWER, 572.0°F $\leq T'' \leq$ 587.4°F

S = as defined in Note 1”

The proposed change (grey highlight) revises the VCSNS Unit 1 TS Table 2.2-1, Note 1 and Note 3, to read as follows:

“NOTE 1: Overtemperature ΔT

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

Where:

ΔT = Measured ΔT by RTD Instrumentation

ΔT_0 ≤ Indicated ΔT at RATED THERMAL POWER

K_1 ≤ $[*]$

K_2 ≥ $[*]$/°F

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, τ_2 = Time constants utilized in lead-lag controller for T_{avg}

$\tau_1 \ge [*]\ secs, \tau_2 \le [*]\ secs$

T = Average temperature, °F

T' ≤ Indicated T_{avg} at RATED THERMAL POWER, $T' \le [*]$°F

K_3 ≥ $[*]$/psi

P = Pressurizer pressure, psig

P' ≥ $[*]$ psig, Nominal RCS operating pressure

S = Laplace transform operator, sec^{-1}

$f_1(\Delta I)$ = $[*]\{[*] - (q_t - q_b)\}$ when $q_t - q_b \le [*]$% RTP

0% of RTP when $[*]$% RTP < $q_t - q_b \le [*]$% RTP

$[*]\{(q_t - q_b) - [*]\}$ when $q_t - q_b > [*]$% RTP

The values denoted with $[*]$ are specified in the COLR.

NOTE 3: Overpower ΔT

$$\Delta T \le \Delta T_0 \left[K_4 - K_5 \frac{(\tau_3 S)}{(1 + \tau_3 S)} T - K_6[T - T'']\right]$$

Where:

ΔT = as defined in Note 1

ΔT_0 = as defined in Note 1

K_4 ≤ $[*]$

K_5 ≥ $[*]$/°F for increasing T_{avg}, K_5 = $[*]$/°F for decreasing T_{avg}

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \ge [*]$ secs.

K_6 ≥ [∗]/°F for T > T'', and K_6 = [∗] for T ≤ T''
T = as defined in Note 1
T'' ≤ Indicated T_{avg} at RATED THERMAL POWER, T'' ≤ [∗]°F
S = as defined in Note 1

The values denoted with [∗] are specified in the COLR."

Markups of the proposed TS changes are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

<u>Technical Evaluation</u>

The guidance of WCAP-14483-A and TSTF-339-A are used to relocate the OTDT and OPDT setpoint parameters in Notes 1 and 3 for TS Table 2.2-1 to the COLR. OTDT and OPDT setpoint parameters are verified on cycle-specific basis to ensure RSCLs and Axial Offset Limits are met. WCAP-8745-P-A [Reference 14], the NRC-approved methodology that was implemented at VCSNS in License Amendment No. 75 (ADAMS Accession No. ML012210039), is used to derive the OTDT and OPDT setpoint parameters. It is not currently listed in the methodology section in TS 6.9.1.11 and will therefore be added with TS Table 2.2-1 included as an applicable section.

The TS Table 2.2-1 OTDT and OPDT lower bounds on indicated average reactor coolant temperature at rated thermal power, 572°F ≤ T' on Note 1 and 572°F ≤ T'' on Note 3, were added in License Amendment No. 119 [Reference 12] and are being removed as a conforming change to be consistent with NUREG-0452, NUREG-1431, and TSTF-339. Both NUREG-1431 [Reference 2] and the VCSNS initial license [Reference 15] do not include the lower temperature threshold. The lower temperature thresholds were added in 1994 to support Steam Generator replacement [References 12, 13]. Reference 13 discusses the addition of a lower bound to the T' and T'' in Notes 1 and 3 of TS Table 2.2-1 to reflect updated safety analyses performed for an expanded vessel average temperature range at hot full power (HFP). Per WCAP-8745-P-A [Reference 14], there is no technical basis for a lower bound on the indicated HFP RCS T_{avg} terms (i.e., T' and T'' in TS Table 2.2-1, Notes 1 and 3, respectively) used in the OTDT and OPDT trip functions. For these reasons, the lower bound on T' and T'' is proposed to be removed.

The OTDT and OPDT equations and terms are defined differently in the TSTF than in the current VCSNS TS. However, both include terms with values that may vary by fuel cycle. Some of the parameter definitions in TS Table 2.2-1 Note 3 of the OPDT equation, specifically ΔT, ΔT_0, T, and S, defer to the definition provided in Note 1, whereas WCAP-14483-A explicitly defines each parameter for both functions. The definition of those cross-referenced parameters in Note 3 have been reviewed against Note 1 in the VCSNS TS and Note 2 in WCAP-14483-A and are confirmed to be consistent.

The inequality directions for the OTDT parameters P’ and K_3 in Note 1 of the VCSNS TS differ from those given in Note 1 of TSTF-339-A and WCAP-14483-A; however, the Note 1 inequality directions for both P’ and K_3 in the VCSNS TS are consistent with NUREG-1431, Volume 1, Revision 5 [Reference 2].

2.3 Justification of Proposed Change to TS 3/4.1.1.1, “Shutdown Margin – Modes 1 and 2”

System Design and Operation

A sufficient shutdown margin ensures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}.

Current Technical Specification Requirement

TS 3.1.1.1 defines the allowed shutdown margin for Modes 1 and 2 and requires boration to be initiated immediately to restore shutdown margin if the TS Limiting Conditions for Operation (LCO) is not met.

Reason for the Proposed Change

The proposed change incorporates TSTF-9-A [Reference 6] to the VCSNS TS, which provides core design and operational flexibility that can be used for improved fuel management and to solve cycle-specific issues by relocating the value for shutdown margin to the COLR.

Description of Proposed Changes

The proposed change will relocate the shutdown margin value from TS 3.1.1.1 and TS 4.1.1.1.1 to the COLR. Consequently, TS sections and Actions referencing TS 3.1.1.1 will be reworded to reference the COLR. The affected sections include TS 3.1.3.1, TS 3.10.1, and TS 4.10.1.2. TS 3.1.3.1 Action d.3 is repeated on pages 3/4 1-14 and 3/4 1-15; thus, the first instance of the repeated section on page 3/4 1-14 is proposed to be deleted.

The current version of the affected VCSNS Unit 1 TSs state:

"3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k for 3 loop operation.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

4.1.1.1.1 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to 1.77% delta k/k.

3.1.3.1 ACTION:

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

d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

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

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

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1."

The proposed change (grey highlight) revises the affected TSs to read as follows:

"3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

ACTION:
With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

4.1.1.1.1 The SHUTDOWN MARGIN shall be demonstrated to be within the limits specified in the COLR.

3.1.3.1 ACTION:

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

d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

3) The rod is declared inoperable and the SHUTDOWN MARGIN requirement as defined in the COLR is satisfied. POWER OPERATION may then continue provided that:

b) The SHUTDOWN MARGIN requirement as defined in the COLR is determined at least once per 12 hours.

3.10.1 The SHUTDOWN MARGIN requirement as defined in the COLR may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the required SHUTDOWN MARGIN as defined in the COLR is restored.

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the required SHUTDOWN MARGIN as defined in the COLR is restored.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits as defined in the COLR."

Markups of the proposed changes are provided in Attachment 3 and a revised version is provided in Attachment 4.

Technical Evaluation

The guidance of TSTF-9-A is used to relocate the cycle-specific TS 3.1.1.1 shutdown margin for Modes 1 and 2 to the COLR. Proposed changes to TS 3.1.3.1, TS 3.10.1, and TS 4.10.1.2 are affiliated changes, which are limited to replacing reference to TS 3.1.1.1 with reference to the COLR. Since the shutdown margin is being relocated to the COLR for Modes 1 and 2, each reference to Modes 1 and 2 shutdown margin is proposed to be replaced with reference to the COLR. TSTF-9-A stated “SDM shall be within the limits provided in the COLR.” The word “provided” is being replaced with “specified,” which aligns with NUREG-1431 and current VCSNS TS. This shutdown margin limit is determined using NRC-approved methodology, WCAP-9272-P-A, which is already included in TS 6.9.1.11 as an NRC-approved methodology for determining core operating limits at VCSNS. The reference to this methodology in TS 6.9.1.11 will be revised to add TS 3.1.1.1 as an applicable section.

2.4 Justification of Proposed Change to TS 3.1.1.2, “Shutdown Margin - Modes 3, 4, and 5”

System Design and Operation

A sufficient shutdown margin ensures that 1) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 2) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}.

Current Technical Specification Requirement

TS 3.1.1.2 defines the allowed shutdown margin for Modes 3 through 5 in TS Figure 3.1-3 and requires boration to be initiated immediately to restore shutdown margin if the TS LCO is not met.

Reason for the Proposed Change

The proposed change incorporates TSTF-9-A [Reference 6] to the VCSNS TS, which provides core design and operational flexibility that can be used for improved fuel management and to solve cycle-specific issues by relocating the value for shutdown margin to the COLR.

Description of Proposed Changes

The proposed change will relocate Figure 3.1-3, ‘Required Shutdown Margin’ and the shutdown margin reference from TS 3.1.1.2 to the COLR. Consequently, affiliated TS Table 3.3-1 ACTION 5 will be reworded. Discrepancy on shutdown margins associated with Actions of TS 3.1.2.2 and TS 3.1.2.6 will also be corrected to reference the COLR for shutdown margin requirement.

The current version of the affected VCSNS Unit 1 TSs state:

> “3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal the limits shown in Figure 3.1-3.
>
> TABLE 3.3-1 ACTION STATEMENTS
>
> ACTION 5 – With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.
>
> 3.1.2.2 <u>ACTION</u>:
>
> With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.
>
> 3.1.2.6 <u>ACTION:</u>
>
> a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.”

The proposed change (grey highlight) revises the affected VCSNS Unit 1 TSs to read as follows:

“3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

TABLE 3.3-1 ACTION STATEMENTS

ACTION 5 – With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements as defined in the COLR, as applicable, within 1 hour and at least once per 12 hours thereafter.

3.1.2.2 ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limits specified in the COLR at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

3.1.2.6 ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limits specified in the COLR at 200°F within the next 6 hours; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.”

TS Figure 3.1-3, “Required Shutdown Margin,” will be deleted and replaced with the wording: “THIS PAGE INTENTIONALLY LEFT BLANK.”

Markups of the proposed TS changes are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

The guidance of TSTF-9-A is used to relocate the cycle-specific TS 3.1.1.2 shutdown margin for Modes 3-5 to the COLR. Proposed change to TS Table 3.3-1 ACTION 5 is an affiliated change, which is limited to replacing reference to TS 3.1.1.2 with reference to the COLR. These shutdown margin limits, as depicted in Figure 3.1-3, are determined using NRC-approved methodology WCAP-9272-P-A, which is already included in TS 6.9.1.11 as an NRC-approved methodology for determining core operating limits at VCSNS. The reference to this methodology in TS 6.9.1.11 will be revised to add TS 3.1.1.2 as an applicable section.

The action statements of TS 3.1.2.2 and TS 3.1.2.6 require the unavailable boron injection flow path to be restored in 72 hours or the unit be placed in at least HOT STANDBY and borated to at least 2 percent delta k/k at 200°F. The minimum 2 percent requirement originated from the initial TS 3.1.1.2. During the implementation of Amendment No. 46 (ADAMS Accession No. ML012200376) in 1985, the minimum 2 percent requirement in TS 3.1.1.2 was changed to a variable shutdown margin requirement as a function of RCS boron concentration, as currently depicted in Figure 3.1-3. However, the affiliated replacement of the 2 percent requirement with reference to the variable shutdown margin requirements in TS 3.1.1.2 was left out for TS 3.1.2.2 and TS 3.1.2.6 and is therefore being updated to reference the COLR.

2.5 Justification of Proposed Change to TS 3/4.2.3, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor"

System Design and Operation

The TS limit on minimum RCS Total Flow Rate in Mode 1, or Minimum Measured Flow (MMF), is used in the FSAR Chapter 15 Safety Analyses initiated from nominal conditions (statistical events) and accounts for the RCS flow measurement uncertainty. The Thermal Design Flow (TDF) rate is used in the FSAR Chapter 15 Safety Analyses that are initiated from deterministic conditions (nominal biased for uncertainty). Thus, the difference between the thermal design flow rate and the RCS total flow rate encompasses the RCS flow uncertainty.

Current Technical Specification Requirement

TS 3.2.3 defines the allowed operational combination of indicated RCS total flowrate and R. R, as calculated in TS 3.2.3 and used in the RCS Total Flow Rate Versus R figure in the COLR, accounts for nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) less than or equal to the $F_{\Delta H}^{N}$ at rated thermal power limit specified in the COLR. TS 3.2.3 is applicable to Mode 1 and requires the plant to restore the combination of RCS total flow rate and R to the allowable region within 2 hours or reduce thermal power to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux – High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours if operability not restored.

Reason for the Proposed Change

The RCS flow measurement uncertainty is included within the COLR figure "RCS Total Flow Rate Versus R for Three Loop Operation," and none of the criteria of 10 CFR 50.36 requires the specific RCS flow measurement uncertainty value to be listed in the TS.

Description of Proposed Changes

The proposed change will delete the RCS flow uncertainty of 2.1%, and the parenthetical stating the uncertainty includes 0.1% for feedwater venturi fouling, from TS 3.2.3.c.

The current version of VCSNS Unit 1 TS 3.2.3.c states:

> "The measured values of $F_{\Delta H}^{N}$ shall be increased by the applicable $F_{\Delta H}^{N}$ measurement uncertainties as specified in the COLR, and used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainties of 2.1% (includes 0.1% for feedwater venturi fouling) for flow."

The proposed change (grey highlight) revises the VCSNS Unit 1 TS 3.2.3.c to read as follows:

> "The measured values of $F_{\Delta H}^{N}$ shall be increased by the applicable $F_{\Delta H}^{N}$ measurement uncertainties as specified in the COLR, and used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainty for flow."

Markups of the proposed TS change are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

RCS flow uncertainty is currently specified in the VCSNS COLR figure titled "RCS Total Flowrate vs. R for Three Loop Operation" [Reference 4]. The RCS flow uncertainty is included in the VCSNS initial license [Reference 15], consistent with the guidance of NUREG-0452 [Reference 8]. Per 10 CFR 50.36, TSs are required to include items from the following categories: 1) Safety limits, limiting safety system settings, and limiting control settings, 2) Limiting conditions for operations, 3) Surveillance requirements, 4) Design features, 5) Administrative controls, 6) Decommissioning, 7) Initial notification, or 8) Written Reports. The RCS flow uncertainty does not meet any of the requirements of the categories listed in the 10 CFR 50.36. In addition, the RCS flow uncertainty does not appear as a TS parameter in the Improved Standard Technical Specifications for Westinghouse plants as documented in NUREG-1431, Volume 1 [Reference 2]. Specifically, no changes to the treatment or derivation of setpoint uncertainty values are proposed as part of this request. As such, it is acceptable to remove the value of the RCS flow uncertainty from VCSNS TS.

2.6 Justification of Proposed Change to TS 3/4.2.5, "DNB Parameters"

System Design and Operation

The limits on DNB parameters assure that the pressurizer pressure and RCS T_{avg} are maintained within the limits of steady-state operation assumed in the accident analyses. These limits are consistent with the initial full power conditions considered in the accident analysis presented in the FSAR (note that the RCS total flow is also an important DNB parameter assumed in the safety analyses and is currently specified in the COLR). For events for which precluding DNB is the primary criterion (i.e., Condition I and II events), the safety analyses have demonstrated that the DNB design basis is satisfied. This conclusion is based on the assumption that the plant is operating in compliance with the TS requirements, particularly the DNB parameter limits, prior to the initiation of the event. In addition, the DNB parameter limits are also the basis for the initial conditions assumed for events for which precluding DNB is not a criterion (i.e., Condition III and IV events).

Current Technical Specification Requirement

TS 3.2.5 defines DNB-related parameters of RCS T_{avg} and pressurizer pressure. TS 3.2.5 is applicable in Mode 1 and requires the plant to restore the parameter(s) within 2 hours or reduce thermal power to less than 5% of RATED THERMAL POWER within the next 4 hours if operability is not restored.

Reason for the Proposed Change

TSTF-339-A [Reference 1] permits the relocation of cycle-specific parameters from the TS to the COLR. The RCS T_{avg} and Pressurizer Pressure DNB parameters had been evaluated for COLR inclusion by WCAP-14483-A [Reference 3].

Description of Proposed Changes

The proposed change will relocate Table 3.2-1, “DNB Parameters” to the COLR and revise LCO 3.2.5 and SR 4.2.5 to replace the reference to TS Table 3.2-1 with the reference to the COLR.

The current version of VCSNS Unit 1 TS 3.2.5 states:

> “3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:
>
> a. Reactor Coolant System T_{avg}
> b. Pressurizer Pressure
>
> 4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.”

The proposed change (grey highlight) revises the VCSNS Unit 1 TS 3.2.5 and SR 4.2.5 to read as follows:

> “3.2.5 The following DNB related parameters shall be maintained within the limits specified in the COLR:
>
> a. Reactor Coolant System T_{avg}
> b. Pressurizer Pressure*
>
> 4.2.5 Each of the parameters listed in Specification 3.2.5 shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.”

TS Table 3.2-1, “DNB Parameters,” will be deleted and replaced with the wording: “THIS PAGE INTENTIONALLY LEFT BLANK.” Editorial changes will be made to relocate the note denoted with [*] on Table 3.2-1 to the bottom of page 3/4 2-15 and add [*] to TS 3.2.5.b to be consistent with the condition identified in Table 3.2-1.

Markups of the proposed TS change are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

The guidance of TSTF-339-A Rev. 2 and WCAP-14483-A has been used to relocate the cycle-specific TS Table 3.2-1 DNB parameters for RCS T_{avg} and Pressurizer Pressure to the COLR. Proposed changes to TS 3.2.5 and SR 4.2.5 are affiliated changes which replace reference to TS Table 3.2-1 with reference to the COLR. It should be noted that TSTF-339 lists DNB parameters, including RCS total flowrate, in TS 3.4.1 per NUREG-1431. However, the VCSNS TS lists DNB parameters in TS 3.2.5, but RCS total flowrate is in TS 3.2.3 per NUREG-0452 (as previously discussed). RCS total flowrate was previously relocated to the COLR in Amendment No. 88 (NRC ADAMS Accession No. ML012260132) using the guidance from Generic Letter 88-16 [Reference 5]. The NRC-approved methodology used to derive the DNB parameters being relocated to the COLR is WCAP-9272-P-A. WCAP-9272-P-A is already included in TS 6.9.1.11 as an NRC-approved methodology for determining core operating limits at VCSNS. The reference to this methodology in TS 6.9.1.11 will be revised to add TS 3.2.5 as an applicable section.

2.7 Justification of Proposed Change to TS 3/4.9.1, "Refueling Operations, Boron Concentration"

System Design and Operation

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the Mode 6 boron dilution accident event in the accident analyses.

Current Technical Specification Requirement

TS 3.9.1 defines the required reactivity conditions in the RCS and refueling canal during Mode 6. If TS 3.9.1 cannot be satisfied, all operations involving CORE ALTERATIONS or positive reactivity changes must be immediately suspended and boration must be initiated and continued at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent, until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

Reason for the Proposed Change

The proposed change incorporates GL 88-16 [Reference 5] to the VCSNS TS, which provides core design and operational flexibility that can be used for improved fuel management and to solve cycle-specific issues. The proposed change is also consistent with NUREG-1431 [Reference 2].

Description of Proposed Changes

The proposed change will relocate the specific boron concentration requirement from TS 3.9.1 to the COLR and redirect the referenced boron concentration value to the COLR.

The current version of VCSNS Unit 1 TS 3.9.1 states:

> "3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:
>
> a. Either a K_{eff} of 0.95 or less, or
> b. A boron concentration of greater than or equal to 2000 ppm.
>
> ACTION:
>
> With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is more restrictive."

The proposed change (grey highlight) revises the VCSNS Unit 1 TS 3.9.1 to read as follows:

> "3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:
>
> a. Either a K_{eff} of 0.95 or less, or
> b. A boron concentration of greater than or equal to the limits specified in the COLR.
>
> ACTION:
>
> With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its

> equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to the limits specified in the COLR, whichever is more restrictive."

Markups of the proposed TS change are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

GL 88-16 provides guidance for relocation of cycle-specific parameter limits from TS to the COLR. The refueling boron concentration is a cycle-specific parameter. The proposed change is also consistent with NUREG-1431, Rev. 5. This limit is determined using NRC-approved methodology WCAP-9272-P-A, which is already included in TS 6.9.1.11 as an NRC-approved methodology for determining core operating limits at VCSNS. The reference to this methodology in TS 6.9.1.11 will be revised to add TS 3.9.1 as an applicable section.

2.8 Justification of Proposed Change to TS 6.9, "Reporting Requirements, Core Operating Limits Report"

Reason for the Proposed Change

The proposed changes are the result of the preceding proposed changes to expand the COLR.

Description of Proposed Changes

The proposed changes will expand the list of core operating limit parameter methodologies in Administrative Controls TS 6.9.1.11, add TS sections being relocated to the COLR to the TS's applicable methodology, and make editorial changes, e.g., standardizing how Westinghouse is referenced. Specifically, the additional COLR parameters are for TS sections 2.1.1, 2.2.1, 3.1.1.1, 3.1.1.2, 3.2.5, and 3.9.1. The methodology added is WCAP-8745-P-A and is associated with the change to TS 2.2.1. The remaining TS sections (TS 2.1.1, 3.1.1.1, 3.1.1.2, 3.2.5, and 3.9.1) are added as applicable sections to the currently listed methodology WCAP-9272-P-A.

The current version of VCSNS Unit 1 Administrative Controls TS 6.9.1.11 states:

> "6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient BOL and EOL Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
b. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,
c. Control Rod Insertion Limits for Specification 3/4.1.3.6,
d. Axial Flux Difference Limits, target band, and $\mathrm{APL^{ND}}$ for Specification 3/4.2.1,
e. Heat Flux Hot Channel Factor, $\mathrm{F_Q^{RTP}}$, K(z), W(z), $\mathrm{APL^{ND}}$, $W(z)_{BL}$, and $F_Q(z)$ manufacturing/measurement uncertainties for Specification 3/4.2.2,
f. Nuclear Enthalpy Rise Hot Channel Factor, $\mathrm{F_{\Delta H}^{RTP}}$, Power Factor Multiplier, $PF_{\Delta H}$, and $\mathrm{F_{\Delta H}^{N}}$ measurement uncertainties limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (F_Q Methodology for W(z) surveillance requirements).)

c. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (Westinghouse Proprietary).

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

   WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING

AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary)

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

f. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary). WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)"

The current TS method 'e' (in blue above) is under review at the NRC for deletion [Reference 7]. The proposed changes below assume WCAP-13749 has been deleted.

The proposed change (grey highlight) revises the VCSNS Unit 1 Administrative Controls TS 6.9.1.11 to read as follows:

"6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Reactor Core Safety Limits for Specification 2.1.1,
b. Overtemperature ΔT and Overpower ΔT for Specification 2.2.1,
c. Shutdown Margin for Modes 1 and 2 for Specification 3/4.1.1.1,
d. Shutdown Margin for Modes 3, 4, and 5 for Specification 3/4.1.1.2,
e. Moderator Temperature Coefficient for Specification 3/4.1.1.3,
f. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,
g. Control Rod Insertion Limits for Specification 3/4.1.3.6,
h. Axial Flux Difference Limits, target band, and $\mathrm{APL}^{\mathrm{ND}}$ for Specification 3/4.2.1,
i. Heat Flux Hot Channel Factor, $\mathrm{F}_{\mathrm{Q}}^{\mathrm{RTP}}$, K(z), W(z), $\mathrm{APL}^{\mathrm{ND}}$, $W(z)_{BL}$, and $F_Q(z)$ manufacturing/measurement uncertainties for Specification 3/4.2.2,

j. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, for Specification 3/4.2.3,
k. DNB Parameters for Specification 3/4.2.5,
l. Refueling Operations Boron Concentration for Specification 3/4.9.1.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 2.1.1 - Reactor Core Safety Limits, 3.1.1.1 & 3.1.1.2 - Shutdown Margin, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 - DNB Parameters, and 3.9.1 - Refueling Operations Boron Concentration.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (F_Q Methodology for W(z) surveillance requirements).)

c. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

   WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary).

   (Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-12610-P-A, “VANTAGE + Fuel Assembly Reference Core Report,” April 1995 (W Proprietary).

WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™,” July 2006 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

f. WCAP-8745-P-A, “Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions,” September 1986 (W Proprietary).

(Methodology for Specification 2.2.1 - Overtemperature ΔT and Overpower ΔT Trip Functions.)”

Markups of the proposed TS change are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Note: Attachment 2 of this submittal also includes changes to TS 6.9.1.11. These changes are not incorporated in Attachment 1. The full list of changes to TS 6.9.1.11 are provided as TS markups in Attachment 3, and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

The proposed change for TS 6.9.1.11 corresponds to the proposed relocation of TS core operating limits to the COLR as documented above. This complies with NRC GL 88-16, which mandates the NRC-approved methodologies used for core operating limits be specified. The applicability of methodologies associated with proposed TS sections is documented within the sections above. In addition to the above changes, minor grammatical and punctuation changes are made on the proposed TS pages to be consistent with the rest of the TS format.

2.9 Justification of Proposed Change to TS INDEX and TS BASES

The proposed change to TS INDEX is administrative. The proposed changes to TS BASES are licensee-controlled and therefore do not require NRC approval. The proposed TS INDEX markup is provided in Attachment 3 and a clean version is provided in Attachment 4. The draft TS BASES markup is provided as Attachment 5 and is included for information only.

For the TS INDEX, the figures/titles being relocated to the COLR should be replaced with the word “Deleted.”

The proposed change will revise the TS BASES for:

- TS Bases 2.1.1 and 2.2.1 to reflect the relocation of Figure 2.1-1 to the COLR.
- TS Bases 2.1.1 to reword the fuel design criteria for clarification and to align with TSTF-339-A.
- TS Bases 3/4.1.1.1 and 3/4.1.1.2 and 3/4.1.3.6 to reflect the relocation of shutdown margin to the COLR.
- TS Bases 3/4.1.2 to reflect the relocation of Figure 3.1-3 and shutdown margin to the COLR.
- TS Bases 3/4.2.2 and 3/4.2.3 to delete RCS measurement uncertainties that do not meet the requirements to be contained in TS.
- TS Bases 3/4.2.5 to reflect the relocation of DNB parameters to the COLR.
- TS Bases 3/4.9.1 to reflect the relocation of refueling boron concentration to the COLR.

3.0 REGULATORY EVALUATION

3.1 Applicable Regulatory Requirements and Criteria

10 CFR 50.36 provides the regulatory requirements for Technical Specifications.

- 10 CFR 50.36(b) states in part: “The technical specifications will be derived from the analysis and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34.”

- 10 CFR 50.36(c) states that TSs are required to include items from the following categories: 1) Safety limits, limiting safety system settings, and limiting control settings, 2) Limiting conditions for operation, 3) Surveillance requirements, 4) Design features, 5) Administrative controls, 6) Decommissioning, 7) Initial notification, and 8) Written Reports.

NRC GL 88-16 states that it is acceptable for licensees to control reactor physics parameter limits by specifying the limits associated with NRC-approved calculation methodologies in the TS. The cycle-specific parameter limits may then be removed from the TS and placed in a cycle-specific COLR, which is required to be submitted to the NRC every operating cycle or each time it is revised.

The proposed changes are reviewed with respect to regulatory requirements listed above and determined that the proposed changes do not affect compliance with these regulations and will ensure that the lowest functional capabilities of performance levels of equipment required for safe operation are met.

3.2 Precedent

TS 2.1.1, TS Table 2.2-1, TS 3/4.2.5 changes:

- Catawba Units 1 and 2 (NRC ADAMS Accession No. ML033570127)
- Callaway Unit 1 (NRC ADAMS Accession No. ML070600786)
- Vogtle Units 1 and 2 (NRC ADAMS Accession No. ML23101A159)

TS 3/4.1.1 changes:

- Millstone Unit 3 (NRC ADAMS Accession No. ML033450435)
- South Texas Project Units 1 and 2 (NRC ADAMS Accession No. ML030360230)

TS 3/4.2.3 changes:

- Millstone Unit 3 (NRC ADAMS Accession No. ML063540109)

TS 3/4.9.1 changes:

- Millstone Unit 3 (NRC ADAMS Accession No. ML033450435)
- Seabrook Unit 1 (NRC ADAMS Accession No. ML040350482)

3.3 No Significant Hazards Consideration Analysis

Pursuant to the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Dominion Energy South Carolina, Inc. (DESC) hereby requests an amendment to Virgil C. Summer Station Nuclear Station (VCSNS) Unit 1 renewed facility operating license NPF-12.

The proposed change would revise Technical Specifications (TS) 2.1.1, “Safety Limits, Reactor Core,” TS Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints,” TS 3/4.1.1, “Reactivity Control Systems, Boration Control,” TS 3/4.2.5, “DNB Parameters,” and TS 3/4.9.1, “Refueling Operations, Boron Concentration,” to move several cycle-specific parameter limits from the TS to the Core Operating Limits Report (COLR), and to remove Reactor Coolant System flow measurement uncertainty from TS 3/4.2.3, “RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.”

Additionally, the proposed change would modify TS 6.9.1.11, “Reporting Requirements, Core Operating Limits Report,” to include the list of cycle-specific parameters and NRC-approved methodologies used to develop the cycle-specific parameters.

DESC has evaluated whether 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?***

Response: No

The relocation of cycle-specific core operating limits from the Technical Specifications (TS) to the Core Operating Limits Report (COLR), addition of two safety limits to TS 2.1.1, and expansion of TS 6.9.1.11, have no influence or impact on the probability or consequences of a Design Basis Accident. The two new safety limits, which are referenced in the VCSNS Final Safety Analysis Report (FSAR), are derived using a Nuclear Regulatory Commission (NRC) approved method, which is already listed in TS 6.9.1.11. Adherence to the COLR and NRC-approved methodologies for establishing COLR parameters continues to be controlled by TS. The proposed changes still require the same actions to be taken if or when limits are exceeded. Each accident analysis addressed in the FSAR will be examined with respect to changes in cycle-dependent parameters, using NRC-approved methodologies, to ensure that the transient evaluation of new core designs is bounded by previously accepted analyses via the cycle-specific reload safety evaluation (RSE). The RSE, which will be performed in accordance with the requirements of 10 CFR 50.59, ensures that future core designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, the proposed changes will not 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?***

Response: No

The relocation of cycle-specific core operating limits from the TS to the COLR, addition of two safety limits to TS 2.1.1, and expansion of TS 6.9.1.11 do not influence, impact, nor contribute in any way to the increased probability or consequences of an accident. No safety-related equipment, safety function, or plant operations will be altered as the result of proposed changes. The cycle-specific core operating limits are calculated using the NRC-approved methods and submitted to the NRC each cycle to allow the NRC Staff to continue to trend the values of these limits. The TS will continue to require operation within the required core operating limits and appropriate actions will be taken if or when limits are exceeded. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and containment. The performance of these fission product barriers is not adversely affected by the proposed change.

   The proposed changes do not revise any setpoints assumed in the analyses and do not affect the acceptance criteria of any analyses. The relocated cycle-specific parameters will continue to be calculated using NRC-approved methodologies and will provide the same margin of safety as the values currently located in the TS. Therefore, the proposed changes will not result in a reduction in a margin of safety.

Based on the above, DESC 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.

3.4 Conclusion

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 Commission’s regulations, and (3) the issuance of the amendment will not be limited to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL CONSIDERATION

A review has determined the proposed change qualifies for a categorical exclusion from environmental review in accordance with 10 CFR 51.22. The proposed action is confined to changes to surveillance, inspection, or testing requirements as described in 10 CFR 51.22(d)(1). The proposed action does not disturb any previously undisturbed ground and there is no significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, does not result in a significant increase in individual or cumulative public or occupational radiation exposure, and does not result in a significant increase in the potential for or consequences from radiological accidents. Therefore, pursuant to 10 CFR 51.22, neither an environmental assessment nor an environmental impact statement is required.

5.0 REFERENCES

1. Technical Specification Task Force, TSTF-339-A, Revision 2, “Relocate TS Parameters to COLR,” June 2000.
2. NUREG-1431, Volumes 1&2, Revision 5, “Standard Technical Specifications Westinghouse Plants,” USNRC, September 2021.
3. WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report,” January 1999.
4. COLR-V1C29, Revision 0, “CORE OPERATING LIMITS REPORT V. C. Summer Unit 1 Cycle 29,” September 2024 (NRC ADAMS Accession No. ML24290A105).
5. Generic Letter, GL 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications,” October 04, 1988.
6. Technical Specification Task Force, TSTF-9-A, Revision 1, “Relocate value for shutdown margin to COLR,” September 1996.
7. License Amendment Request, Serial No. 25-016, “Request for Approval of Fleet Report DOM-NGF-4-P/NP, Rev. 0, and Proposed License Amendment Request; Moderator Temperature Coefficient Analytical Verification for Technical Specification Surveillances,” November 24, 2025 (NRC ADAMS Accession No. ML25329A238).
8. NUREG-0452, Revision 4, “Standard Technical Specifications for Westinghouse Pressurized Water Reactors,” November 1981.
9. WCAP-17642-P-A, Revision 1, “Westinghouse Performance Analysis and Design Model (PAD5),” November 2017.
10. V. C. Summer Nuclear Station Safety Analysis Report, Chapter 4, Revision 24.00, “Reactor,” June 2024.
11. WCAP-10444-P-A, Revision 0, “Reference Core Report VANTAGE 5 Fuel Assembly,” September 1985.
12. ISSUANCE OF AMENDMENT NO.119 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING STEAM GENERATOR REPLACEMENT – VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO.1 (TAC NO. M88172), November 18, 1994 (NRC ADAMS Accession No. ML012250175).
13. SECL-93-036, “Licensing Submittal to Support Replacement SG TS Changes for VSNS,” October 31, 1993 (NRC ADAMS Accession No. ML20059G306).
14. WCAP-8745-P-A, “Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions,” September 1986.
15. Virgil C. Summer Nuclear Station, Unit 1 – Issuance of Facility Operating License, August 1982 (NRC ADAMS Accession No. ML012200333).

Attachment 2

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGE – IMPROVED F_Q TECHNICAL SPECIFICATION

Dominion Energy South Carolina, Inc.
Virgil C. Summer Nuclear Station Unit 1

TABLE OF CONTENTS

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Energy South Carolina, Inc. (DESC) requests an amendment to Renewed Facility Operating License, NPF-12, for Virgil. C. Summer Nuclear Station (VCSNS), Unit 1 to revise the Technical Specifications (TS). DESC is proposing to change the following TS:

- TS 3/4.2.1 Axial Flux Difference (AFD)
- TS 3/4.2.2 Heat Flux Hot Channel Factor – $F_Q(z)$
- TS 3/4.10.2 Group Height, Insertion and Power Distribution Limits
- TS 6.9.1.11 Core Operating Limits Report

The proposed TS changes to TS 3/4.2.2, TS 3/4.10.2.2, TS 6.9.1.11, and the TS Bases are part of the implementation of WCAP-17661-P-A (Ref. 1). This topical report received NRC approval for the Improved F_Q methodology. The implementation of this methodology addresses non-conservatisms in NSAL-09-5 (Ref. 5) and NSAL-15-1 (Ref. 17) and removes impacts from differences in Measured vs. Predicted Axial Offset on F_Q surveillances. Additionally, steady-state and transient F_Q surveillances are made distinct and separate consistent with the improved methodology and NUREG-1431 (Ref. 18), as are the Limiting Condition for Operation (LCO) actions to restore operability. Note that the sample TS in Appendix B of Ref. 1 are in the "improved" Tech Spec (ITS) format of NUREG-1431; the VCSNS TS have the older format of NUREG-0452.

The proposed change also includes the adoptions of several Technical Specification Task Force (TSTF) change travelers to align VCSNS TS with the F_Q formulations and required actions of TS 3.2.1B of NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 5, (STS), upon which the improved Relaxed Axial Offset Control (RAOC) and Constant Axial Offset Control (CAOC) surveillance formulations were based. The following NRC-approved TSTF change travelers are proposed for adoption, as applicable to VCSNS TS 3.2.2:

- TSTF-95-A, Rev. 0, "Revise complete time for reducing Power Range High trip setpoint from 8 hours to 72 hours"
- TSTF-241-A, Rev. 4, "Allow time for stabilization after reducing power due to QPTR out of limit"
- TSTF-290-A, Rev. 0, "Revisions to hot channel factor specifications"

The revisions to TS 3/4.2.2 also require a conforming change in TS 3/4.10.2. Current TS 4.10.2.2 refers to TS 4.2.2.2, 4.2.2.4, and 4.2.2.5. This will be changed to refer to TS 4.2.2.2 and 4.2.2.3 to reflect the restructuring of proposed TS 3/4.2.2.

Additionally, the proposed TS would modify VCSNS TS 6.9.1.11, "Core Operating Limits Report" (COLR) to include WCAP-17661-P-A, Revision 1, in the list of the NRC-approved methodologies for TS 3.2.2, as well as update the list of limits and parameters required to be in the COLR.

The changes outlined are based on the sample TS but reformatted to align with the formatting and organization of current VCSNS TS which are largely based off NUREG-0452 (Ref. 8).

Licensee-controlled licensing basis documents such as the COLR, the TS Bases, and the FSAR will be updated to reflect these changes at the time of implementation of the LAR. Revisions to the TS Bases are provided in Attachment 5, for information only.

2.0 DETAILED DESCRIPTIONS AND TECHNICAL EVALUATIONS

2.1 Justification of Proposed Change to TS 3.2.1, "Axial Flux Difference (AFD)"

Reason for the Proposed Change

The proposed change to TS 3.2.1 corrects two typographical errors.

Description of Proposed Changes

The proposed changes will correct the typographical errors in TS 3.2.1 and 3.2.1.b.

VCSNS Unit 1 TS 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)," currently states:

> "3.2.1 The indicated AXIAL FLUX DIFFERNECE (AFD) shall be maintained within:
>
> a. The allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or
> b. Within the targe band specified in the COLR about the target flux difference during base load operation."

The proposed change (grey highlight) revises TS 3.2.1 to read as follows:

> "3.2.1 The indicated "AXIAL FLUX DIFFERENCE (AFD)," shall be maintained within:
>
> a. The allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or
> b. Within the target band specified in the COLR about the target flux difference

during base load operation."

Markups of the proposed TS changes are provided in Attachment 3 and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

Since these are typographical errors that are being corrected and they do not change the action or intent of the TS, no further evaluation is required.

2.2 Justification of Proposed Change to TS 3.2.2, "Heat Flux Hot Channel Factor $F_Q(z)$"

System Design and Operation

$F_Q(z)$ is defined as the maximum local fuel rod power density divided by the average fuel rod power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(z)$ is a measure of the peak fuel pellet power within the reactor core. $F_Q(z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution, and is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three-dimensional power distributions, it is possible to derive a measured value for $F_Q(z)$. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(z)$ which are present during non-equilibrium situations such as load following or power ascension. Currently, a W(z) factor is multiplicatively applied to measured $F_Q(z)$ values to ensure sufficient margin to the limit for any potential transient or non-equilibrium operation. This limit helps ensure that the peak fuel clad temperature does not exceed its acceptance criteria limit in the event of a LOCA.

Current Technical Specification Requirement

Currently, using WCAP-10216-P-A methodology (Ref. 6), TS 3.2.2 does not differentiate between steady-state and transient F_Q limits in evaluation or measurement of F_Q; there is one set of LCO actions if F_Q is found in violation of the limit. F_Q is calculated using the RAOC methodology defined in WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification," (Ref. 6). A set of relationships in current Surveillance Requirement (SR) 4.2.2.2 provides the limits for $F_Q(z)$ using this methodology:

$$F_Q^M(z) \leq \frac{[F_Q^{RTP}]*[K(z)]}{P*W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \le \frac{[F_Q^{RTP}]*[K(z)]}{0.5*W(z)} \text{ for P} \le 0.5$$

Where F_Q^M (z) is measured F_Q(z) increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, F_Q^{RTP} is the F_Q limit, K(z) is the normalized F_Q(z) as a function of core height, P is the relative thermal power, and W(z) is the cycle-dependent function. W(z) factors are generated for each available operating space; historically at least two RAOC operating spaces and one Base Load (CAOC) operating space have been analyzed.

The current action for violating the F_Q(z) limit involves reducing thermal power at least 1% for each 1% F_Q(z) exceeds the limit within 15 minutes. It also mandates changing Flux-High Trip and Overpower delta T trip setpoints accordingly within 4 hours and 72 hours, respectively. Due to non-conservatisms identified in NSAL-09-5 (Ref. 5), discussed below, compensating actions are procedurally in place to perform a 3% power reduction per 1% limit violation.

In NSAL-09-5 (Ref. 5), it is noted that the current applicable TS Action of 1% AFD band reduction per 1% F_Q exceeded may not be sufficient to correct a violation of the transient $\mathrm{F_Q^M(Z)}$ TS limit for plants operating under the Relaxed Axial Offset Control (RAOC) methodology, such as VCSNS. This issue was deemed to not have safety significance due to significant conservatisms in the methodology; however, Ref. 5 recommends implementation of an administrative limit that requires a 3% power reduction and a 1% ΔI narrowing of the Axial Flux Difference (AFD) bands for every 1% that the $\mathrm{F_Q^W(Z)}$ (including W(z) multiplier) exceeds the TS limit. The non-conservatism identified and addressed with interim changes is rooted in the W(z) methodology of calculating Transient F_Q. Section 1 of Ref. 1 (WCAP-17661) provides more background and history of the methodologies.

Current SR 4.2.2 contains requirements for both entering and remaining in the Base Load CAOC operating space. The definition and equation for the maximum allowable power (if below 100%) for Base Load operation, APL^{BL}, are currently in the COLR. Directions requiring a change in operating space after an $\mathrm{F_Q^M(Z)}$ violation are also located in SR 4.2.2.

<u>Exclusion Zones</u>

Current SR 4.2.2.2.g and 4.2.2.4.g define the exclusion zones where F_Q is not required to be measured and evaluated against the limit (due to low power and detector inaccuracy at both the top and bottom of the core). These exclusion zones are currently the top 10% and bottom 10% of the core.

Reason for the Proposed Change

NSAL-09-5 (Ref. 5) addressed a potential non-conservatism in standard TS. The reduction of 1% power for each 1% F_Q exceeds the limit is potentially non-conservative; specifically, if the limiting F_Q is near the middle of the core, a 1% power reduction may not analogously reduce peaking factor in these axial planes of the core. Additionally, Ref. 5 discusses that the W(z) method may not fully account for the potential impact on transient F_Q of differences between measured and predicted Axial Offset, and thus the calculated transient F_Q may be non-conservative. The NSAL recommended a reduction of 3% for each 1% F_Q exceeds the limit to address this. Implementation of the proposed change to the improved F_Q methodology in WCAP-17661-P-A (Ref. 1) removes impact on F_Q of differences between measured and predicted axial offset and therefore removes the need for this temporary compensating action.

Description of Proposed Changes

COLR Equations

New equations for calculating transient F_Q, $\mathrm{F_Q^W(Z)}$, will be included in the cycle-specific COLR following implementation of this improved methodology. Because VCSNS has both RAOC and CAOC (Base Load) operating spaces, guidance from both Appendix C and Appendix F of Ref. 1 were followed for COLR equations and discussion.

For RAOC Operating spaces (ROS):

$$F_Q^W(z) = [F_{xy}(z)]_{Surv}^M * \frac{T(z)}{P} * A_{XY}(z) * R_j * U_F$$

and $[F_{xy}(z)]_{Surv}^M$ is the measured planar radial peaking factor.

For Base Load Operating space:

$$F_Q^W(z) = F_Q^M(z) * \frac{W(z)}{P} * A_Q(z) * R_j * U_F$$

The T(z) function is a pre-calculated set of analytical ratios used in the improved F_Q surveillance formulation for RAOC plants to characterize the maximum transient P(z) power shape. P is the relative power and U_F is uncertainty factor. This factor accounts for manufacturing and measurement uncertainties; it was historically set at 1.0815 to be consistent with the assumptions in the LOCA analysis, but a real-time value calculated by BEACON may be used instead if allowable (Ref. 1 and Addendum 4 of Ref. 1).

The $A_{xy}(z)$ and $A_Q(z)$ factors adjust the surveillance to the reference conditions assumed in generating the T(z) or W(z) factors, respectively. $A_{xy}(z)$ and $A_Q(z)$ may be assumed to equal 1.0 or may be determined for specific surveillance conditions using the approved methods in Ref. 1. Given measured data at below full power, BEACON is still able to model a power distribution and calculate F_Q at full power, so these factors will remain 1.0 for most normal operation.

R_j is a penalty factor used to account for the potential decrease in transient F_Q margin between surveillances. It is burnup-dependent and a table of factors for various burnups will be generated for each operating space. Burnups between given points will use interpolated R_j factors. Under the previous methodology, a similar factor was applied only if F_Q was increasing between surveillances, and the factor defaulted to 2% for any penalties calculated below that. This new R_j is applied to all calculated transient F_Q values; the default value is 1.0 meaning no penalty is applied.

TS 3/4.2.2

TS 3.2.2 is being substantially revised to implement the methodology in Ref. 1. The sample markups in Appendix A of Ref. 1 are based on “standard” TS and NUREG-1431 (Ref. 18); the VCSNS TS have the older format of NUREG-0452 (Ref. 8). The proposed changes in Attachment 3 are functionally identical to the sample markups, but they are reformatted to align with the nature of VCSNS TS more closely. The numbering scheme was kept consistent (which allows the TS Bases markups taken from Appendix B of Ref. 1 to still line up with the TS), but they were written in longer paragraph form to align with the rest of VCSNS TS.

VCSNS TS include a CAOC operating space referred to as “Base Load”, which is retained in the proposed TS changes. Section 10.4 of Ref. 1 discusses retaining this operating space within the framework of the improved methodology. Provisions and stipulations for entering Base Load were retained from the original TS.

Due to the extensive nature of the proposed changes to implement WCAP-17661-P-A (Ref. 1), the numerous TSTFs, and overall alignment of VCSNS TS with those in Appendix A of Ref. 1, almost all of TS 3.2.2 will be replaced with Insert A (found in Attachment 3).

One of the most significant and noticeable changes in proposed LCO 3.2.2 is the replacement of $F_Q^M(z)$ with separated steady-state and transient F_Q, $F_Q^C(z)$ and $F_Q^W(z)$. Separating F_Q evaluation into independently calculated and evaluated steady-state and transient values is consistent with NUREG-1431 (Ref. 18) for F_Q TS based on W(z) methodology, as well as the improved F_Q methodology being implemented in WCAP-17661 (Ref. 1).

Another component to proposed LCO 3.2.2 is that the following requirement was relocated from current SR 4.2.2.3.a to proposed LCO 3.2.2:

"During Base Load Operation, THERMAL POWER shall be maintained between APL^{ND} and APL^{BL}* or between APL^{ND} and 100% (whichever is most limiting)."

The above requirement is a condition for operation within Base Load CAOC operating space, and as such is moved to be included as an LCO. This is in line with Turkey Point's Improved TS Conversion submittal, while retaining a Base Load operating space (Ref. 4). The note on APL^{BL} was added to define the term and point to the cycle-specific COLR for its definition. This is consistent with the APL^{ND} footnote in TS 3.2.1 Action b. It was determined that no TS Actions are required to be added along with this requirement. If thermal power cannot be maintained within the required range, then the unit can change over from Base Load to RAOC operation to exit the LCO condition.

The Turkey Point LAR (Ref. 4) includes Action D.1, which requires a thermal power reduction specifically when thermal power is above the threshold for Base Load operation. Adding this action would be redundant for VCSNS as proposed TS 3.2.2 Action a requires a 1% power reduction for each 1% $F_Q^C(z)$ violation, as well as other actions, for any type of operating space. Similarly, proposed TS 3.2.2 Action b includes steps required for $F_Q^W(z)$ violation regardless of type of operating space. Therefore, additional actions beyond those in proposed TS 3.2.2 to address limit violations when in Base Load operating space are not necessary.

TS 3.2.2 Action A

Proposed Action A in Insert A of Attachment 3 is functionally identical to that of sample LCO 3.2.1B Action A in Appendix A of Ref. 1. It was re-formatted to conform with VCSNS TS, but the actions and logic are identical; the notes within the Actions of the Sample TS in Ref. 1 have been moved to the bottom of the page; asterisks within the text connect the notes to the appropriate step. As discussed later, the note in Action A of the Sample TS in Ref. 1 was split into 2 notes for clarity.

TS 3.2.2 Action B

Proposed Action B in Insert A of Attachment 3 is functionally identical to that of sample LCO 3.2.1B, Action B in Appendix A of Ref. 1, with notable changes, which are discussed in more detail in the Applicability of WCAP-17661-P-A section below. The Action b.1.a wording in the proposed TS is consistent with a Southern Nuclear Company submittal (Ref. 22). The method of restoration, i.e., implementing an alternate RAOC or Base Load operating space, is elaborated upon in the TS Bases; however, simpler wording for the Action was chosen to be more concise while preserving the intent and result of the Action as written in Appendix A of Ref. 1. This also allows VCSNS to retain a Base Load CAOC operating space in addition to RAOC operating spaces without cumbersome wording and instructions. Proposed Specification 4.2.2.4.a contains requirements that SR 4.2.2.2 and SR 4.2.2.3 are satisfied prior to entering Base Load operation, and proposed Action b.1.b requires the same SR 4.2.2.2 and 4.2.2.3 to be performed after implementing the Base Load operating space.

Section 10.4 of Ref. 1 allows for the option to retain a CAOC operating space within the approved methodology. Current TS 3.2.2 Action b includes a directive to “identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a…”. The NUREG-1431 TS and sample TS in WCAP-17661 do not contain this requirement prior to increasing power. The note on proposed Action b.2.a states that Action b.2.e shall be completed whenever thermal power is limited by Action b.2.a prior to increasing thermal power above this limit. This proposed Action ensures that F_Q is within limits prior to increasing power and is functionally the same as the current TS 3.2.2 Action b directive in regard to ensuring margin to the limit. Current TS 3.2.2 Action a does not contain a requirement to reduce AFD limits when limiting thermal power due to a limit exceeded. Proposed TS 3.2.2 Action b.2.b contains this new requirement, which is consistent with NUREG-1431 LCO 3.2.1B (Ref. 18). This action provides an additional operational limitation to ensure F_Q remains within the limits defined in the COLR. The proposed completion time for this Action is also consistent with NUREG-1431, as well as TSTF-99-A.

TS 3.2.2 Action C

This action requires the plant to be in Mode 2 within 6 hours if Actions A or B are not met. This is directly from NUREG-1431 LCO 3.2.1B (Ref. 18).

<u>TS 3/4.2.2</u>

SR 4.2.2.1

Current SR 4.2.2.1 states that the provisions of Specification 4.0.4 are not applicable. SR 4.0.4 reads as the following:

> “Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.”

SR 4.0.4 aligns with standard TS in NUREG-0452 (Ref 8). Updated TS such as those in NUREG-1431 (Ref. 18) have slightly expanded SR Applicability (analogous SR 3.0.4) to contain exceptions pointing to SR 3.0.3 and LCO 3.0.4, which detail more guidelines on LCO actions and how they are impacted by Surveillance Requirements. Current SR 4.0.4 has no such allowances, and as such, current SR

4.2.2.1 cannot be removed without altering SR 4.0.4 to align with NUREG-1431. This exception will be left to avoid altering current SR 4.0.4.

SR 4.2.2.2 and 4.2.2.3

Current SR 4.2.2.2, 4.2.2.3, and 4.2.2.4 will be replaced by proposed TS in Insert B (found in Attachment 3). Proposed changes to SR 4.2.2.2 and 4.2.2.3 are directly from Appendix A of Ref. 1, with some additions. Notably, changes were made when incorporating the sample TS to continue to allow for a Base Load CAOC operating space. Additionally, modifications were made to the proposed TS in Appendix A of Ref. 1 to help align current VCSNS TS and those in WCAP-17661 (which were based on NUREG-1431). One such change is the removal of the $F_Q^M(z)$ equations in current SR 4.2.2.2.c. These equations are fundamentally changed due to both the transition to separate steady-state and transient F_Q, $F_Q^C(z)$ and $F_Q^W(z)$, as discussed in proposed TS 3.2.2 above, and the implementation of WCAP-17661. In line with NUREG-1431, and the Southern Nuclear Company submittal (Ref. 22), the equations used to calculate $F_Q^C(z)$ and $F_Q^W(z)$ using the improved F_Q methodology from WCAP-17661-P-A will be located in the cycle-specific COLR, which is transmitted to the NRC as part of the reload process. The removal of these equations, as well as discussion of methods such as PDMS (Power Distribution Monitoring System) versus movable incores, are procedural details that were justified for removal in ITS LAR precedents for plants such as Turkey Point in Ref. 4.

Proposed SR 4.2.2.2 and 4.2.2.3 a through c are directly in line with those in SR 3.2.1.1 and 3.2.1.2 of NUREG-1431, which also align closely with the corresponding requirement in the approved WCAP-17661-P-A (Ref. 1). Proposed SR 4.2.2.2.a and 4.2.2.3.a are changes from NUREG-1431 that are being implemented to update VCSNS TS and help align them with the sample TS in Appendix A of Ref. 1. The purpose of proposed SR 4.2.2.2 is to verify that $F_Q^C(z)$ is within the limit assumed in the safety analysis. The proposed change modifies the current TS by adding a new Frequency in SR 4.2.2.2.a “once after each refueling prior to THERMAL POWER exceeding 75% RATED THERMAL POWER” [RTP]. This change is acceptable because adopting the new requirement to confirm $F_Q^C(z)$ is within the limits prior to exceeding 75% RTP following each core reload will ensure that some determination of $F_Q^C(z)$ is made at a lower power level where adequate margin is available before going to 100% RTP. This change is more restrictive than current TS because it applies a new requirement that does not exist in current TS.

Proposed SR 4.2.2.2/4.2.2.3 b and c are functionally the same as current SR 4.2.2.2.d; so while this content is being moved within the SR, they are not new requirements but are instead reformatted to align with NUREG-1431. Proposed SR 4.2.2.2.b and 4.2.2.3.b include updated timelines of 24 hours, but current TS do not include any time requirement for this measurement. Including a timeline for this SR is in line with NUREG-1431, which has a time of 12 hours. However, it should be noted that WCAP-17661-P-A mentions that many plants, including the Southern Nuclear Company precedent (Ref. 22), have a time allotment of 24 hours. Including an upper bound time for this action to be performed is more restrictive than current SR 4.2.2.2.b and 4.2.2.3.b, and utilizing a time interval of 24 hours does not impact the purpose of this SR.

Changes to proposed SR 4.2.2.2 and 4.2.2.3 from those in NUREG-1431 and the sample TS in Ref. 1 are purely re-formatting to conform with VCSNS TS format.

SR 4.2.2.2.d and 4.2.2.3.d were added to retain a provision for verifying $F_Q^C(z)$ and $F_Q^W(z)$ prior to entering Base Load operation. This verification is in the current TS (SR 4.2.2.4.d.1) and has been retained to ensure there are no changes to the retention of the Base Load CAOC operating space outside of those directly impacted by the Improved F_Q Methodology in Ref. 1. A footnote was added for proposed SR 4.2.2.2.d and 4.2.2.3.d providing an exception if core power distribution measurement has been obtained with thermal power above APL^{ND} for the 24 hours prior to measurement. This exception is in current SR 4.2.2.4.d.1 and the move to a footnote is purely a formatting change.

Current SR 4.2.2.3 and 4.2.2.4 contain requirements for entering and staying in Base Load operation, as well as the steps and requirements for evaluating F_Q while in Base Load operation. The requirements of current SR 4.2.2.3 for entering Base Load operation (SR 4.2.2.3.a) and re-entering (SR 4.2.2.3.b) if power is decreased below APL^{ND} are functionally retained in proposed SR 4.2.2.4. Current SR 4.2.2.4 (evaluation of F_Q while in Base Load operation) is not required upon implementation of Ref. 1. The sample TS in Appendix A and D of Ref. 1 (for RAOC and CAOC, respectively) are very similar overall. Thus, it was decided that sample TS from Ref. 1, Appendix A would be mirrored in proposed SR 4.2.2.2 and 4.2.2.3. Specifically, proposed SR 4.2.2.2.d and 4.2.2.3.d were added to retain the SR prior to entering Base Load operating space. The COLR and TS Bases clarify which equation to use depending on RAOC or CAOC operating space, so no notes or provisions are required for these considerations in the proposed TS section. Differences between proposed TS and those in Ref. 1, including any that are due to retaining a Base Load (CAOC) operating space, are further discussed in the Applicability of WCAP-17661-P-A section below.

Current TS 3.2.1 contains a footnote that points to the COLR for the APL^{ND} value, the minimum allowable power during Base Load operation, and current SR 4.2.2.3 provides the equation to calculate APL^{BL}, the maximum allowable power (if under 100%) during Base Load operation. Proposed TS 3.2.2 contains a note analogous to the one in current TS 3.2.1 for APL^{ND} for APL^{BL}. This value and its equation, respectively, will be provided in the COLR on a cycle-specific basis. This APL^{BL} equation is currently in SR 4.2.2.3.a, but relocation to the COLR is a conforming change that aligns with the rest of the proposed TS. All other equations and factors regarding F_Q surveillances and limits, RAOC operating spaces, and Base Load CAOC operating space are to be included in the COLR. Therefore, the relocation of the equation for APL^{BL} is appropriate.

SR 4.2.2.4

Proposed SR 4.2.2.4 contains condensed requirements from current SR 4.2.2.3 and 4.2.2.4. This consolidation is primarily a formatting change as the logic is retained, but the supporting equations for APL^{BL} and the calculation of $F_Q^W(z)$ using W(z) factors are referenced in the cycle-specific COLR. Note that the calculation of $F_Q^W(z)$ using W(z) factors is in agreement with Appendix F of Ref. 1, the sample COLR for a CAOC plant, since Base Load is a CAOC operating space. Additionally, the requirement for maintaining thermal power "between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting)" was relocated to LCO 3.2.2, as it is a condition for Base Load operation and not a surveillance with a set frequency.

Current SR 4.2.2.5

Current TS 4.2.2.5 is analogous to SR 4.2.2.3 in NUREG-0452 (Ref. 8) and requires that any F_Q determinations made for reasons other than satisfying SR 4.2.2.2 still apply uncertainties as described in the COLR. NUREG-1431 (Ref. 18) does not have a corresponding requirement, nor does Ref. 1; there is no other basis for requiring SR 4.2.2.5. Due to SR 4.0.1, including an SR that specifies the applicable constraints when F_Q is measured for purposes other than performing the SR is unnecessary. Removal of the procedural details related to using PDMS or movable incores, applying manufacturing and measurement uncertainties, etc., is also consistent with the other proposed SR changes. This justification is consistent with change L03 from the Turkey Point ITS submittal (Ref. 4).

Exclusion Zones

Proposed changes relocate cycle-specific exclusion zones to the COLR as allowed by Ref. 1 and discussed in the sample TS Bases (Appendix B of Ref. 1). Further discussion of historically-utilized exclusion zones is in the Technical Evaluation section below.

Technical Evaluation

TSTF Change Traveler Adoption and Variations

TSTF-95-A "Revise completion time for reducing Power Range High trip setpoint from 8 to 72 hours"

The incorporation of this TSTF is used to modify the completion time of Required Action A.2 in TS 3.2.2 of NUREG-1431, from 8 hours to 72 hours. Current VCSNS

TS have the time to reduce power range high trip setpoints as 4 hours, in line with NUREG-0452 (Ref. 8). The incorporation of this TSTF serves to align current VCSNS TS and those in both NUREG-1431 (Ref. 18) and Appendix A of Ref. 1. Proposed TS 3.2.2 Action a.2 and Action b.2.c reflect this revised completion time. This TSTF traveler was approved by the NRC in September 1996 for incorporation in NUREG-1431, Rev. 2 in June 2001.

The current completion time of 4 hours potentially results in unnecessary trip setpoint changes, which could result in plant transients and human error. A completion time of 72 hours, per TSTF-95-A, would allow time for a flux map to be performed, or otherwise determine that a condition was temporary, without implementing an unnecessary trip setpoint change.

<u>TSTF-241-A, Rev. 4, "Allow time for stabilization after reducing power due to QPTR out of limit</u>"

TSTF-241-A was approved by the NRC in January 1999, and incorporated in NUREG-1431, Rev. 2 in June 2001. NUREG-1431, LCO 3.2.1B, was revised to provide more appropriate actions and surveillances. In proposed TS 3.2.2, Actions a.1, a.2, and a.3 were modified to be repeated after each subsequent $F_Q^C(z)$ determination if $F_Q^C(z)$ was not within the limit. This ensures that actions are continued until the parameter is restored to within its limit. The maximum allowable power level and trip setpoint reductions initially determined by proposed Actions a.1, a.2, and a.3 may be affected by subsequent determinations of $F_Q^C(z)$, and would require power and setpoint reductions within the specified time period of the $F_Q^C(z)$ determination, if necessary, to comply with the decreased maximum allowable power level and trip setpoint. Decreases in $F_Q^C(z)$ would allow increasing the maximum allowable power level and trip setpoints and raising power and setpoints to the revised limits.

The following plant specific variations include:

- This proposed change is not requesting adoption of the TSTF-241 changes associated with TS 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." Therefore, TSTF-241 markup pages of TS 3.2.4 are not provided.

- In the TSTF-241 TS Bases markup, Inserts B.2 and B.3 incorrectly refer to Required Actions A.3 and A.4, respectively. These references appear to be an error in the TSTF change traveler when copying text from Inserts A.3 and A.4. The proposed TS Bases changes refer to the correct required actions (Required Actions A.2 and A.3 of NUREG-1431, respectively). The plant-specific variations to TSTF-241-A are considered administrative variations or changes. These variations have no adverse impact that precludes adoption of this TSTF traveler to TS 3.2.2.

TSTF-290-A, Rev. 0, “Revisions to hot channel factor specifications”

TSTF-290-A was approved by the NRC in June 1999, and incorporated in NUREG-1431, Rev. 2 in June 2001. TSTF-290-A revised STS 3.2.2B (3.2.1 in the TSTF Traveler example), along with STS 3.2.2A, to reflect the approved methodologies and also added a new STS 3.2.2C to reflect this methodology. The increase in completion times approved in TSTF-95 and TSTF-99 are applied as part of the TSTF-290-A change.

Some Westinghouse designed cores began experiencing increases in the measured $F_Q(Z)$ between monthly flux maps over certain burnup ranges. Therefore, a larger penalty factor was necessary for cores predicted to have larger increases in $F_Q(Z)$ over certain burnup ranges. The NRC accepted the changes to STS TS 3.2.2B as documented in NRC acceptance letter for referencing the revised version of licensing topical report WCAP-10216, Rev. 1 "Relaxation of Constant Axial Offset Control – F_Q Surveillance Technical Specification" (Ref. 6).

Current VCSNS TS do not differentiate $F_Q(z)$ into $F_Q^C(z)$ and $F_Q^W(z)$, so there is no current action analogous to STS 3.2.2B. With the implementation of WCAP-17661-P-A (Ref. 1) and other updates to applicable TS sections, the actions and completion times in TSTF-290-A (as well as NUREG-1431 (Ref. 18) and Appendix A of Ref. 1) were mirrored in the proposed VCSNS TS. The completion times for proposed TS 3.2.2 Actions b.1 and b.2 were set to be consistent with those specified in TSTF-99-A and TSTF-95-A (4 hours for b.1 and 72 hours for b.2).

New sub-actions are being added to TS 3.2.2 Action a.4 to “perform SR 4.2.2.2 and 4.2.2.3 prior to increasing thermal power above the limit of Action a.1”. These requirements ensure that core conditions during operation at higher power levels are consistent with safety analyses assumptions, specifically $F_Q(Z)$ transient limits.

A note consistent with TSTF-290 is being added to proposed TS 3.2.2 Actions a and b to require completion of Actions a.4 and b.2.e, respectively. This condition ensures the appropriate surveillances were performed prior to increasing thermal power above the limit of Required Action a.1 or b.2.a even when requirements are met. Performance of SR 4.2.2.2 and SR 4.2.2.3 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing thermal power above the maximum allowed power level.

The following plant specific variations include:

- The change from "F_Q Methodology" to "RAOC-W(Z) Methodology" in this TSTF is not incorporated into the title and header of proposed TS 3.2.2. Current VCSNS TS do not make this clarification or even mention "F_Q Methodology". The description of the specific methodology in TSTF-290-A and NUREG-1431 is used to distinguish between the heat flux hot channel factor methodologies specified in STS 3.2.1A, 3.2.1B, and 3.2.1C. It is unnecessary to specify the methodology in the plant-specific specification title and header.

  This change is consistent with other specifications that do not provide the specific NUREG-1431 type in the plant-specific specification title and header (e.g., STS 3.3.1A, Reactor Trip System (RTS) Instrumentation (Without Setpoint Control Program), STS 3.3.1B, Reactor Trip System (RTS) Instrumentation (With Setpoint Control Program), STS 3.6.4A, Containment Pressure (Atmospheric, Dual, and Ice Condenser), STS 3.6.4B, Containment Pressure (Subatmospheric), STS 3.6.6.C, Containment Spray System (Ice Condenser), and Quench Spray (QS) System (Subatmospheric)).

- The notes for TS 3.2.2 Actions a and b are revised to incorporate changes from WCAP-17661-P-A, Rev. 1, Appendix A (Ref. 1). Refer to "Applicability of WCAP-17661-P-A, Rev. 1 Safety Evaluation and Variations," herein for further plant-specific variations.

- TSTF-290 Insert Note B incorrectly refers to Condition A in the TS Bases markup. This reference appears to be an error in the TSTF change traveler when copying text from Insert Note A. The proposed TS Bases changes refer to the correct condition (Condition B). This variation is superseded by WCAP-17661-P-A, Appendix B (Ref. 1).

- Corresponding changes to TS 6.9.1.11 are not included in TSTF-290 but are necessary to reflect the use of the WCAP-17661-P-A (Ref. 1) methodology for development of core operating limits.

- TSTF-290 Insert 4 in the TS Bases markup is a bracketed [ ] NRC reviewer's note for NUREG-1431. This insert is not included in the plant-specific TS Bases.

The plant-specific variations to TSTF-290-A are considered administrative variations and have no adverse impact that precludes adoption of this TSTF traveler.

Other Changes

Exclusion Zones

Note that VCSNS exclusion zones differ from those listed in WCAP-17661 (Ref. 1). In WCAP-17661, the standard exclusion zones of 15% are mentioned throughout, and exclusion zones are also mentioned to be “cycle-specific”. Historically, TS 4.2.2.2.g and 4.2.2.4.g have listed the VCSNS exclusion zones as 10%.

In line with NUREG-1431 (Ref. 18), exclusion zones will be removed from the TS. This information is considered procedural detail and will be discussed in the TS Bases as 10% of the top and bottom of the core, but the TS Bases will point to the COLR for cycle-specific values that may be less than 10% to ensure that the axial location of minimum margin is surveilled. This provides future flexibility without needing to change TS.

K(z)

K(z) is an axially-dependent F_Q penalty, historically included in the COLR and TS, and is applied when evaluating F_Q by multiplying the F_Q at each axial node by the corresponding K(z) value. K(z) only applies to the Best Estimate Loss of Coolant Accident (LOCA) methodology and is thus no longer applicable. With the implementation of Full Spectrum LOCA (FSLOCA) (Ref. 15) and PAD5 (Ref. 9), K(z) was changed to 1.0 for all core heights, thus rendering it unnecessary. The K(z) terms were still included in the COLR for Cycle 28 because it is referenced in current TS. With the other changes being made, all instances of K(z) are being removed from the proposed TS markups as well as the COLR.

Applicability of WCAP-17661-P-A, Rev. 1 Safety Evaluation and Variations

DESC has reviewed the NRC safety evaluation (SE) provided in the NRC letter to Westinghouse Electric Company dated November 23, 2018 (included in Ref. 11) that supported approval of WCAP-17661-P-A (Ref. 1). DESC has concluded that the STS 3.2.1B and accompanying bases changes provided in Sections 4.2 through 4.8 of the SE are applicable to VCSNS Unit 1. A list is provided herein of plant-specific variations to the proposed TS changes in Appendix A of WCAP-17661-P-A, Rev. 1 with justification for the variations:

1. As mentioned in the previous discussion of TSTF-290-A, the header for LCO 3.2.2 will not be changed to specify “T(z)” methodology. The clarification in the headers of the WCAP and NUREG-1431 are necessary to distinguish between TS for different methodologies. There is no such need for a single plant, and the retention of the header as it stands is purely administrative.

2. As mentioned previously, much of VCSNS TS are still formatted in line with NUREG-0452 (Ref. 8). The impacts of this difference on the proposed TS are as follows:

   a. Numbering: the numbering scheme of the proposed TS sections does not directly reflect that in NUREG-1431 (Ref. 18) or Appendix A (or D) of Ref. 1. Proposed TS differ in both overall section numbers (TS 3/4.2.2 for F_Q versus LCO 3.2.1 and SR 3.2.1) as well as step numbering (e.g., VCSNS proposed TS 3.2.2 Action b.1.a in Ref. 1 correlating to LCO 3.2.1, Condition B, Action B.1.1). However, the order and overall structure of the proposed TS directly follows that of sample TS in Ref. 1. Therefore, these differences are administrative and do not alter the intent of the affected requirements.

   b. Structure: NUREG-1431 (Ref. 18) has “CONDITION,” “REQUIRED ACTION,” and “COMPLETION TIME” columns for LCOs; and “SURVEILLANCE,” and “FREQUENCY” for SRs. NUREG-0452 (Ref. 8) and VCSNS TS have a sentence-based structure. These differences were carefully considered to ensure that proposed TS contain each element of the sample TS in the WCAP while ensuring overall readability. Therefore, these differences are administrative and do not alter the intent of the affected requirements.

3. The second sentence of the note in Condition A in WCAP-17661-P-A (Ref. 1) is separated into two notes in the proposed TS; this change is made for the clarity of the user. TS Bases are revised to reflect two notes to VCSNS TS 3.2.2 Action a. Section 2.1.4 of the ISTS Writer’s Guide (Ref. 20) states that “if more than one issue is addressed in the Note, separated the items as an ordered list”. Section 3.3.1.a of the ISTS Writer’s Guide suggests to “minimize the use of compound sentences that combine related actions or thoughts”, and “do not combine unrelated actions or thoughts into a compound sentence”. Separating the requirement to complete a required action when the Condition is entered and the allowance to not perform a surveillance under certain conditions into two distinct notes disconnects these two unrelated thoughts and abides by the Writer’s Guide (Ref. 20).

4. Required Action B.2.1 in Appendix A of WCAP-17661-P-A (Ref. 1) could be misread as “limit thermal power to less than rated thermal power,” which is already required by the plant operating license, rendering the action redundant. Proposed TS Action b.2.a is revised to more clearly reflect the intent of the action, which is to limit thermal power to less than the rated thermal power by the amount specified in the COLR. To further reduce confusion, the requirement to reduce the AFD limits by the amount specified in the COLR is

revised to be a separate required action (Required Action b.2.b) with a completion time of 4 hours. Subsequent required actions are re-numbered and TS bases associated with the change are revised to reflect two separate actions. Section 3.3.1.a of the ISTS Writer's Guide (Ref. 20) suggests to "minimize the use of compound sentences that combine related actions or thoughts", and "do not combine unrelated actions or thoughts into a compound sentence". Separating the requirement to limit thermal power and the requirement to reduce AFD limits disconnects these two distinct actions.

Precedent for separating the actions of reducing thermal power and reducing AFD limits was approved for North Anna Units 1 and 2 TS in Amendment Nos. 278 and 261, respectively, on October 17, 2016 (NRC ADAMS Accession No. ML16252A478) (Ref. 21). The North Anna Units 1 and 2 submittal as well as Southern Nuclear Company submittal (Ref. 22) included the proposed clarification in VCSNS TS 3.2.2 Action b.2.a to specify power reduction shall be by the amount in the COLR. This variation is considered administrative and continues to meet the intent of the requirement specified in Appendix A of WCAP-17661-P-A (Ref. 1).

5. In the proposed TS, a note was added to LCO 3.2.2 Action b.2.b (analogous to Condition B.2.1 in Appendices A and D in Ref. 1), stating "During Base Load operation, AFD limits do not need to be reduced during Action b.2.b." Appendix D of Ref. 1 contains sample TS for CAOC plants and notably does not contain this AFD reduction when implementing a different CAOC operating space. Therefore, the proposed note is in line with the methodology in Ref. 1.

6. Required Actions B.1.1 and B.1.2 and the associated note in WCAP-17661-P-A are revised and re-arranged into proposed TS 3.2.2 Action b.1.a and b.1.b. The end result of Required Action B.1.1 (analogous to proposed VCSNS TS 3.2.2 b.1.a) as presented in the WCAP is restoration of the parameter to within limits specified in the COLR. The method of restoration, in this case "Implement a RAOC operating space specified in the COLR" is moved to the TS Bases to provide an action that is as brief as possible, consistent with Section 4.1.6.d of the ISTS Writer's Guide (Ref. 20). The end result of this action is restored compliance of the parameters to "within the limits specified in the COLR," and thus, compliance with the LCO requirements. Restored compliance with the LCO allows for exiting the Action; therefore, the intent of TS 3.2.2 Action b.1.b would no longer be required. To ensure that Action b.1.b is performed when "control rod motion is required to comply with the new operating space implemented by Action b.1.a", a Note that requires performance of Action b.1.b is included. This ensures that Action b.1.b must be performed if the conditions of the Note are met, even if compliance is restored.

This phrasing change in TS 3.2.2 Action b.1.a (Required Action B.1.1 of the WCAP) was approved for Southern Nuclear Company in Ref. 22.

For VCSNS, since the Base Load (CAOC) operating space is being retained, changing the phrasing of this action to the proposed "Restore $F_Q^W(z)$ within the limits specified in the COLR" also allows entry into Base Load operation. SR 4.2.2.4 still includes all requirements for entering Base Load operation, and the proposed TS Bases will be updated to include this option. This permits implementation of both types of operating spaces (RAOC and CAOC) without additional actions or additional wording within the TS Action. Retention of Base Load operating space within the methodology is acceptable per Section 10.4 in Ref. 1; this plant-specific change achieves the end result originally intended.

This variation is considered an administrative preference in TS presentation and is consistent with the intent of the affected requirements.

7. The addition of WCAP-17661-P-A in the TS list of approved COLR methodologies is not discussed in the SE but is consistent with the changes identified in the WCAP.

Discussion of NRC Approval Limitations in WCAP-17661-P-A, Rev. 1

The NRC's Final SE (contained at the beginning of Ref. 1) includes two limitations in Section 5 that are required to ensure acceptable implementation of Ref. 1. They are included below along with discussion as to how compliance with each will be ensured for implementation at VCSNS.

Limitation 1: Use of A_{XY} and A_Q

1. NRC-approved methods in response to RAI 15.b must be used to perform the surveillance-specific A_{XY} or A_Q calculations.

   The response to RAI 15.b discusses the use of BEACON software in performing surveillances. The differences between BEACON surveillance-specific A_{XY} or A_Q calculations and those performed in the Westinghouse ANC code prior to the end of the previous cycle are marginal or conservative. T(z) and W(z) factors are calculated for both Long and Short windows, and the maximum T(z) or W(z) value for each burnup and axial height is documented. Ensuring that these windows bound the actual end of cycle operation for the previous cycle is current practice for VCSNS in line with the reload methodology in WCAP-9272-P-A (Ref. 2). Additional analysis or disposition is performed if the end of cycle burnup is not within the originally evaluated windows. BEACON will be utilized for F_Q surveillances as described in the response to RAI 15.b (Ref. 1), so other potential discrepancies between

BEACON and ANC models (as-built IFBA loading, small burnup discrepancies, etc.) fall within the uncertainties as described.

2. The depletion calculation used to determine the numerator and denominator of the A_{XY} or A_Q factor must be performed similarly to the original design calculation (RAI 15.c).

   ANC performs the design calculations and predictive cycle depletions. BEACON runs a version of ANC and will be used as described in the response to RAI 15.c (Ref. 1) to calculate A_{XY} or A_Q factors, if needed. Therefore, the calculations will be performed similarly to the design calculation.

3. The use of Method 1 for calculating A_Q is only acceptable subject to the constraints discussed in the response to RAI 15.a. Surveillance Axial Offset must be within 1.5 percent of the target AO, and there must be assurance that the limiting $F_Q^W(z)$ location does not lie within a rodded elevation at the time of surveillance.

   A brief discussion of these constraints and considerations for using Method 1 to calculate A_Q (setting it equal to 1.0) was added to the TS Bases (Attachment 5). It is specified therein that this method only applies to Base Load operation, as it is the only applicable CAOC operating space for VCSNS and therefore the only use case for A_Q.

Limitation 2: Power Level Reduction to 50 Percent RTP

The use of 50% as the final power level reduction in the event of failed F_Q surveillance is not included in TS, but it is to be included in the TS Bases and COLR.

The new TS Bases in Attachment 5 were largely derived from Appendix B of Ref. 1, and discussion of T.S. 3/4.2.2, Action b.2.b, includes a description of this 50% final power reduction.

This 50% final power level will be included in the COLR within the “Required THERMAL POWER Limits and AFD Reductions” Table (analogous to Table C-7 in Appendix C of Ref. 1). The required thermal power limit column will have <50% RTP for any required $F_Q^W(z)$ margin improvement above the specified value.

2.3 Justification of Proposed Change to TS 3/4.10.2, "Group Height, Insertion and Power Distribution Limits"

Reason for the Proposed Change

TS revisions are needed to align SR 4.10.2.2.a with the proposed changes to TS 3/4.2.2.

Description of Proposed Changes

In SR 4.10.2.2.a, "Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5." will be changed to "Specifications 4.2.2.2 and 4.2.2.3."

Technical Evaluation

Current SR 4.2.2.2 and 4.2.2.4 describe evaluation of F_Q for RAOC and Base Load operating spaces, respectively. These requirements are being replaced by SR 4.2.2.2 for evaluating steady-state $F_Q^C(z)$ for both types of operating spaces, and TS 4.2.2.3 for evaluating transient $F_Q^W(z)$ for both types of operating spaces. Therefore, SR 4.2.2.2 and 4.2.2.3 are included in proposed SR 4.10.2.2.a.

Current SR 4.2.2.5 will be deleted, as discussed in a previous section, thus it no longer needs to be included in SR 4.10.2.2.a.

2.4 Justification of Proposed Change to TS 6.9.1.11, "Reporting Requirements, Core Operating Limits Report"

Reason for the Proposed Change

TS 6.9.1.11 is updated to reflect the transition to T(z) factors with implementation of WCAP-17661-P-A (Ref. 1). Additionally, the K(z) axial dependent factor was set equal to 1.0 at all core heights with the implementation of FSLOCA at VCSNS (Ref. 15), and as such, this factor is unnecessary for F_Q surveillances.

Description of Proposed Changes

The proposed changes to the COLR TS list includes removal of specific operating parameters (e.g., the K(z) factor, power factor multiplier, and uncertainties). The COLR analytical methods in TS 6.9.1.11 are revised to include WCAP-17661-P-A as the F_Q Methodology.

The current version of TS 6.9.1.11 core operating limits 'd-f' reads:

"d. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,
e. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(z), W(z), APL^{ND}, $W(z)_{BL}$, and $F_Q(z)$ manufacturing/measurement uncertainties for Specification 3/4.2.2,
f. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, Power Factor Multiplier, $PF_{\Delta H}$, and $F_{\Delta H}^{N}$ measurement uncertainties limits for Specification 3/4.2.3."

The proposed change (grey highlight) revises TS 6.9.1.11 core operating limits 'd-f' will read:

"d. Axial Flux Difference Limits~~, target band, and~~ ~~APL^{ND}~~ for Specification 3/4.2.1,
e. Heat Flux Hot Channel Factor Limits~~,~~ ~~F_Q^{RTP}~~~~, K(z), W(z),~~ ~~APL^{ND}~~~~,~~ ~~$W(z)_{BL}$~~~~, and~~ ~~$F_Q(z)$~~ ~~manufacturing/measurement uncertainties~~ for Specification 3/4.2.2,
f. Nuclear Enthalpy Rise Hot Channel Factor~~,~~ ~~$F_{\Delta H}^{RTP}$~~~~,~~ and Power Factor Multiplier~~,~~ ~~$PF_{\Delta H}$~~~~, and~~ ~~$F_{\Delta H}^{N}$~~ ~~measurement uncertainties limits~~ for Specification 3/4.2.3."

The current version of TS 6.9.1.11 analytical method 'b' states that WCAP-10216-P-A (Ref. 6) provides the methodology for both TS 3.2.1 and 3.2.2. The proposed change removes TS 3.2.2 from this method; WCAP-10216-P-A (Ref. 6) will still be the methodology for TS 3.2.1. Item 'g' is proposed to be added to the list of analytical methods for TS 3.2.2. The changes are shown below (grey highlight):

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994, (W Proprietary).

(Methodology for Specifications 3.2.1 – Axial Fluence Difference (Relaxed Axial Offset Control) ~~and 3.2.2 – Heat Flux Hot Channel Factor (~~~~F_Q~~ ~~Methodology for W(z) surveillance requirements).~~)

g. WCAP-17661-P-A, Revision 1, "Improved RAOC and CAOC F_Q Surveillance Technical Specifications," February 2019, (W Proprietary).

(Methodology for Specification 3.2.2 – Heat Flux Hot Channel Factor.)

Note: Attachment 1 of this submittal also includes changes to TS 6.9.1.11. These changes are not incorporated in Attachment 2. The full list of changes to TS 6.9.1.11 are provided as TS markups in Attachment 3, and a revised version of the TS is provided in Attachment 4.

Technical Evaluation

Proposed TS 6.9.1.11 changes are discussed below and will be separated by the change driving the TS update, as some apply to multiple items.

Removal of "target band" and APL^{ND} in item 'd':

These parameters support Axial Flux Difference Limits for Specification 3/4.2.1; specifically, Base Load operation. The "COLR specified target band" is specified in TS 3.2.1 Action b.1 and APL^{ND} is described as being in the COLR in the footnote on the same page. As such, these parameters will continue to be generated and provided in the COLR each cycle to support the methodologies for TS 3/4.2.1 and 3/4.2.2 as they pertain to Base Load operation, which is consistent with other proposed changes to TS 6.9.1.11 as well as other units in the Dominion Energy fleet. This level of detail aligns with NUREG-1431 (Ref. 18).

Removal of requirement to provide T(z), W(z), and R_j factors in the cycle-specific COLR in item 'e':

These factors do not need to be delineated in this section as they are not limits; they will be calculated in accordance with Ref. 1 and documented on a cycle-specific basis to support F_Q surveillances using the new methodology. This aligns with other TS and COLRs in the Dominion Energy fleet.

Removal of the K(z) factor in item 'e':

K(z) is the normalized $F_Q(z)$ for a given core height. It is an axial-dependent function applied to the measured F_Q value when calculating the surveillance F_Q. With the implementation of FSLOCA (Ref. 15), K(z) was set to 1.0 in the COLR for all core heights. Thus, the function no longer needs to be included in calculating surveillance F_Q values and can be removed from the TS, as well as the COLR.

Removal of uncertainties in items 'e' and 'f':

Uncertainties are applied to $F_Q(z)$ in accordance with Ref. 1, and to $F_{\Delta H}^{N}(z)$ in accordance with Ref. 3. These are utilized in confirmation of the limits provided in the COLR, but the uncertainties are not considered limits. Therefore, they do not need to be delineated in this section of TS as per 10 CFR 50.36.

<u>Changes to analytical methods:</u>

The changes made to the analytical methods reflect the implementation of WCAP-17661-P-A (Ref. 1). The methodology in item 'b' is still used for TS 3.2.1, but item 'g' provides the improved F_Q methodology proposed for use in TS 3.2.2. Section 4.4.3.2.2 of the VCSNS Final Safety Analysis Report (FSAR) (Ref. 10) states that "the initial conditions for which DNB [departure from nucleate boiling] protection is required are assumed to be those permissible within the [RAOC] strategy for the load maneuvers described in [WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification," (Ref. 6) February 1994.]" This approved topical will still be utilized for TS 3.2.1 (Axial Flux Difference) with the proposed changes. The applicability of this report is only removed for the F_Q methodology, which is not explicitly discussed in the FSAR (Ref. 10).

3.0 REGULATORY EVALUATION

3.1 Applicable Regulatory Requirements and Criteria

Technical Specifications (TS) are included in the Part 50 operating licenses and in the Part 52 combined licenses in accordance with 10 CFR 50.36. The TS include limiting conditions for operation which provide the lowest functional capability or performance levels of equipment required for safe operation of the facility. NRC Administrative Letter 98-10 indicates that when a TS is determined to be insufficient to assure plant safety, an amendment should be requested to revise the non-conservative TS. WCAP-17661-P was developed to provide revised TSs to resolve a non-conservatism identified in NSAL-09-5. WCAP-17661-P was reviewed and approved by the NRC to resolve the issue.

DESC has reviewed the NRC's Safety Evaluation (SE), provided in the NRC letter to Westinghouse Electric Company dated November 23, 2018 (Ref. 11), that supported approval of WCAP-17661-P-A, Revision 1. DESC has concluded that the applicable regulatory performance and TS requirements and design criteria listed in Sections 2.1 and 2.2 of the SE are applicable to VCSNS Unit 1. The NRC concluded in the SE that the modified TS contained in WCAP-17661-P-A are acceptable. Thus, the proposed change is consistent with the regulatory requirements identified in the SE. In particular, (1) the revised TS Limiting Conditions for Operations (LCO) maintain facility operation within the bounds established by the safety analysis, (2) the reformulated Surveillance Requirements (SR) confirm that facility operation meets the LCOs, and (3) the revised required actions and completion times, applicable if the LCO is not met, are appropriate to restore compliance with the unmet LCO, and maintain safe facility operation. Other changes are included based on TSTF travelers TSTF-95-A, TSTF-241-A, and TSTF-290-A. TSTF travelers identify revisions to the Standardized Technical Specifications and represent NRC-approved changes for consideration with plant-specific TS amendments, in order to streamline reviews and reduce unnecessary regulatory burden.

3.2 No Significant Hazards Consideration Analysis

Pursuant to the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Dominion Energy South Carolina, Inc. (DESC) hereby requests an amendment to Virgil C. Summer Station Nuclear Station (VCSNS) Unit 1 renewed facility operating license NPF-12.

The proposed change would revise Technical Specifications (TS) 3/4.2.2 “Heat Flux Hot Channel Factor – $F_Q(z)$, TS 3/4.10.2 “Group Height, Insertion and Power Distribution Limits, and TS 6.9.1.11 “Core Operating Limits Report” to adopt the TS changes described in Westinghouse Topical Report WCAP-17661-P-A, Revision 1. The revised TS address the issues identified in Nuclear Safety Advisory Letter (NSAL) 09-05, Revision 1, “Relaxed Axial Offset Control F_Q Technical Specification Actions” and NSAL-15-1, “Heat Flux Hot Channel Factor Technical Specification Surveillance”. The proposed change includes, to the extent necessary, the adoption of several Technical Specification Task Force (TSTFs) change travelers as applicable to the VCSNS TS.

DESC has evaluated whether 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?***

   Response: No

   The proposed change resolves non-conservative TS required actions identified in Westinghouse NSAL-09-05. The Improved F_Q methodology in WCAP-17661-P-A is NRC-approved and removes impacts of measured versus predicted Axial Offset on F_Q and thus will no longer require compensating actions from NSAL-09-5.

   Adherence to the Core Operating Limits Report (COLR) and NRC-approved methodologies for establishing COLR parameters continues to be controlled by TS. Cycle-specific core operating limits are calculated using the NRC-approved methods and submitted to the NRC in the COLR each cycle, to allow the Staff to continue to trend the values of these limits. Operation in accordance with the revised TS maintains the assumptions for initial conditions of key parameter values in the safety analyses as valid and does not result in actions that would increase the probability or consequences of any accident previously evaluated.

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

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

Response: No

The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The implementation of the Improved F_Q methodology in WCAP-17661-P-A requires changes to surveillance requirements and procedures as well as TS and operator actions but does not introduce new accidents or alter those already analyzed. Operation in accordance with the revised TS and its limits precludes new challenges to systems, structures, or components that might introduce a new type of accident. Applicable design and performance criteria will continue to be met, and no new single failure mechanisms will be created.

The proposed change for resolution of Westinghouse NSAL-09-5, Revision 1 and NSAL-15-1 does not involve the alteration of plant equipment or introduce unique operational modes or accident precursors. The TS will continue to require operation within the required core operating limits and appropriate actions will be taken if limits are exceeded.

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

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

Response: No

The margin of safety is related to the ability of the fission product barriers to perform their design functions during and after an accident. These barriers include the fuel cladding, the reactor coolant system, and containment. The performance of these fission product barriers is not adversely affected by the proposed change.

The proposed changes address non-conservative TS required actions and surveillance requirements identified via NSAL-09-5, Revision 1 and NSAL-15-1, respectively. Operation in accordance with the revised TS maintains the assumptions for initial conditions of key parameter values in the safety analyses as valid. This confirms applicable design and performance criteria associated with the safety analysis will continue to be met and that the margin of safety is not adversely affected. The cycle-specific core operating limits are also calculated using the NRC-approved methods and submitted to the NRC in the COLR each cycle, to allow the Staff to continue to trend the values of these limits.

Section 6.7 of NRC-approved WCAP-17661-P-A provides a margin analysis and concludes that although the axial location of minimum margin may differ using the Improved F_Q methodology, the overall margin is essentially the same.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, DESC 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.

3.3 Conclusion

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 Commission’s regulations, and (3) the issuance of the amendment will not be limited to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL IMPACT CONSIDERATION

A review has determined the proposed change qualifies for a categorical exclusion from environmental review in accordance with 10 CFR 51.22. The proposed action is confined to changes to surveillance, inspection, or testing requirements as described in 10 CFR 51.22(d)(1). The proposed action does not disturb any previously undisturbed ground and there is no significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, does not result in a significant increase in individual or cumulative public or occupational radiation exposure, and does not result in a significant increase in the potential for or consequences from radiological accidents. Therefore, pursuant to 10 CFR 51.22, neither and environmental assessment nor an environmental impact statement is required.

5.0 PRECEDENTS

The changes proposed in the Technical Specification for implementing the NRC-approved Improved F_Q Methodology in WCAP-17661-P-A are consistent with similar changes approved by the NRC for other nuclear power plants. Some of the plants are listed below:

- Farley Units 1 and 2; Vogtle Units 1 and 2 (NRC ADAMS Accession No. ML23187A148)
- Turkey Point Units 3 and 4 (NRC ADAMS Accession No. ML23151A445)

6.0 REFERENCES

1. WCAP-17661-P-A, Rev. 1, "Improved RAOC and CAOC F_Q Surveillance Technical Specifications," February 2019.
2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
3. WCAP-12472-P-A, "BEACON™ Core Monitoring and Operation Support System," August 1994. (Original was April 1990).
4. Enclosure 2, Volume 7; Turkey Point Nuclear Generating Station Unit 3 and Unit 4, "Improved Technical Specifications Conversion, ITS Section 3.2 Power Distribution Limits", Revision 2, dated May 24, 2023 (NRC ADAMS Accession No. ML23151A445).
5. Westinghouse Nuclear Safety Advisory Letter, NSAL-09-5, Rev. 1, "Relaxed Axial Offset Control F_Q Technical Specification Actions," September 2009.
6. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification," February 1994.
7. Not Used.
8. NUREG-0452, Revision 4, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," November 1981.
9. WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.
10. V. C. Summer Nuclear Station Safety Analysis Report, Chapter 4, Revision 24.00, "Reactor," June 2024.
11. Letter from D. C. Morey (NRC) to Nowinoski (Westinghouse Electric Company), "Final Safety Evaluation for Pressurized Water Reactors Owners Group Topical Report WCAP-17661-P-A, Rev. 1, 'Improved RAOC and CAOC F_Q Surveillance Technical Specifications' (CAC NO. MF3348)," dated November 23, 2018 (NRC ADAMS Accession No. ML18298A321).
12. Not Used.
13. Not Used.
14. Not Used.

15. Letter, V. V. Thomas (USNRC) to M. D. Sartain and D. G. Stoddard, Virgil C. Summer Nuclear Station, Unit 1— Issuance of Amendment No. 219 to Modify Technical Specification 6.9.1.11, “Core Operating Limits Report” (EPID I-2020-LLA-0124). June 30, 2021. (NRC ADAMS Accession No. ML21112A108).
16. Not Used.
17. Westinghouse Nuclear Safety Advisory Letter, NSAL 15-1, “Heat Flux Hot Channel Factor Technical Specification Surveillance,” February 2015.
18. NUREG-1431, Rev. 5, Vol. 1, “Standard Technical Specifications – Westinghouse Plants,” September 2021.
19. Not Used.
20. Technical Specification Task Force Document TSTF-GG-05-01, “Writer’s Guide for Plant-Specific Improved Technical Specifications, June 2005. (NRC ADAMS Accession No. ML070660229)
21. North Anna Power Station, Units 1 and 2 – Issuance of Amendments to Revise Technical Specifications to Address Issues Identified in Westinghouse NSAL-09-5. Revision 1, and NSAL-15-1, Revision 0 (CAC Nos. MF7186 and MF7187), dated October 17, 2016 (NRC ADAMS Accession No. ML16252A478).
22. Joseph M. Farley Nuclear Plant, Units 1 and 2; and Vogtle Electric Generating Plant, Unit 1 and 2 – Issuance of Amendment Nos. 248, 245, 222, and 205 Re: Revising Technical Specification 3.2.1, ‘Heat Flux Hot Channel Factor ($F_Q(z)$),” to Implement Methodology from WCAP-17661, Revision 1 (EPID L-2022-LLA-0148), dated October 23, 2023 (NRC ADAMS Accession No. ML23187A148).

Attachment 3

MARKED-UP TECHNICAL SPECIFICATION PAGES

Dominion Energy South Carolina, Inc.
Virgil C. Summer Nuclear Station Unit 1

Attachment 1 Mark-Ups

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

Attachment 1 Mark-Ups

3.0 AND 4.0 LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENT

Re-label as "Deleted"

Attachment 1 Mark-Ups

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

Replace with INSERT 1

APPLICABILITY: MODES 1 and 2.

ACTION:

Replace with 'limits specified in the COLR,'

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

INSERT 1:

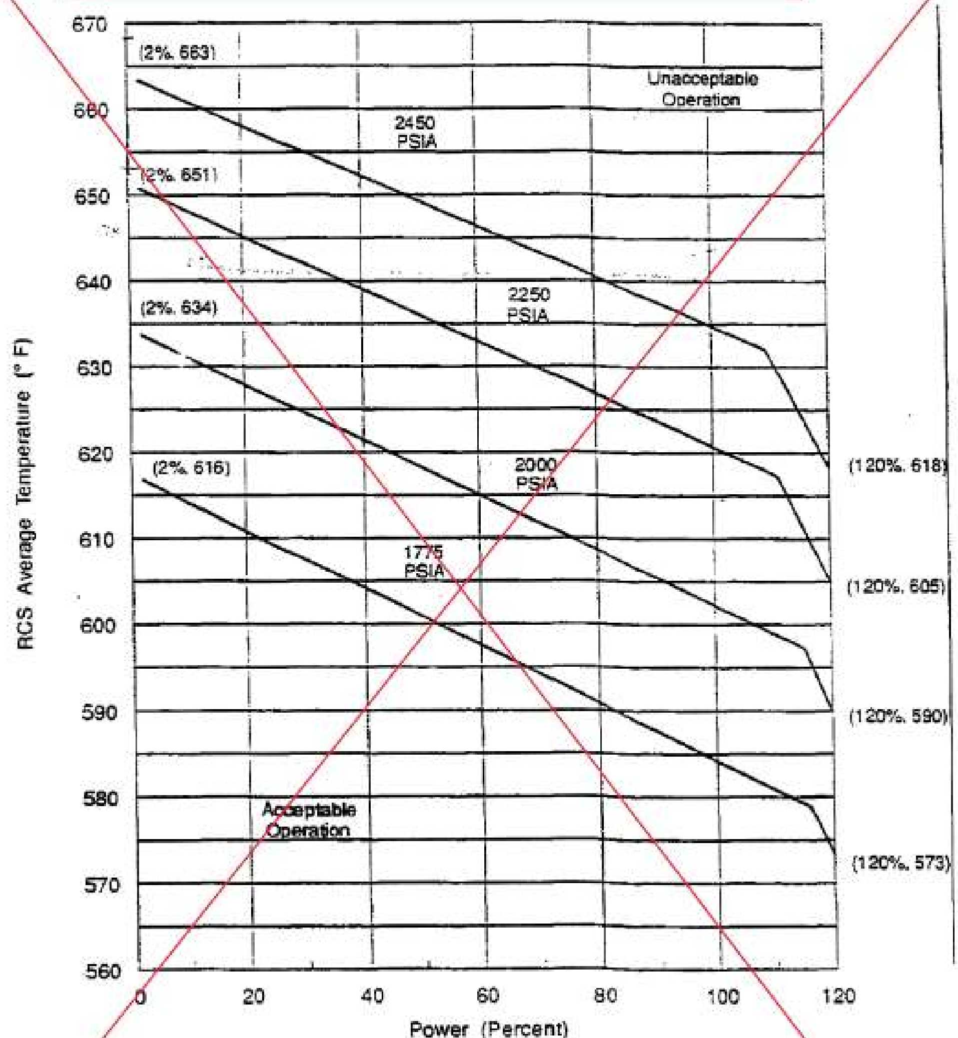
specified in the COLR; and the following SLs shall not be exceeded:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.17 for the WRB-2 DNB correlation.

b. The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.

Attachment 1 – Mark-Ups

Delete this figure and replace with wording "THIS PAGE INTENTIONALLY LEFT BLANK"



When operating in the reduced RTP region of Technical Specification 3.2.3 the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1
Reactor Core Safety Limits - Three Loop Operation

Attachment 1 – Mark-Ups

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \le \Delta T_o \left[K_1 - K_2 \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left[T - T' \right] + K_3 (P - P') - f_1(\Delta I) \right]$$

Replace with INSERT 2

Where:	ΔT	=	Measured ΔT by RTD Instrumentation
	ΔT_o	≤	Indicated ΔT at RATED THERMAL POWER
	K_1	≤	1.23
	K_2	≥	0.0292/°F
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The funtion generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_{avg}, $\tau_1 \ge 28$ secs., $\tau_2 \le 4$ secs.
	T	=	Average temperature, °F
	T'	≤	Indicated T_{avg} at RATED THERMAL POWER, 572.0°F ≤ T' ≤ 587.4°F
	K_3	≥	0.00161/psi
	P	=	Pressurizer pressure, psig
	P'	≥	2235 psig, Nominal RCS operating pressure
	S	=	Laplace transform operator, sec^{-1}.

Attachment 1 – Mark-Ups

SUMMER - UNIT 1

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 1: (Continued)

~~and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:~~

~~(i) for q_t - q_b between -35 percent and +6 percent f_1 (ΔI) = 0 where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER.~~

~~(ii) for each percent that the magnitude of q_t - q_b exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 2.46 percent of its value at RATED THERMAL POWER.~~

~~(iii) for each percent that the magnitude of q_t - q_b exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 3.29 percent of its value at RATED THERMAL POWER.~~

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.2 percent ΔT Span.

NOTE 3: OVERPOWER ΔT

$$\cancel{\Delta T \le \Delta T_o \left[K_4 - K_5 \frac{(\tau_3 S)}{(1 + \tau_3 S)} T - K_6 \left[T - T'' \right] \right]}$$

Replace with INSERT 3

~~Where: ΔT = as defined in Note 1~~

~~ΔT_o = as defined in Note 1~~

~~K_4 ≤ 1.078~~

~~K_5 ≥ 0.02/°F for increasing average temperature and 0 for decreasing average temperature~~

~~$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation~~

2-9

Amendment No. ~~24~~, ~~77~~, ~~79~~, ~~98~~, 120

Attachment 1 – Mark-Ups

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (continued)

NOTE 3: (continued)

~~τ_3~~	~~=~~	~~Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs.~~
~~K_6~~	~~$\geq$~~	~~0.00198/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$~~
~~T~~	~~=~~	~~as defined in Note 1~~
~~T''~~	~~$\leq$~~	~~Indicated T_{avg} at RATED THERMAL POWER, 572.0°F $\leq T'' \leq$ 587.4°F~~
~~S~~	~~=~~	~~as defined in Note 1~~

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.3 percent ΔT Span.

Attachment 1 – Mark-Ups

INSERT 2:

NOTE 1: Overtemperature ΔT

$$\Delta T \le \Delta T_O \left[K_1 - K_2 \frac{(1+\tau_1 S)}{(1+\tau_2 S)}[T - T'] + K_3(P - P') - f_1(\Delta I)\right]$$

Where:

ΔT	=	Measured ΔT by RTD Instrumentation
ΔT_O	≤	Indicated ΔT at RATED THERMAL POWER
K_1	≤	[*]
K_2	≥	[*]/°F
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, τ_2	=	Time constants utilized in lead – lag controller for T_{avg} $\tau_1 \ge [*] secs$, $\tau_2 \le [*]\ secs$
T	=	Average temperature, °F
T′	≤	Indicated T_{avg} at RATED THERMAL POWER, T′ ≤ [*]°F
K_3	≥	[*]/psi
P	=	Pressurizer pressure, psig
P'	≥	[*] psig, Nominal RCS operating pressure
S	=	Laplace transform operator, sec^{-1}
$f_1(\Delta I)$	=	[*] {[*] – (q_t -q_b)} when q_t -q_b ≤ [*]% RTP 0% of RTP when [*]% RTP < q_t -q_b ≤ [*]% RTP [*] {(q_t -q_b) – [*]} when q_t -q_b > [*]% RTP

The values denoted with [*] are specified in the COLR.

INSERT 3:

NOTE 3: Overpower ΔT

$$\Delta T \le \Delta T_O \left[K_4 - K_5 \frac{(\tau_3 S)}{(1+\tau_3 S)} T - K_6[T - T'']\right]$$

Where:

ΔT	=	as defined in Note 1
ΔT_O	=	as defined in Note 1
K_4	≤	[*]
K_5	≥	[*]/°F for increasing T_{avg}, K_5 = [*]/°F for decreasing T_{avg}
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in rate-lag controller for T_{avg}, τ_3 ≥ [*]secs
K_6	≥	[*] /°F for T > T″, and K_6 = [*] for T ≤ T″
T	=	as defined in Note 1
T″	=	Indicated T_{avg} at RATED THERMAL POWER, T″ ≤ [*]°F
S	=	as defined in Note 1

The values denoted with [*] are specified in the COLR.

Attachment 1 – Mark-Ups

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.77% delta k/k for 3 loop operation~~.

Replace with "...the required value,"

Replace with "...within the limits specified in the COLR."

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than ~~1.77% delta k/k,~~ immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be demonstrated to be ~~greater than or equal to 1.77% delta k/k:~~

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Surveillance Requirement 4.1.1.1.2 with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be ~~greater than or equal the limits shown in Figure 3.1-3.~~

Revise to "...within the limits specified in the COLR."

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to the required value:

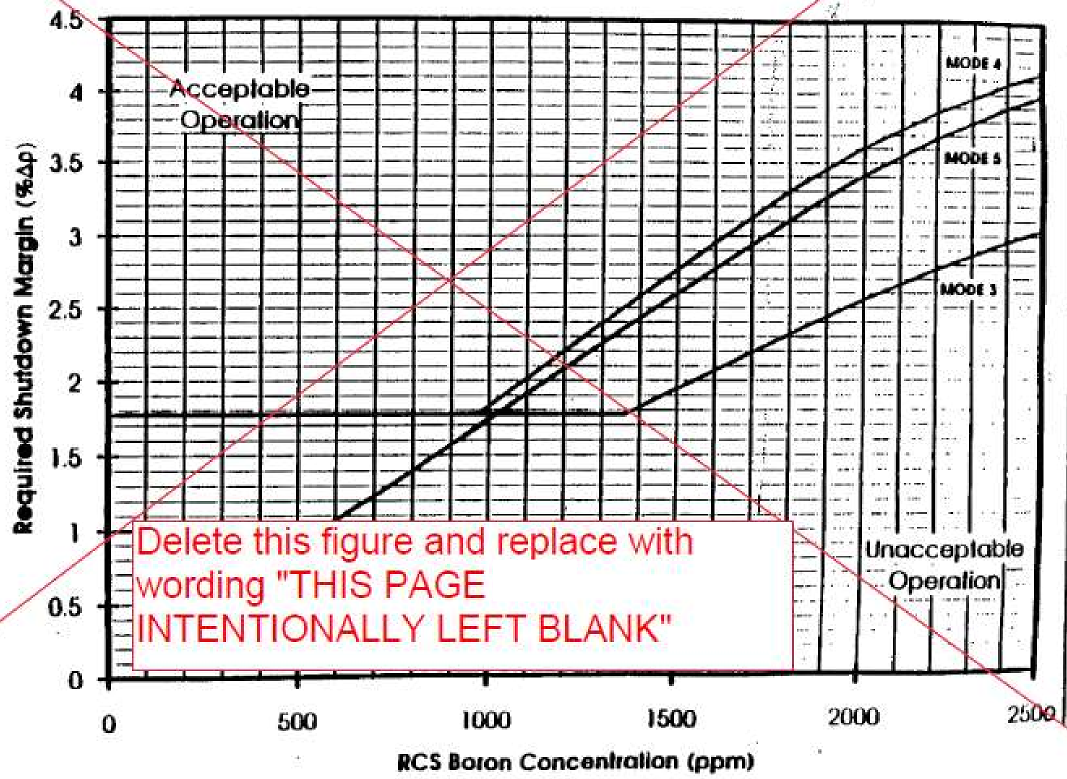
a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the operable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

   1. Reactor coolant system boron concentration,
   2. Control rod position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

Attachment 1 – Mark-Ups

FIGURE 3.1-3
REQUIRED SHUTDOWN MARGIN
(MODES 3, 4, AND 5)



Delete this figure and replace with wording "THIS PAGE INTENTIONALLY LEFT BLANK"

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~2 percent delta k/k~~ the limits specified in the COLR at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300° F.

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 14,000 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 453,800 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least ~~2 percent delta k/k~~ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

the limits specified in the COLR

b. With the refueling wate[illegible]ble, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

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

b. With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

replace with "...as defined in the COLR..."

c. With more than one full length rod inoperable due to a rod control urgent failure alarm or obvious electrical problem in the rod control system for greater than 72 hours, be in HOT STANDBY within the following 6 hours.

d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT (COLR); the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

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

replace with "...as defined in the COLR..."

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

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

c) A core power distribution measurement is obtained and $F_Q(Z)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours, and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

Attachment 1 – Mark-Ups

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled RCS Total Flow Rate Versus R For Three Loop Operation.

Where:

a. R = $\dfrac{F^N_{\Delta H}}{F^{RTP}_{\Delta H}[1.0 + PF_{\Delta H}(1.0 - P)]}$,

b. P = $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by

1. Using the movable incore detectors to obtain a power distribution map when THERMAL POWER is ≤ 25% but > 5% of RATED THERMAL POWER, or when PDMS is inoperable, and

2. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER.

replace the word 'uncertainties' with 'uncertainty'

The measured values of $F^N_{\Delta H}$ shall be increased by the applicable $F^N_{\Delta H}$ measurement uncertainties as specified in the COLR, and used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainties ~~of 2.1% (includes 0.1% for feedwater venturi fouling)~~ for flow, Delete

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

e. $PF_{\Delta H}$ = The Power Factor Multiplier specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

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

b. Within 24 hours of initially being outside the above limits, verify through a core power distribution measurement and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

Attachment 1 – Mark-Ups

POWER DISTRIBUTION LIMITS

3/4 2.5 DNB PARAMETERS

Replace with "...within the limits specified in the COLR:"

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits ~~shown on Table 3.2-1:~~

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

Add * for note below

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limits, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters ~~of Table 3.2-1~~ shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.

listed in Specification 3.2.5

Add note

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER

Attachment 1 – Mark-Ups

TABLE 3.2-1
DNB Parameters

Limits

PARAMETER	3 Loops In Operation	2 Loops In Operation
Indicated Reactor Coolant System T_{avg}	≤ 589.2°F	**
Indicated Pressurizer Pressure	> 2206 psig*	**

Delete this figure and replace with wording "THIS PAGE INTENTIONALLY LEFT BLANK"

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THEMRAL POWER step in excess of 10% of RATED THERMAL POWER

~~** These values left blank pending NRC approval of two loop operation.~~

Delete the [*] Note and move to TS 3.2.5.b

Attachment 1 – Mark-Ups

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements ~~of Specification 3.1.1.1 or 3.1.1.2~~, as applicable, within 1 hour and at least once per 12 hours thereafter.

replace with "...as defined in the COLR..."

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

Attachment 1 – Mark-Ups

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to ~~2000 ppm~~.

replace with "... the limits specified in the COLR."

APPLICABILITY: MODE 6 * with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to ~~2000 ppm~~, whichever is the more restrictive.

replace with "...the limits specified in the COLR,"

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

4.9.1.3 The following valves shall be verified locked closed ** in accordance with the Surveillance Frequency Control Program: 8430, 8454, 8441 and 8439.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** Valves may be opened under administrative control to add borated makeup.

Attachment 1 – Mark-Ups

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement ~~of Specification 3.1.1.1~~ may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

replace with "...as defined in the COLR..."

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN ~~required by Specification 3.1.1.1~~ is restored.

add "required" before "SHUTDOWN MARGIN"

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN ~~required by Specification 3.1.1.1~~ is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits ~~of Specification 3.1.1.1.~~

replace with "...as defined in the COLR..."

Attachment 1 & 2 – Mark-Ups

6.9.1.9 Not used.

6.9.1.10 Not used.

CORE OPERATING LIMITS REPORT

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient BOL and EOL Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

Attachment 2: Changes to 'd' through 'f' content (included in INSERT 4)

c. Control Rod Insertion Limits for Specification 3/4.1.3.6,

d. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,

e. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(z), W(z), APL^{ND}, $W(z)_{BL}$, and $F_Q(z)$ manufacturing/measurement uncertainties for Specification 3/4.2.2,

f. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, Power Factor Multiplier, $PF_{\Delta H}$, and $F_{\Delta H}^{N}$ measurement uncertainties limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Attachment 1: Replace with INSERT 4

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (F_Q Methodology for W(z) surveillance requirements).)

Attachment 2: Remove 3.2.2 from 'b' (included in INSERT 4)

Attachment 1 & 2 – Mark-Ups

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

~~c. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (Westinghouse Proprietary).~~

~~d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).~~

~~WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary)~~

~~(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)~~

~~e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).~~

~~(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)~~

~~f. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary). WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).~~

~~(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)~~

Attachment 2: Insert C

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Attachment 1 & 2– Mark-Ups

INSERT 4:

INSERT 4:

ADMINISTRATIVE CONTROLS

6.9.1.9 Not used.

6.9.1.10 Not used.

CORE OPERATING LIMITS REPORT

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Reactor Core Safety Limits for Specification 2.1.1,

b. Overtemperature ΔT and Overpower ΔT for Specification 2.2.1,

c. Shutdown Margin for Modes 1 and 2 for Specification 3/4.1.1.1,

d. Shutdown Margin for Modes 3, 4, and 5 for Specification 3/4.1.1.2,

e. Moderator Temperature Coefficient for Specification 3/4.1.1.3,

f. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

g. Control Rod Insertion Limits for Specification 3/4.1.3.6,

h. Axial Flux Difference Limits for Specification 3/4.2.1,

i. Heat Flux Hot Channel Factor Limits for Specification 3/4.2.2,

j. RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3,

k. DNB Parameters for Specification 3/4.2.5,

l. Refueling Operations Boron Concentration for Specification 3/4.9.1.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).
(Methodology for Specifications 2.1.1 - Reactor Core Safety Limits, 3.1.1.1 & 3.1.1.2-Shutdown Margin, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 - DNB Parameters, and 3.9.1 – Refueling Operations Boron Concentration.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control).)

Attachment 1 & 2 – Mark-Ups

INSERT 4:

CORE OPERATING LIMITS REPORT (Continued)

c. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).
WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary).
(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary).
WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

f. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (W Proprietary).
(Methodology for Specification 2.2.1-Overtemperature ΔT and Overpower ΔT Trip Functions).

g. WCAP-17661-P-A, Revision 1, "Improved RAOC and CAOC F_Q Surveillance Technical Specifications," February 2019, (W Proprietary).
(Methodology for Specification 3.2.2 – Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Attachment 2 – Mark-Ups

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

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

target

b. Within the ~~targe~~ band specified in the COLR about the target flux difference during base load operation.

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

ACTION:

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

1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux – High Trip setpoints to less than or equal 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND**} with the indicated AFD outside of the applicable target band about the target flux differences:

1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

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

*See Special Test Exception 3.10.2
** APL^{ND} is the minimum allowable power level for base load operation and will be specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.11.

Attachment 2 – Mark-Ups

POWER DISTRIBUTION LIMITS

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

Insert A

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$ shall be limited by the following relationships:

$$F_Q(z) \leq \frac{[F_Q^{RTP}]}{P}\,[K(z)] \quad \text{for } P > 0.5$$

$$F_Q(z) \leq \left[\frac{F_Q^{RTP}}{0.5}\right][K(z)] \quad \text{for } P \leq 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

K(z) = the normalized $F_Q(z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(z)$ is demonstrated through core power distribution measurement to be within its limit.

Attachment 2 – Mark-Ups

Insert A

3.2.2 $F_Q(z)$, as approximated by $F_Q^C(z)$ and $F_Q^W(z)$, shall be within the limits specified in the COLR,

and

During Base Load Operation, THERMAL POWER shall be maintained between APL^{ND} and APL^{BL}* or between APL^{ND} and 100% (whichever is most limiting).

Applicability: MODE 1.

Action:

a. With $F_Q^C(z)$ exceeding its limit** #:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q^C(z)$ exceeds the limit within 15 minutes after each $F_Q^C(z)$ determination, and
2. Reduce Power Range Neutron Flux-High trip setpoints at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action a.1 within 72 hours after each $F_Q^C(z)$ determination, and
3. Reduce the Overpower ΔT Trip Setpoints by at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action a.1 within 72 hours after each $F_Q^C(z)$ determination, and
4. Perform SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit of Action a.1.

*APL^{BL} is the maximum allowable power level (if below 100%) for Base Load Operation and its definition will be specified in the CORE OPERATING LIMITS REPORT.

**Required Action a.4 shall be completed whenever this Condition is entered prior to increasing THERMAL POWER above the limit of Required Action a.1.

#SR 4.2.2.3 is not required to be performed if this condition is entered prior to THERMAL POWER exceeding 75% RTP after refueling.

Attachment 2 – Mark-Ups

Insert A (Continued)

b. With $F_Q^W(z)$ exceeding its limit:

1.a. Restore $F_Q^W(z)$ to within limits specified in the COLR within 4 hours, and

1.b. Perform SR 4.2.2.2 and SR 4.2.2.3 within 72 hours*,

Or

2.a. Limit THERMAL POWER to less than RATED THERMAL POWER as specified in the COLR within 4 hours**, and

2.b. Reduce AFD limits as specified in the COLR within 4 hours***, and

2.c. Reduce Power Range Neutron Flux-High trip setpoints at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action b.2.a within the next 72 hours, and

2.d. Reduce the Overpower ΔT Trip Setpoints by at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action b.2.a within 72 hours, and

2.e. Perform SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit of Action b.2.a.

c. If Actions a. or b. are not completed within the provided times, then be in MODE 2 within 6 hours.

*Action b.1.b shall be completed if control rod motion is required to comply with the new operating space implemented by Action b.1.a.

**Action b.2.e shall be completed whenever Action b.2.a is performed prior to increasing THERMAL POWER above the limit of Action b.2.a.

***During Base Load operation, AFD limits do not need to be reduced during Action b.2.b.

Attachment 2 – Mark-Ups

Insert B

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

~~4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:~~

~~a. Using the movable incore detectors to obtain a power distribution map~~

~~1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or~~

~~2. When the Power Distribution Monitoring System (PDMS) is inoperable;~~

~~and increasing the Measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.~~

~~b. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.~~

~~c. Satisfying the following relationship:~~

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \le 0.5$$

~~where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and W(z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.~~

~~d. Measuring $F_Q^M(z)$ according to the following schedule:~~

~~1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or~~

~~2. In accordance with the Surveillance Frequency Control Program, whichever occurs first.~~

~~* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and the core power distribution measurement is obtained.~~

Attachment 2 – Mark-Ups

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. Increase $F_Q^M(z)$ by the appropriate penalty factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive core power distribution measurements indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.2.c. above not being satisfied~~:~~

1. Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)}\right]-1\right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{\left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)}\right]-1\right\} \times 100 \text{ for } P < 0.5$$

Attachment 2 – Mark-Ups

SURVEILLANCE REQUIREMENTS (Continued)

2. One of the following actions shall be taken:

(a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the applicable AFD limits by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2.f.(1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

(b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

(c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

g. The limits specified in Specifications 4.2.2.2.c., 4.2.2.2.e., and 4.2.2.2.f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 10%, inclusive.
2. Upper core region from 90 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(z)}{F_Q^M(z) \times W(z)_{BL}} \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR. The F_Q limit is F_Q^{RTP}. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, $K(z)$, and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

Attachment 2 – Mark-Ups

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} when the Power Distribution Monitoring System (PDMS) is inoperable; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS at any THERMAL POWER greater than APL^{ND}, and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing and measurement uncertainties as specified in the COLR. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a core power distribution measurement has been obtained in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to measurement, and

2. In accordance with the Surveillance Frequency Control Program.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

Attachment 2 – Mark-Ups

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. Increase $F_Q^M(z)$ by the appropriate penalty factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.4.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive core power distribution measurements indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(z)$, or

2. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left\{\left[\frac{F_Q^M(z)xW(z)_{BL}}{\frac{F_Q^{RTP}}{P}xK(z)}\right]-1\right\} x\ 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 10%, inclusive.

2. Upper core region 90 to 100%, inclusive.

4.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained:

a. From a power distribution map

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. From the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

Attachment 2 – Mark-Ups

Insert B

4.2.2.2 $F_Q^C(z)$ shall be verified to be within its limit according to the following schedule:

a. Once after each refueling prior to THERMAL POWER exceeding 75% RATED THERMAL POWER, and
b. Once within 24 hours after achieving equilibrium conditions after exceeding, by at least 10% of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^C(z)$ was last verified, and
c. In accordance with the Surveillance Frequency Control Program.
d. Prior to entering Base Load Operation, in conjunction with target axial flux difference determination after satisfying Specification 4.2.2.4.*

4.2.2.3 $F_Q^W(z)$ shall be verified to be within its limit according to the following schedule:

a. Once after each refueling within 24 hours after achieving equilibrium conditions after THERMAL POWER exceeds 75% RATED THERMAL POWER, and
b. Once within 24 hours after achieving equilibrium conditions after exceeding, by at least 10% of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^W(z)$ was last verified, and
c. In accordance with the Surveillance Frequency Control Program.
d. Prior to entering Base Load Operation, in conjunction with target axial flux difference determination after satisfying Specification 4.2.2.4.*

*Not required if a core power distribution measurement has been obtained in accordance with the Surveillance Frequency Control Program with the THERMAL POWER having been maintained above APL^{ND} for the 24 hours prior to measurement.

Attachment 2 – Mark-Ups

Insert B (continued)

4.2.2.4 For Base Load operation, the following conditions shall be satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by LCO 3.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the target flux difference) during this time period.

b. During Base Load operation, the F_Q surveillances shall be met pursuant to Specifications 4.2.2.2 and 4.2.2.3. If the THERMAL POWER is decreased below APL^{ND} then RAOC operation is required, and the conditions of Specification 4.2.2.4.a shall be satisfied before re-entering Base Load operation.

Attachment 2 – Mark-Ups

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specifications 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirement of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirement of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

Specifications 4.2.2.2 and 4.2.2.3.

a. ~~Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5.~~

b. Specification 4.2.3.2.

Attachment 4

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Dominion Energy South Carolina, Inc.
Virgil C. Summer Nuclear Station Unit 1

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

3.0 AND 4.0 LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENT

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.17 for the WRB-2 DNB correlation.

b. The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

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TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Instrumentation
ΔT_0	$\leq$	Indicated ΔT at RATED THERMAL POWER
K_1	$\leq$	[*]
K_2	$\geq$	[*]/°F
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_{avg}, $\tau_1 \geq [*] secs$, $\tau_2 \leq [*] secs$
T	=	Average temperature, °F
T'	$\leq$	Indicated T_{avg} at RATED THERMAL POWER, T' $\leq$ [*]°F
K_3	$\geq$	[*]/psi
P	=	Pressurizer pressure, psig
P'	$\geq$	[*] psig, Nominal RCS operating pressure
S	=	Laplace transform operator, sec^{-1}
$f_1(\Delta I)$	=	[*] {[*] – (q_t -q_b)} when q_t -$q_b \leq$ [*]% RTP 0% of RTP when [*]% RTP < q_t -$q_b \leq$ [*]%RTP [*] {(q_t -q_b) – [*]} when q_t -q_b > [*]% RTP

The values denoted with [*] are specified in the COLR.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.2 percent ΔT Span.
NOTE 3: OVERPOWER ΔT

$$\Delta T \leq \Delta T_O \left[K_4 - K_5 \frac{(\tau_3 S)}{(1+\tau_3 S)} T - K_6 [T - T'']\right]$$

Where:	ΔT	=	as defined in Note 1
	ΔT_O	=	as defined in Note 1
	K_4	$\leq$	[*]
	K_5	$\geq$	[*]/°F for increasing T_{avg}, K_5 = [*]/°F for decreasing T_{avg}
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \geq$ [*]secs
	K_6	$\geq$	[*] /°F for T > T″, and K_6 = [*] for T $\leq$ T″
	T	=	as defined in Note 1
	T″	=	Indicated T_{avg} at RATED THERMAL POWER, T″ $\leq$ [*]°F
	S	=	as defined in Note 1

The values denoted with [*] are specified in the COLR.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.3 percent ΔT Span.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be demonstrated to be within the limits specified in the COLR.

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Surveillance Requirement 4.1.1.1.2 with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to the required value:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

   1. Reactor coolant system boron concentration,
   2. Control rod position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

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REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limits specified in the COLR at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300° F.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 14,000 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 453,800 gallons,

2. A minimum boron concentration of 2300 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least the limits specified in the COLR at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

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

b. With more than one full length rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With more than one full length rod inoperable due to a rod control urgent failure alarm or obvious electrical problem in the rod control system for greater than 72 hours, be in HOT STANDBY within the following 6 hours.

d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT (COLR); the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

*See Special Test Exceptions 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement as defined in the COLR is satisfied. POWER OPERATION may then continue provided that:

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

   b) The SHUTDOWN MARGIN requirement as defined in the COLR is determined at least once per 12 hours.

   c) A core power distribution measurement is obtained and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

   d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation as specified in the CORE OPERATING LIMITS REPORTS (COLR) figure entitled RCS Total Flow Rate Versus R For Three Loop Operation.

Where:

a. R = $\dfrac{F^{N}_{\Delta H}}{F^{RTP}_{\Delta H}\left[1.0 + PF_{\Delta H}\left(1.0 - P\right)\right]}$,

b. P = $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^{N}_{\Delta H}$ = Measured values of $F^{N}_{\Delta H}$ obtained by

1. Using the movable incore detectors to obtain a power distribution map when THERMAL POWER is <= 25% but > 5% of RATED THERMAL POWER, or when PDMS is inoperable, and

2. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER.

The measured values of $F^{N}_{\Delta H}$ shall be increased by the applicable $F^{N}_{\Delta H}$ measurement uncertainties as specified in the COLR, and used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainty for flow,

d. $F^{RTP}_{\Delta H}$ = The $F^{N}_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

e. $PF_{\Delta H}$ = The Power Factor Multiplier specified in the COLR.

APPLICABILITY : MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

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

b. Within 24 hours of initially being outside the above limits, verify through a core power distribution measurement and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

POWER DISTRIBUTION LIMITS

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits specified in the COLR.

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure*

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limits, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters listed in Specification 3.2.5 shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED POWER.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements as defined in the COLR, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to the limits specified in the COLR.

APPLICABILITY: MODE 6 * with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to the limits specified in the COLR, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

4.9.1.3 The following valves shall be verified locked closed ** in accordance with the Surveillance Frequency Control Program: 8430, 8454, 8441 and 8439.

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* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** Valves may be opened under administrative control to add borated makeup.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement as defined in the COLR may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the required SHUTDOWN MARGIN as defined in the COLR is restored.

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the required SHUTDOWN MARGIN as defined in the COLR is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits as defined in the COLR.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specifications 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3.

b. Specification 4.2.3.2.

6.9.1.9 Not used.

6.9.1.10 Not used.

CORE OPERATING LIMITS REPORT

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Reactor Core Safety Limits for Specification 2.1.1,

b. Overtemperature ΔT and Overpower ΔT for Specification 2.2.1,

c. Shutdown Margin for Modes 1 and 2 for Specification 3/4.1.1.1,

d. Shutdown Margin for Modes 3, 4, and 5 for Specification 3/4.1.1.2,

e. Moderator Temperature Coefficient for Specification 3/4.1.1.3,

f. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

g. Control Rod Insertion Limits for Specification 3/4.1.3.6,

h. Axial Flux Difference Limits for Specification 3/4.2.1,

i. Heat Flux Hot Channel Factor Limits for Specification 3/4.2.2,

j. RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3,

k. DNB Parameters for Specification 3/4.2.5,

l. Refueling Operations Boron Concentration for Specification 3/4.9.1.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, “WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY,” July 1985 (W Proprietary).

(Methodology for Specifications 2.1.1 - Reactor Core Safety Limits, 3.1.1.1 & 3.1.1.2- Shutdown Margin, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 - DNB Parameters, and 3.9.1 – Refueling Operations Boron Concentration.)

b. WCAP-10216-P-A, Rev. 1A, “RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION,” February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control).)

CORE OPERATING LIMITS REPORT (Continued)

c. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary).

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

f. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (W Proprietary).

(Methodology for Specification 2.2.1-Overtemperature ΔT and Overpower ΔT Trip Functions).

g. WCAP-17661-P-A, Revision 1, "Improved RAOC and CAOC F_Q Surveillance Technical Specifications," February 2019, (W Proprietary).

(Methodology for Specification 3.2.2 – Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

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

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

b. Within the target band specified in the COLR about the target flux difference during base load operation.

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

ACTION:

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

   1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux – High Trip setpoints to less than or equal 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AFD outside of the applicable target band about the target flux differences:

   1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

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

*See Special Test Exception 3.10.2
** APL^{ND} is the minimum allowable power level for base load operation and will be specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.11.

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$, as approximated by $F_Q^C(z)$ and $F_Q^W(z)$, shall be within the limits specified in the COLR,

and

During Base Load Operation, THERMAL POWER shall be maintained between APL^{ND} and APL^{BL*} or between APL^{ND} and 100% (whichever is most limiting).

APPLICABILITY: MODE 1.

ACTION:

a. With $F_Q^C(z)$ exceeding its limit**#:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q^C(z)$ exceeds the limit within 15 minutes after each $F_Q^C(z)$ determination, and
2. Reduce Power Range Neutron Flux-High trip setpoints at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action a.1 within 72 hours after each $F_Q^C(z)$ determination, and
3. Reduce the Overpower ΔT Trip Setpoints by at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action a.1 within 72 hours after each $F_Q^C(z)$ determination, and
4. Perform SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit of Action a.1.

*APL^{BL} is the maximum allowable power level (if below 100%) for Base Load Operation and its definition will be specified in the CORE OPERATING LIMITS REPORT.

**Required Action a.4 shall be completed whenever this Condition is entered prior to increasing THERMAL POWER above the limit of Required Action a.1.

#SR 4.2.2.3 is not required to be performed if this condition is entered prior to THERMAL POWER exceeding 75% RTP after refueling.

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION (Continued)

b. With $F_Q^W(z)$ exceeding its limit:

1.a. Restore $F_Q^W(z)$ to within limits specified in the COLR within 4 hours, and

1.b. Perform SR 4.2.2.2 and SR 4.2.2.3 within 72 hours*,

Or

2.a. Limit THERMAL POWER to less than RATED THERMAL POWER as specified in the COLR within 4 hours**, and

2.b. Reduce AFD limits as specified in the COLR within 4 hours***, and

2.c. Reduce Power Range Neutron Flux-High trip setpoints at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action b.2.a within the next 72 hours, and

2.d. Reduce the Overpower ΔT Trip Setpoints by at least 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action b.2.a within 72 hours, and

2.e. Perform SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit of Action b.2.a.

c. If Actions a. or b. are not completed within the provided times, then be in MODE 2 within 6 hours.

*Action b.1.b shall be completed if control rod motion is required to comply with the new operating space implemented by Action b.1.a.

**Action b.2.e shall be completed whenever Action b.2.a is performed prior to increasing THERMAL POWER above the limit of Action b.2.a.

***During Base Load operation, AFD limits do not need to be reduced during Action b.2.b.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^C(z)$ shall be verified to be within its limit according to the following schedule:

a. Once after each refueling prior to THERMAL POWER exceeding 75% RATED THERMAL POWER, and

b. Once within 24 hours after achieving equilibrium conditions after exceeding, by at least 10% of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^C(z)$ was last verified, and

c. In accordance with the Surveillance Frequency Control Program.

d. Prior to entering Base Load Operation, in conjunction with target axial flux difference determination after satisfying Specification 4.2.2.4.*

4.2.2.3 $F_Q^W(z)$ shall be verified to be within its limit according to the following schedule:

a. Once after each refueling within 24 hours after achieving equilibrium conditions after THERMAL POWER exceeds 75% RATED THERMAL POWER, and

b. Once within 24 hours after achieving equilibrium conditions after exceeding, by at least 10% of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^W(z)$ was last verified, and

c. In accordance with the Surveillance Frequency Control Program.

d. Prior to entering Base Load Operation, in conjunction with target axial flux difference determination after satisfying Specification 4.2.2.4.*

*Not required if a core power distribution measurement has been obtained in accordance with the Surveillance Frequency Control Program with the THERMAL POWER having been maintained above APL^{ND} for the 24 hours prior to measurement.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.4 For Base Load operation, the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by LCO 3.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the target flux difference) during this time period.

b. During Base Load operation, the F_Q surveillances shall be met pursuant to Specification 4.2.2.2 and 4.2.2.3. If the THERMAL POWER is decreased below APL^{ND} then RAOC operation is required, and the conditions of Specification 4.2.2.4.a shall be satisfied before re-entering Base Load operation.

Attachment 5

ASSOCIATED TECHNICAL SPECIFICATION BASES CHANGES

(FOR INFORMATION ONLY)

Dominion Energy South Carolina, Inc.
Virgil C. Summer Nuclear Station Unit 1

Attachment 1 – Mark-Ups

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

"A figure provided in the COLR shows..."

~~The curves of Figure 2.1-1 show~~ the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

SAFETY LIMITS

BASES

REACTOR CORE (Continued)

Replace with Insert 1

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.56 (includes measurement uncertainty) and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.56\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants, 1971 Edition which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are also designed to Section III of the ASME Code for Nuclear Power Plants, 1971 Edition which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

Attachment 1 – Mark-Ups

INSERT 1:

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and
b. There must be at least 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various Reactor Protection System (RPS) functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

Attachment 1 – Mark-Ups

LIMITING SAFETY SYSTEM SETTINGS

BASES

Intermediate and Source Range, Nuclear Flux (Continued)

uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The purpose of the P-6 setpoint, which is above the lower end of the intermediate range scale, is to give the operators sufficient time to actuate the source range reactor trip block. The Intermediate Range channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds) plus thermal delays associated with the RTD's mounted in the thermowells (about 5 seconds), and pressure is within the range between the Pressurizer high and low pressure trips. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, 2) pressurizer pressure, and 3) axial power distribution. With normal axial power distribution, this reactor trip limit is always below the core safety limit as ~~shown in Figure 2.1-1.~~ If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

revise to "...as specified in the COLR."

Overpower ΔT

The Overpower delta T trip provides assurance of fuel integrity (e.g., no fuel melting and less than 1 percent cladding strain) under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and 2) rate of change of temperature for dynamic compensation for piping and thermal delays from the core to the loop temperature detectors to ensure that the allowable heat generation rate (Kw/ft) is not exceeded. The overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

Pressurizer Pressure

In each of the pressure channels, there are two independent bistables, each with its own trip setting to provide for a high and low pressure trip thus limiting the pressure range in which reactor operation is permitted. The low setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

Attachment 1 – Mark-Ups

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2 the most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN ~~of 1.77 percent delta k/k~~ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4 and 5 the most limiting accident is a boron dilution accident. The SHUTDOWN MARGIN is varied as a function of average RCS boron concentration in order to provide adequate protection in these MODES.

replace with "...as defined in the COLR..."

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide the required SHUTDOWN MARGIN ~~from expected operating conditions of 1.77% delta k/k or as required by Figure 3.1-3~~ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs from full power equilibrium xenon conditions and is satisfied by 13269 gallons of 7000 ppm borated water from the boric acid storage tanks or 98631 gallons of 2300 ppm borated water from the refueling water tank.

replace with "...specified within the COLR, as applicable,..."

With the RCS temperature below 200°F, one injection system is acceptable without failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN ~~of 1 percent delta k/k or as required by Figure 3.1-3~~ after xenon decay and cooldown from 200°F to 140°F. This condition is satisfied by either 2000 gallons of 7000 ppm borated water from the boric acid storage tanks or 23266 gallons of 2300 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

Attachment 1 – Mark-Ups

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.3.6 CONTROL ROD INSERTION LIMITS

The limits on control banks sequence, overlap, and physical insertion as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

replace with "...as defined in the COLR..."

If the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits. If control bank sequence or overlap configuration is not in accordance with the COLR, while in Modes 1 or 2, then verify SDM ~~per TS 3.1.1.1~~ within 1 hour or comply with the associated TS 3.1.1.1 action statements. Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

If the above action cannot be completed within the associated completion times, the plant must be brought to HOT STANDBY, where the LCO is not applicable. The allowed completion time of 6 hours is reasonable, based on operating experience, for reaching HOT STANDBY from full power conditions in an orderly manner and without challenging plant systems.

Attachment 1 – Mark-Ups

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

For measurements obtained using the Power Distribution Monitoring System (PDMS), the appropriate measurement uncertainty is determined using the measurement uncertainty methodology contained in WCAP-12472-P-A. The cycle and plant specific uncertainty calculation information needed to support the PDMS calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty, and apply a 3% allowance for manufacturing tolerance.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, W(z) or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$ is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. If two most recent $F_Q(z)$ evaluations show an increase in the maximum value of $\left[\frac{F_Q^M(z)}{K(z)}\right]$ over the core height (z), it is not guaranteed that $F_Q^M(z)$ will remain within the transient limit during the following surveillance interval. Technical Specification Surveillance Requirement 4.2.2 requires that $F_Q^M(z)$ be increased by a penalty factor as specified in the COLR and compared to the transient $F_Q(z)$ limit. If there is insufficient margin, i.e., this value exceeds the limit, the $F_Q^M(z)$ must be measured once per 7 EFPD until either $F_Q^M(z)$ increased by the penalty factor is within the transient limit, or two successive power distribution measurements indicate the maximum value of $\left[\frac{F_Q^M(z)}{K(z)}\right]$ over the core height (z) has not increased. The W(z) and W(z)$_{BL}$ functions described above for normal operation are specified in the CORE OPERATI Delete EPORT (COLR) per Specification 6.9.1.11.

When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of the RCS Total Rate Versus R figure in the COLR. Measurement errors ~~of 2.1%~~ for RCS total flow rate~~, including 0.1% for feedwater venturi fouling,~~ have been allowed for in determining the limits of RCS Total Flow Rate Versus R Figure in the COLR.

For $F_{\Delta H}^N$ measurements obtained from a full core flux map taken with the incore detector flux mapping system, a 4% measurement uncertainty allowance should be applied to the measured $F_{\Delta H}^N$ value prior to comparison with the limits of the RCS Total Flow Rate Versus R Figure in the COLR. The appropriate measurement uncertainty for $F_{\Delta H}^N$ measurements obtained using the Power Distribution Monitoring System (PDMS) is determined using the uncertainty methodology described in WCAP-12472-P-A. The cycle and plant specific uncertainty calculation information needed to support the PDMS uncertainty calculation is

Attachment 1 – Mark-Ups

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F^N_{\Delta H}$ value.

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

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt power ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or the PDMS are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum of DNBR in the core at or above the design limit throughout each analyzed transient. ~~The maximum indicated T_{avg} limit of 589.2°F and the minimum indicated pressure limit of 2206 psig correspond to analytical limits of 591.4°F and 2185 psig respectively, read from control board indications.~~

Replace with:
The DNB related parameters are specified in the COLR.

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

Attachment 1 – Mark-Ups

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration ~~value of 2000 ppm or greater~~ specified in the COLR includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum time of 72 hours for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum decay time of 72 hours is consistent with the assumptions used in the accident analysis.

The tabulated hold times associated with Component Cooling Water (CCW) temperature ensure that the spent fuel heat load is reduced sufficiently to allow the spent fuel pool cooling system to maintain the bulk pool temperature below 170°F. These hold times ensure that adequate cooling is provided to the Spent Fuel Pool under the highest possible heat load conditions. The hold times are based on the performance of the cooling system, which is dependent upon CCW temperature and recognizes that the spent fuel pool cooling system is capable of increased flow rates up to 2400 gpm during single loop operation. This higher flow rate may be required when only a single cooling loop is operable during a refueling outage.

The CCW temperature limits defined in Figure 3.9-1 are adjusted for uncertainty in the implementing procedure.

3/4.9.4 DELETED BY AMENDMENT 183

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^{N}_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on the RCS Total Flow Rate Versus R figure in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if core power is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^{N}_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in the RCS Total Flow Rate Versus R figure in the COLR, accounts for $F^{N}_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^{N}_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

~~When a F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.~~

Attachment 2 – Mark-Ups

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

For measurements obtained using the Power Distribution Monitoring System (PDMS), the appropriate measurement uncertainty is determined using the measurement uncertainty methodology contained in WCAP-12472-P-A. The cycle and plant specific uncertainty calculation information needed to support the PDMS calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty, and apply a 3% allowance for manufacturing tolerance.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, W(z) or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$ is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. If two most recent $F_Q(z)$ evaluations show an increase in the maximum value of $\left[\frac{F_Q^M(z)}{K(z)}\right]$ over the core height (z), it is not guaranteed that $F_Q^M(z)$ will remain within the transient limit during the following surveillance interval. Technical Specification Surveillance Requirement 4.2.2 requires that $F_Q^M(z)$ be increased by a penalty factor as specified in the COLR and compared to the transient $F_Q(z)$ limit. If there is insufficient margin, i.e., this value exceeds the limit, the $F_Q^M(z)$ must be measured once per 7 EFPD until either $F_Q^M(z)$ increased by the penalty factor is within the transient limit, or two successive power distribution measurements indicate the maximum value of $\left[\frac{F_Q^M(z)}{K(z)}\right]$ over the core height (z) has not increased. The W(z) and $W(z)_{BL}$ functions described above for normal operation are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.11.

When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of the RCS Total Rate Versus R figure in the COLR. Measurement errors ~~of 2.1%~~ for RCS total flow rate, ~~including 0.1% for feedwater venturi fouling,~~ have been allowed for in determining the limits of RCS Total Flow Rate Versus R Figure in the COLR.

Delete

For $F_{\Delta H}^N$ measurements obtained from a full core flux map taken with the incore detector flux mapping system, a 4% measurement uncertainty allowance should be applied to the measured $F_{\Delta H}^N$ value prior to comparison with the limits of the RCS Total Flow Rate Versus R Figure in the COLR. The appropriate measurement uncertainty for $F_{\Delta H}^N$ measurements obtained using the Power Distribution Monitoring System (PDMS) is determined using the uncertainty methodology described in WCAP-12472-P-A. The cycle and plant specific uncertainty calculation information needed to support the PDMS uncertainty calculation is

Attachment 2 – Mark-Ups

POWER DISTRIBUTION LIMIT

BASES

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F^N_{\Delta H}$ value.

Insert F

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

*Deletion is included as part of Insert F

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt power ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or the PDMS are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum of DNBR in the core at or above the design limit throughout each analyzed transient. The maximum indicated T_{avg} limit of 589.2°F and the minimum indicated pressure limit of 2206 psig correspond to analytical limits of 591.4°F and 2185 psig respectively, read from control board indications.

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

Attachment 2 – Mark-Ups

INSERT F: Attachment 2 Pages 14-26

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.3.6, "CONTROL ROD INSERTION LIMITS," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using PDMS (Power Distribution Measurement System). PDMS refers to an incore flux map used to measure $F_Q(Z)$ and subsequently calibrate BEACON, or a surveillance performed by BEACON between flux maps within the timeframe allowed by the Surveillance Frequency Control Program. PDMS measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three-dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ which are present during nonequilibrium situations, such as load following or power ascension.

To account for these possible variations, the elevation dependent measured planar radial peaking factors, $F_{XY}(z)$, are increased by an elevation dependent factor, T(z), that accounts for the expected maximum values of the transient axial power shapes postulated to occur during RAOC operation. Thus, T(z) accounts for the worst-case non-equilibrium power shapes that are expected for the assumed RAOC operating space.

The operating space is defined as the combination of AFD, THERMAL POWER, and Control Bank Insertion Limits assumed in the calculation of a particular T(z) function (or W(z) for Base Load operation). The T(z) and W(z) factors are directly dependent on the AFD Limits and Control Bank Insertion Limit assumptions. Different sets of T(z) or W(z) functions that reflect different operating space assumptions are generated in accordance with Reference 6. If the limit on $F_Q(z)$ is exceeded, a more restrictive operating space may be implemented to gain margin for future non-equilibrium operation.

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the allowed short time period will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24-hour waiting period at a power level above APL^{ND} and within the RAOC operating space is necessary. During this time period, load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period, extended Base Load operation is permissible.

APPLICABLE SAFETY LIMITS

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1),

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition,

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2), and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting of the LOCA-related criteria.

$F_Q(z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(z) \leq (CFQ / P) \quad \text{for } P > 0.5$$
$$F_Q(z) \leq (CFQ / 0.5) \quad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(z)$ limit at RTP provided in the COLR, and
P = THERMAL POWER / RTP

For this facility, the actual values of CFQ are given in the COLR; however, CFQ is normally on the order of 2.50.

For Relaxed Axial Offset Control and Base Load operation, $F_Q(z)$ is approximated by $F_Q^C(z)$ and $F_Q^W(z)$. Thus, both $F_Q^C(z)$ and $F_Q^W(z)$ must meet the preceding limits on $F_Q(z)$.

An $F_Q^C(z)$ evaluation requires a PDMS measurement in MODE 1. From the PDMS results, the measured value ($F_Q^M(z)$) of $F_Q(z)$ is obtained. Then,

$F_Q^C(z) = F_Q^M(z)$ [U_F]

where U_F is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

$F_Q^C(z)$ is an excellent approximation for $F_Q(z)$ when the reactor is at the steady state power at which the PDMS measurement was taken.

The expression for $F_Q^W(z)$ during Relaxed Axial Offset Control operation is:

$$F_Q^W(z) = F_{XY}^M(z) * \frac{T(z)}{P} * A_{XY}(z) * Rj * U_F$$

The expression for $F_Q^W(z)$ during Base Load operation is:

$$F_Q^W(z) = F_Q^C(z) * \frac{W(z)}{P} * A_Q(z) * Rj * U_F$$

The various factors in these expressions are defined below:

$F_{XY}^M(z)$ is the measured radial peaking factor at axial location z and is equal to the value of $F_Q^M(z) / P^M(z)$, where $P^M(z)$ is the measured core average axial power shape.

T(z) and W(z) are the cycle and burnup dependent functions, calculated in accordance with WCAP-17661-P-A (Ref. 6), which account for power distribution transients encountered during non-equilibrium normal operation. T(z) and W(z) functions are specified for each analyzed operating space (i.e., each unique combination of AFD limits and Control Bank Insertion Limits). The T(z) and W(z) functions account for the limiting non-equilibrium axial power shapes postulated to occur during normal operation for each operating space. Limiting power shapes at both full and reduced power operation are considered in determining the maximum values of T(z) and W(z). The T(z) and W(z) functions also account for the following effects: (1 – T(z) only) the presence of spacer grids in the fuel assembly, (2) the increase in radial peaking in rodded core planes due to the presence of control rods during non-equilibrium normal operation, (3) the increase in radial peaking that occurs during part-power operation due to reduced fuel and moderator temperatures, and (4) the increase in radial peaking due to non-equilibrium xenon effects. The T(z) functions are normally calculated assuming that the Surveillance is performed at nominal RTP conditions with all shutdown and control rods fully withdrawn, i.e., all rods out (ARO). The W(z) functions are normally calculated assuming that the Surveillance is performed at the Target Axial Offset core conditions. Surveillance-specific T(z) or W(z) values may be generated for a given surveillance core condition.

P is the THERMAL POWER / RTP.

$A_{XY}(z)$ (or, analogously, $A_Q(z)$) is a function that adjusts the $F_Q^W(z)$ Surveillance for differences between the reference core condition assumed in generating the T(z) (or W(z)) function and the actual core condition that exists when the Surveillance is performed. Normally this reference core condition is 100% RTP, all rods out, and equilibrium xenon. For simplicity, $A_{XY}(z)$ or $A_Q(z)$ may be assumed to be 1.0, as this will typically result in an accurate $F_Q^W(z)$ Surveillance result for a Surveillance that is performed at or near the reference core condition, and an underestimation of the available margin to the F_Q limit for Surveillances that are performed at core conditions different from the reference condition. Alternatively, the $A_{XY}(z)$ or $A_Q(z)$ function may be calculated using the NRC-approved methodology in Reference 6.

During Base Load operation, setting $A_Q(z)$ to 1.0 is acceptable for both nominal surveillances performed at the reference core conditions and off-nominal surveillances performed at less than 90% RTP where the Axial Offset (AO) is maintained near the target (within 1.5 percent) per Reference 6. There must also be reasonable assurance that the limiting $F_Q^W(z)$ does not lie within a rodded elevation at the time of surveillance, if applicable.

U_F is a factor that accounts for fuel manufacturing tolerances and measurement uncertainty. Historically, it has been set at 1.0815, but a real-time value calculated by BEACON may be used instead.

R_j is a cycle and burnup dependent analytical factor generated in accordance with Reference 6 that accounts for potential increases in $F_Q^W(z)$ between Surveillances. R_j values are provided for each operating space.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

APPLICABILITY

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

ACTIONS

a.1

Reducing THERMAL POWER by at least 1% RTP for each 1% by which $F_Q^C(z)$ exceeds its limit maintains an acceptable absolute power density. $F_Q^C(z)$ is $F_Q^M(z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(z)$ is the measured value of $F_Q(z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. The maximum allowable power level initially determined by Action a.1 may be affected by subsequent determinations of $F_Q^C(z)$ and would require power reductions within 15 minutes of the $F_Q^C(z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in $F_Q^C(z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

If an F_Q surveillance is performed at 100% RTP conditions, and both $F_Q^C(z)$ and $F_Q^W(z)$ exceed their limits, the option to reduce the THERMAL POWER limit in accordance with proposed Action b.2.1 instead of implementing a new operating space in accordance with proposed Action b.1, will result in a further power reduction after Action a.1 has been completed. This further power reduction would be permitted to occur over the next 4 hours. In the event the evaluated THERMAL POWER reduction in the COLR for proposed Action b.2.1 does not result in a further power reduction (for example, if both Condition a and Condition b were entered at less than 100% RTP conditions), then the THERMAL POWER level established as a result of completing Action a.1 will take precedence, and will establish the effective operating power level limit for the unit until both Conditions ‘a’ and ‘b’ are exited.

a.2

A reduction of the Power Range Neutron Flux – High trip setpoints by $\geq$ 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action a.1, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Action a.1. The maximum allowable Power Range Neutron Flux – High trip setpoints initially determined by Action a.2 may be affected by subsequent determinations of $F_Q^C(z)$ and would require Power Range Neutron Flux- High trip setpoint reductions within 72 hours of the $F_Q^C(z)$ determination, if necessary to comply with the decreased maximum allowable Power Range Neutron Flux – High trip setpoints. Decreases in $F_Q^C(z)$ would allow increasing the maximum allowable Power Range Neutron Flux – High trip setpoints.

a.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by ≥1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER by Action a.1, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Action a.1. The maximum allowable Overpower ΔT trip setpoints initially determined by Action a.3 may be affected by subsequent determinations of F_Q (z) and would require Overpower ΔT trip setpoint reductions within 72 hours of the $F_Q^C(z)$ determination, if necessary to comply with the decreased maximum allowable Overpower ΔT trip setpoints. Decreases in $F_Q^C(z)$ would allow increasing the maximum allowable Overpower ΔT trip setpoints.

a.4

Verification that $F_Q^C(z)$ has been restored to within its limit, by performing SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit imposed by Action a.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition ‘a’ is modified by 2 Notes, the first of which requires Action a.4 to be performed whenever the Condition is entered prior to increasing THERMAL POWER above the limits of Action a.1. The second Note states that SR 4.2.2.3 is not required to be performed if this Condition is entered prior to THERMAL POWER exceeding 75% RTP after a refueling. This ensures that SR 4.2.2.2 and SR 4.2.2.3 (if required) will be performed prior to increasing THERMAL POWER above the limit of Action a.1 even when Condition ‘a’ is exited prior to performing Action a.4. Performance of SR 4.2.2.2 and SR 4.2.2.3 are necessary to assure $F_Q(z)$ is properly evaluated prior to increasing THERMAL POWER.

b.1.a

If it is found that the maximum calculated value of $F_Q(z)$ that can occur during normal maneuvers, $F_Q^W(z)$, exceeds it specified limits, there exists a potential for $F_Q^C(z)$ to become excessively high if a normal operational transient occurs. Action b.1.a can be achieved by implementing a more restrictive operating space, as specified in the COLR, within the allowed Completion Time of 4 hours. This will restrict the AFD such that peaking factor limits will not be exceeded during non-equilibrium normal operation. Several operating spaces, representing successively smaller AFD envelopes and, optionally, shallower Control Bank Insertion Limits, may be specified in the COLR. The corresponding T(z) functions for these operating spaces can be used to determine which operating space will result in acceptable non-equilibrium operation within the F_Q limit. Note that the implemented operating space may be a RAOC operating space or Base Load CAOC operating space, provided SR 4.2.2.4 is satisfied prior to entering Base Load operation.

b.1.b

If it is found that the maximum calculated value of $F_Q(z)$ that can occur during normal maneuvers, $F_Q^W(z)$, exceeds its specified limits, there exists a potential for $F_Q^C(z)$ to become excessively high if a normal operational transient occurs. As discussed above, Action b.1.a allows a new operating space to be implemented as a method of restoring $F_Q^W(z)$ to within its limits. A note on Action b.1.b also requires that SR 4.2.2.2 and SR 4.2.2.3 be performed if control rod motion occurs as a result of implementing the new operating space in accordance with Action b.1.a. The performance of SR 4.2.2.2 and SR 4.2.2.3 is necessary to assure $F_Q(z)$ is properly evaluated after any rod motion resulting from the implementation of a new operating space in accordance with Action b.1.a.

b.2.a and b.2.b

When $F_Q^W(z)$ exceeds its limit, Action b.2.a may be implemented instead of Action b.1. Action b.2.a limits THERMAL POWER to less than RATED THERMAL POWER by the amount specified in the COLR. b.2.b requires reductions in the AFD limits by the amount specified in the COLR. This maintains an acceptable absolute power density relative to the maximum power density value assumed in the safety analyses.

A Note on b.2.b provides an exception; during Base Load operation, AFD limits do not need to be reduced. This is in line with WCAP-17661-P-A (Reference 6) and is due to Base Load being a CAOC operating space, which has different AFD targets and operational strategy.

If the required $F_Q^W(z)$ margin improvement exceeds the margin improvement available from the pre-analyzed THERMAL POWER and AFD reductions provided in the COLR, then THERMAL POWER must be further reduced to less than or equal to 50% RTP. In this case, reducing THERMAL POWER to less than or equal to 50% RTP will provide additional margin in the transient F_Q by the required change in THERMAL POWER and the increase in the F_Q limit. This will ensure that the F_Q limit is met during transient operation that may occur at or below 50% RTP.

The Completion Time of 4 hours for both b.2.a and b.2.b provides an acceptable time to reduce the THERMAL POWER and AFD limits in an orderly manner to preclude entering an unacceptable condition during future non-equilibrium operation. The limit on THERMAL POWER initially determined by Action b.2.a may be affected by subsequent determinations of $F_Q^W(z)$ and would require power reductions within 4 hours of the $F_Q^W(z)$ determination, if necessary to comply with the decreased THERMAL POWER limit.

Decreases in $F_Q^W(z)$ would allow increasing the THERMAL POWER limit and increasing THERMAL POWER up to this revised limit.

Action b.2.a is modified by a Note that states Action b.2.e shall be completed whenever Action b.2 is performed prior to increasing THERMAL POWER above the limit of Action b.2.a. Action b.2.e requires the performance of SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the limit established by Action b.2.a. The Note ensures that the SRs will be performed even if Condition b may be exited prior to performing Action b.2.e. The performance of SR 4.2.2.2 and SR 4.2.2.3 is necessary to assure $F_Q(z)$ is properly evaluated prior to increasing THERMAL POWER.

If an F_Q surveillance is performed at 100% RTP conditions, and both $F_Q^C(z)$ and $F_Q^W(z)$ exceed their limits, the option to reduce the THERMAL POWER limit in accordance with proposed Action b.2 instead of implementing a new operating space in accordance with proposed Action b.1 will result in a further power reduction after Action a.1 has been completed. However, this further power reduction would be permitted to occur over the next 4 hours. In the event the evaluated THERMAL POWER reduction in the COLR for proposed Action b.2 did not result in a further power reduction (for example, if both Condition ‘a’ and Condition b were entered at less than 100% RTP conditions), then the THERMAL POWER level established as a result of completing Action a.1 will take precedence, and will establish the effective operating power level limit for the unit until both Conditions A and B are exited.

b.2.c

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥1% for each 1% by which the maximum allowable power reduced; is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in the THERMAL POWER limit and AFD limits in accordance with Action b.2.a and b.2.b.

b.2.d

Reduction in the Overpower ΔT trip setpoints value by ≥1% for each 1% by which the maximum allowable power is reduced is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in the THERMAL POWER limit and AFD limits in accordance with Action b.2.

b.2.e

Verification that $F_Q^W(z)$ has been restored to within its limit, by performing SR 4.2.2.2 and SR 4.2.2.3 prior to increasing THERMAL POWER above the maximum allowable power limit imposed by Action b.2.a ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Action c

If Actions a.1 through a.4 or b.1 through b.2.e are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

4.2.2.2

Verification that $F_Q^C(z)$ is within its specified limits involves increasing $F_Q^M(z)$ to allow for manufacturing tolerance and measurement uncertainties to obtain $F_Q^C(z)$. Specifically, $F_Q^M(z)$ is the measured value of $F_Q(z)$ obtained from PDMS measurement results and

$F_Q^C(z) = F_Q^M(z)*U_F$ (Reference 4). $F_Q^C(z)$ is then compared to its specified limits.

The limit with which $F_Q(z)$ is compared varies inversely with power above RTP. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that some determination of $F_Q^C(z)$ is made prior to achieving a significant power level where the peak linear heat rate could approach the limits assumed in the safety analyses.

If THERMAL POWER has been increased by ≥ 10% RTP since the initial or most recent determination of $F_Q^C(z)$, another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(z)$ values are being reduced sufficiently with power increase to stay within the LCO limits). Equilibrium conditions are achieved when the core is sufficiently stable at the intended operation conditions required to perform the surveillance.

The allowance of up to 24 hours after achieving equilibrium conditions at the increased THERMAL POWER level to complete the next $F_Q^C(z)$ surveillance applies to situations where the $F_Q^C(z)$ has already been measured at least once at a reduced THERMAL POWER level. The observed margin in the previous surveillance will provide assurance that increasing power up to the next plateau will not exceed the F_Q limit, and that the core is behaving as designed.

This Frequency condition is not intended to require verification of these parameters after every 10% increase in RTP above the THERMAL POWER at which the last verification was performed. It only requires verification after a THERMAL POWER is achieved for extended operation (to equilibrium conditions) that is 10% higher than the THERMAL POWER at which $F_Q^C(z)$ was last measured.

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

SR 4.2.2.3

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are conservatively calculated by considering a wide range of unit maneuvers in normal operation.

The measured $F_Q(z)$ can be determined through a synthesis of the measured planar radial peaking factors, $F_{XY}^M(z)$, and the measured core average axial power shape, $P^M(z)$. Thus, $F_Q^C(z)$ is given by the following expression:

$$F_Q^C(z) = F_{XY}^M(z)\ P^M(z) * U_F = F_Q^M(z) * U_F$$

For RAOC operation, the analytical T(z) functions, generated in accordance with Reference 6 for each RAOC operating space, are used together with the measured $F_{XY}(z)$ values to estimate $F_Q(z)$ for nonequilibrium operation within the operating space. When the $F_{XY}(z)$ values are measured at HFP ARO conditions ($A_{XY}(z)$ equals 1.0), $F_Q^W(z)$ is given by the following expression:

$$F_Q^W(z) = F_{XY}^M(z) * T(z) * Rj * U_F$$

For CAOC (Base Load) operation, W(z) functions generated in accordance with Reference 6 are used together with the measured $F_Q(z)$ values to estimate $F_Q(z)$ for nonequilibrium operation within the operating space. When the $F_Q(z)$ values are measured at HFP ARO conditions ($A_Q(z)$ equals 1.0), $F_Q^W(z)$ is given by the following expression:

$$F_Q^W(z) = F_Q^M(z) * W(z) * Rj * U_F$$

Non-equilibrium operation can result in significant changes to the axial power shape. To a lesser extent, non-equilibrium operation can increase the radial peaking factors, $F_{XY}(z)$, through control rod insertion and through reduced Doppler and moderator feedback at part-power conditions.

The T(z) functions quantify these effects for the range of power shapes, control rod insertion, and power levels characteristic of the operating space. Multiplying T(z) by the measured full power, unrodded $F_{XY}^M(z)$ value, and the factor that accounts for manufacturing and measurement uncertainties gives $F_Q^W(z)$, which is the maximum total peaking factor postulated for non-equilibrium operation.

The limit with which $F_Q^W(z)$ is compared varies inversely with power above 50% RTP.

T(z) and W(z) functions are generated for discrete core elevations. Flux map data are typically taken for 87 core elevations. $F_Q^W(z)$ evaluations are not applicable for axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 10% inclusive,
b. Upper core region, from 90 to 100% inclusive,
c. Grid plane regions, $\pm$ 2% inclusive, and

d. Core plane regions, within ± 2% of the bank demand position of the control banks

These regions of the core are excluded from the evaluation because of the low probability that they would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions. The excluded regions at the top and bottom of the core are specified in the COLR and are defined to ensure that the minimum margin location is adequately surveilled. A slightly smaller exclusion zone may be specified, if necessary, to include the limiting margin location in the surveilled region of the core.

SR 4.2.2.3 requires a Surveillance of $F_Q^W(z)$ during the initial startup following each refueling within [24] hours after exceeding 75% RTP. THERMAL POWER levels below 75% are typically non-limiting with respect to the limit for $F_Q^W(z)$. Furthermore, startup physics testing and flux symmetry measurements, also performed at low power, provide confirmation that the core is operating as expected. This Frequency ensures that verification of $F_Q^W(z)$ is performed prior to extended operation at power levels where the maximum permitted peak LHR could be challenged and that required performance of SR 4.2.2.3 after a refueling is performed at a power level high enough to provide a high level of confidence in the accuracy of the Surveillance result.

Equilibrium conditions are achieved when the core is sufficiently stable at the intended operating conditions required to perform the Surveillance.

If a previous Surveillance of $F_Q^W(z)$ was performed at part power conditions, SR 4.2.2.3 also requires that $F_Q^W(z)$ be verified at power levels ≥10% RTP above the THERMAL POWER of its last verification within [24] hours after achieving equilibrium conditions. This ensures that $F_Q^W(z)$ is within its limit using radial peaking factors measured at the higher power level.

The allowance of up to 24 hours after achieving equilibrium conditions will provide a more accurate measurement of $F_Q^W(Z)$ by allowing sufficient time to achieve equilibrium conditions and obtain the power distribution measurement.

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

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