ENCLOSURE
Evaluation of Proposed Change

License Amendment Request: Technical Specification Revision to Adopt WCAP-14483-A, Generic Methodology for Expanded Core Operating Limits Report

ATTACHMENTS:
   1. Vogtle Electric Generating Plant 1&2 Marked-up TS Pages
   2. Vogtle Electric Generating Plant 1&2 Revised TS Pages
   3. Vogtle Electric Generating Plant 1&2 Marked-up TS Bases Pages
       (for information only)
   4. Vogtle Electric Generating Plant Unit 1 Markup of Cycle 24 Core Operating Limits
       Report (typical, for information only)

1. SUMMARY DESCRIPTION

The proposed amendment would revise the Technical Specifications for Vogtle Electric Generating Plant (VEGP) Units 1 and 2 renewed facility operating licenses NPF-68 and NPF-81, respectively.

The proposed change would revise Technical Specifications (TS) 2.1.1, "Reactor Core Safety Limits," (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," TS 3.4.1, "Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.5, "Core Operating Limits Report (COLR)," to adopt most of the TS and COLR changes described in Appendix A and Appendix B of Westinghouse topical report WCAP-14483-A, to relocate several cycle-specific parameter limits from the TS to the COLR. The proposed change is done following the guidance of technical specification task force change traveler TSTF-339-A, Revision 2. Along with the parameter relocations, this license amendment request (LAR) modifies the VEGP 1&2 TS 5.6.5, "Core Operating Limits Report (COLR)," to include WCAP-8745-P-A and WCAP-11397-P-A, and to revise the Specification applicability for WCAP-9272-P-A, in the list of the NRC approved methodologies used to develop the cycle specific COLR.

Approval of the proposed amendment is requested within one year from the date of this submittal. Due to the core design and safety analysis evaluation needed to support each core design using the methodology in the WCAP, implementation of this amendment for each unit will coincide with the start of the respective fuel cycles for each unit allowing for the necessary implementation prior to the beginning of the fuel cycle.

Three attachments are provided with this enclosure. Attachments 1 and 2 provide the marked-up TS pages and revised TS pages, respectively, for VEGP 1&2. Attachment 3 contains the associated marked-up TS Bases pages, showing the associated changes to be implemented based on Appendix A of WCAP-14483-A, for information only. Finally, Attachment 4 contains the associated marked-up Core Operating Limits Report, showing typical associated changes based on Appendix B of WCAP-14483-A, for information only.

Internally 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.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

General Design Criterion 10 of 10 CFR Part 50 Appendix A requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur on the limiting fuel rods and by requiring that fuel centerline temperature stays below the melting temperature. The restrictions of the Safety Limits prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient.

The Reactor Trip System (RTS) initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and RCS pressure boundary during AOOs and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The Overtemperature ΔT trip Function is provided to maintain DNB ratio within the design limit. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. Protection from violating the DNB ratio limit is assured for those transients that are slow with respect to delays from the core parameters to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase.

The Overpower ΔT trip Function provides protection for the integrity of the fuel under overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range
Neutron Flux — High Setpoint trip. The Overpower ΔT trip Function prevents exceeding the allowable heat generation rate (kW/ft) of the fuel.

The RCS is required to operate within certain parameter limits (for pressurizer pressure, RCS average temperature, and RCS total flow rate) that are assumed in the safety analyses to maintain the DNB within the design criterion in the event of a DNB limited transient.

2.2 Current Technical Specification Requirements

Technical Specification (TS) 2.1.1, "Reactor Core Safety Limits," provides limits for a combination of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure for operation in MODES 1 and 2.

TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," includes several cycle-specific input values for the Overtemperature Delta-T (OTΔT) Function 6 and the Overpower Delta-T (OPΔT) Function 7 in Notes 1 and 2 of Table 3.3.1-1, respectively, for operation in MODES 1 and 2.

Similarly, TS 3.4.1, "Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," also include several cycle-specific parameter limits for MODE 1 operation.

TS 5.6.5.b lists specific NRC approved documents describing analytical methods used to determine the core operating limits and includes parenthetical information identifying which Specifications are supported by the methodology. These currently include core operating limits associated with TS 3.1.1 - "SHUTDOWN MARGIN," TS 3.1.3 - "Moderator Temperature Coefficient," TS 3.1.5 - "Shutdown Bank Insertion Limits," TS 3.1.6 - "Control Bank Insertion Limits," TS 3.2.1 - "Heat Flux Hot Channel Factor," TS 3.2.2 - "Nuclear Enthalpy Rise Hot Channel Factor," TS 3.2.3 - "Axial Flux Difference," and TS 3.9.1 - "Boron Concentration," as listed in TS 5.6.5.a.

2.3 Reason for Proposed Change

NRC Generic Letter 88-16 allows licensees to remove cycle-dependent variables from TS provided that the values of these variables are included in a COLR and are determined with NRC-approved methodology which is referenced in the TS. These variables are moved from TS to the COLR to avoid the need for frequent revision of TS to change the value of those operating limits which cannot be specified to reasonably bound several operating cycles without significant loss of operating flexibility.

A number of Westinghouse-designed plants have been permitted to remove the cycle-specific values in certain limiting conditions for operation (LCOs) from plant TS and to place them in a COLR.

As noted in the NRC approved WCAP-14483-P-A, Generic Methodology for Expanded Core Operating Limits Report, this philosophy is extended to the cycle-dependent Reactor Core Safety Limits parameters, the OTΔT and OPΔT setpoint parameters and to the limits on the DNB parameters of pressurizer pressure, RCS flow, and the RCS T-avg.

This license amendment request proposes to move these cycle-dependent Reactor Core Safety Limits parameters, the OTΔT and OPΔT setpoint parameters and to the limits on the DNB parameters of pressurizer pressure, RCS flow, and the RCS T-avg from the TS to the COLR, and to identify the pertinent methodologies in the COLR TS 5.6.5.

2.4 Description of Proposed Change

The proposed change would modify the following for VEGP 1&2:

TS 2.1.1 Changes

- The Figure showing the limiting combinations of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure is moved to the COLR.

- Safety Limits are added for the DNB ratio and for peak fuel centerline temperature.

<u>TS 3.3.1 Changes</u>

- Table 3.3.1-1 Note 1 related to the OTΔT Function

  Several parameter values are moved from the TS to the COLR including:

τ_1, τ_2	time constants utilized in lead-lag compensator for differential temperature: $\tau_1 = 0$ seconds, $\tau_2 = 0$ seconds
τ_3	time constant utilized in lag compensator for differential temperature, ≤ 6 seconds
K_1	fundamental setpoint, $\leq 114.9\%$ RTP
K_2	modifier for temperature, $= 2.24\%$ RTP per degree F
τ_4, τ_5	time constants utilized in lead-lag compensator for temperature compensation: $\tau_4 \geq 28$ seconds, $\tau_5 \leq 4$ seconds
τ_6	time constant utilized in lag compensator for average temperature, ≤ 6 seconds
T'	indicated loop specific RCS average temperature at RTP, ≤ 588.4 degrees F
K_3	modifier for pressure, $= 0.177\%$ RTP per psig
P'	reference pressure, ≥ 2235 psig
$f_1(AFD)$	modifier for Axial Flux Difference (AFD): 1. for AFD between -23% and +10%, = 0% RTP 2. for each % AFD is below -23%, the trip setpoint shall be reduced by 3.3% RTP 3. for each % AFD is above +10%, the trip setpoint shall be reduced by 1.95% RTP
(p)	The compensated temperature difference [equation] shall be no more negative than 3 degrees F

  The values for each of these parameters is replaced with an asterisk [*] and a Note is included following the listing of parameters identifying that the asterisk denotes the value is specified in the COLR.

- Table 3.3.1-1 Note 2 related to the OPΔT Function

Several parameter values are moved from the TS to the COLR including:

τ_1, τ_2 time Constants utilized in lead-lag compensator for differential temperature: τ_1 = 0 seconds, τ_2 = 0 seconds

τ_3 time constant utilized in lag compensator for differential temperature, ≤ 6 seconds

K_4 fundamental setpoint, $\leq 110\%$ RTP

K_5 modifier for temperature change, $\geq 2\%$ RTP per degree F for increasing temperature, $\geq 0\%$ RTP per degree F for decreasing temperature

τ_7 time constant utilized in rate-lag compensator for temperature compensation, ≥ 10 seconds

τ_6 time constant utilized in lag compensator for average temperature, ≤ 6 seconds

K_6 modifier for temperature: $\geq 0.244\%$ RTP per degree F for T > T″, = 0% RTP for T ≤ T″

T″   indicated loop specific RCS average temperature at RTP, ≤ 588.4 degrees F

f_2(AFD)   modifier for Axial Flux Difference (AFD), = 0% RTP for all AFD

The values for each of these parameters is replaced with an asterisk [*] and a Note is included following the listing of parameters identifying that the asterisk denotes the value is specified in the COLR.

TS 3.4.1 Changes

- LCO 3.4.1

Several parameter values are moved from the TS Limiting Condition for Operation (LCO) and from the Surveillance Requirements (SRs) to the COLR including:

a.   Pressurizer pressure ≥ 2199 psig;

b.   RCS average temperature ≤ 592.5°F; and

c.   RCS total flow rate ≥ 384,509 gpm.

To accomplish this revision, the LCO is changed to read:

LCO 3.4.1   RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR.

- SR 3.4.1.1, SR 3.4.1.2, and SR 3.4.1.4

  To accomplish this revision, the SRs are changed to read:

  For SR 3.4.1.1, Verify pressurizer pressure is within the limits specified in the COLR.

  For SR 3.4.1.2, Verify RCS average temperature is within the limits specified in the COLR.

  And for SR 3.4.1.4, Verify by precision heat balance that RCS total flow rate is within the limits specified in the COLR.

TS 5.6.5 Changes

The listing of Specifications in TS 5.6.5.a is revised to include the following:

    LCO 2.1.1 "Reactor Core Safety Limits"
    LCO 3.3.1 "Reactor Trip System (RTS) Instrumentation"
    LCO 3.4.1 "Reactor Coolant System (RCS) Pressure, Temperature, and Flow
          Departure from Nucleate Boiling (DNB) Limits"

The WCAP-9272-P-A entry in TS 5.6.5.b is revised to identify that it also addresses the Methodology for the above Specifications by adding:

    ", Reactor Trip System Instrumentation, and Reactor Coolant System Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits"

The WCAPs listed in TS 5.6.5.b is revised to include:

    WCAP-8745-P-A, "Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (W Proprietary) . (Methodology for Reactor Trip System Instrumentation.)

    WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 (W Proprietary). (Methodology for Reactor Core Safety Limits and RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

TS Bases Changes (for information only)

Corresponding TS Bases changes are identified for TS 2.1.1, TS 3.3.1, and TS 3.4.1 that will be implemented in accordance with the TS Bases Change Program coincidently with the above TS changes. Thus, the Bases changes are provided for information only.

COLR Changes (for information only)

Corresponding COLR changes are identified for TS 2.1.1, TS 3.3.1, and TS 3.4.1 that will be implemented coincidently with the above TS changes. The COLR changes are cycle

specific and thus cannot be developed until the cycle for implementation is known. Thus, these COLR markups represent typical expected changes and are provided for information only.

3. TECHNICAL EVALUATION

3.1 Basis for the Changes

Southern Nuclear Company has reviewed TSTF-339, Rev. 2, and has concluded that the information in WCAP-14483-A-P and TSTF-339 is applicable to VEGP – Units 1 and 2 and justifies the proposed amendment for the incorporation of the changes to the VEGP – Units 1 and 2 Technical Specifications, as discussed below.

3.2 TSTF Change Traveler Adoption and Variations

TSTF-339-A Revision 2 was approved by the NRC on June 13, 2000, and incorporated in NUREG-1431, Rev. 2, in June 2001. NUREG-1431, SL 2.1.1, LCO 3.3.1, Table 3.3.1-1, Notes 1 and 2, and LCO 3.4.1 and TS 3.4.1 Surveillances were revised to move specific parameter values from the TS to the COLR.

The following VEGP specific variations include:

*   TS 2.1.1 is proposed to be revised as identified in the TSTF with the exception that the 95/95 DNB criterion correlations are proposed to be determined per the methodologies identified in the COLR Specification 5.6.5 rather than including the specific value in the Safety Limits Specification. The correlations change with fuel type and the variation is consistent with previous approvals for the Farley, Comanche Peak and North Anna operating plants.

*   TS 3.3.1 Overtemperature Delta-T and Overpower Delta-T functions are defined differently in the TSTF than in the previously approved TS 3.3.1 Note 1 and Note 2 for Table 3.3.1-1. However, both include terms with values that may vary by fuel cycle. Thus, the cycle-dependent terms are proposed to be moved to the COLR consistent with the TSTF.

*   TS 3.4.1 is reformatted to simply identify that the specified parameters shall be with the limits specified in the COLR. Repeating the specified parameters in the LCO and identifying the values must be greater than or less than the limits (in the LCO and in the Surveillance Requirements) is unnecessary detail. This variation is consistent with the approved change to the Farley Units 1 and 2 TS 3.4.1.

*   TS 5.6.5 is revised. Specific revisions to TS 5.6.5 were not identified in the TSTF and only partially identified in WCAP-14483-P-A. However, with the affected TS 2.1.1, TS 3.3.1, and TS 3.4.1 would need to be listed in TS 5.6.5.a as requiring core operating limits to be established and documented in the COLR. In addition, the WCAP-14483 (Section 2.2 and Section 5.0) identifies that: This approach is supported by the existing NRC-approved reload methodologies, such as the Westinghouse reload methodology described in Reference 5 [WCAP-9272], which examines each of the DNB Parameter TS limits of RCS T-avg, RCS total flow rate and pressurizer pressure as well as the

OTΔT and OPΔT setpoints and the supporting bases for the setpoints, on a cycle-specific basis. Thus, Specifications 3.3.1 and 3.4.1 would also need to be identified under the listing of WCAP-9272-P-A in TS 5.6.5.b. In addition, the WCAP does identify that WCAP-8745-P-A should also be added to the COLR listing of analytical methodologies used to determine the core operating limits in support of the OTΔT and the OPΔT Function input values to be included in the COLR. Finally, the Revised Thermal Design Procedure addressed in WCAP-11397-P-A is added to support the DNB methodology as identified in FSAR Subsection 4.4.1.1.2 for both TS 2.1.1 and TS 3.4.1.

These plant-specific variations have no adverse impact that precludes adoption of this change traveler TSTF-339-A Revision 2 for TS 2.1.1, TS 3.3.1, or TS 3.4.1.

3.3   Applicability of WCAP-14483-P-A, Safety Evaluation and Variations

TSTF-339-A Revision 2 is based on WCAP-14483-P-A. The Safety Evaluation (SE) provided by the NRC is applicable to the changes proposed in the license amendment request. The only variations from the WCAP are those proposed and approved in TSTF-339-A Revision 2 and the variations from the TSTF identified in the previous section. The variations are consistent with the guidance of Generic Letter 88-16 and do not invalidate the SE discussions. There are no conditions or limitations identified in the SE.

4   REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

SNC has reviewed TSTF-339-A and the NRC SE provided in the NRC letter to Westinghouse Electric Company dated January 19, 1999 (included in the WCAP-14483-P-A) that supported approval of WCAP-14483-P. The NRC concluded in the SE that the modified TS contained in WCAP-14483 are acceptable, and subsequently approved minor modifications to the WCAP changes via the TSTF revisions. Thus, the proposed change is consistent with the regulatory requirements identified in the SE and the subsequent supporting TSTF.

4.2 Precedent

Other plants with approved Specifications with similar cycle-dependent parameters relocated to a licensee-controlled COLR include:

- H.B. Robinson Steam Electric Plant [DPR-23; SER ML17039A153]

- Shearon Harris Nuclear Power Plant [NPF-63; SER ML17250A202]

- Comanche Peak Nuclear Power Plant [NPF-87]

- North Anna Power Station [NPF-4]

- Farley Nuclear Plant [NPF-2]

In addition, approval of the license amendment request will make the Vogtle Units 1 and 2 Technical Specifications more consistent with the corresponding Vogtle Units 3 and 4 Technical Specifications and Farely Units 1 and 2 Technical Specifications which include cycle dependent parameters in licensee-controlled documents. Similar fleet requirements tend to improve the consistency of human performance.

4.3 Significant Hazards Consideration

Pursuant to the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Southern Nuclear Operating Company (SNC) hereby requests an amendment to Vogtle Electric Generating Plant (VEGP) Units 1 and 2 renewed facility operating licenses NPF-68 and NPF-81, respectively.

The proposed change would revise Technical Specifications (TS) 2.1.1, "Reactor Core Safety Limits," (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," and TS 3.4.1, "Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," to move several cycle-specific parameter limits from the TS to the Core Operating Limits Report (COLR).

Additionally, the proposed change modifies TS 5.6.5, "Core Operating Limits Report (COLR)," to include changes reflecting the above application of WCAP-14483-P-A by appropriately modifying the list of the NRC approved methodologies used to develop the cycle specific COLR.

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

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

   Response: No

   The proposed change moves some parameter limit values from the Technical Specifications (TS) to a cycle-specific Core Operating Limits Report (COLR) but does not remove the required designation of the limits. Operation within the limits continues to maintain 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 change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change 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). Operation in accordance with the revised TS and the required parameter 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 does not involve the alteration of plant equipment or introduce unique operational modes or accident precursors.

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

3. Does the proposed change 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 the containment. The performance of these fission product barriers is not adversely affected by the proposed change.

   The proposed change moves some parameter limit values from the Technical Specifications (TS) to a cycle-specific Core Operating Limits Report (COLR) but does not remove the required designation of the limits. The limits are not changed and the methodology to determine the limits is not changed. 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.

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

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

4.4 Conclusions

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

5   ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released off site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6   REFERENCES

1.   Westinghouse Topical Report WCAP-14483-P-A, "Generic Methodology for Expanded Core Operating Limits Report," Approved January 1999 [ML020430092]

2.   Technical Specification Task Force (TSTF) Traveler 339, Revision 2, Approved June 2000 [ML003723269]

Attachment 1

Vogtle Electric Generating Plant 1&2 Marked-up TS Pages

(13 total pages including cover page)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the ~~SLs~~ specified in ~~Figure 2.1.1-1.~~

[limits]

[INSERT 2.1.1-1]

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

INSERT 2.1.1-1

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

2.1.1.1   The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB criterion correlations and methodologies specified in Specification 5.6.5. ≥ [1.17 for the WRB-I/WRB-2 DNB correlations].

2.1.1.2   The peak fuel centerline temperature shall be maintained < [5080'F, decreasing by 58°F per 10,000 MWD/MTU of burnup].

Markup shows variation from TSTF-339-A Rev. 2
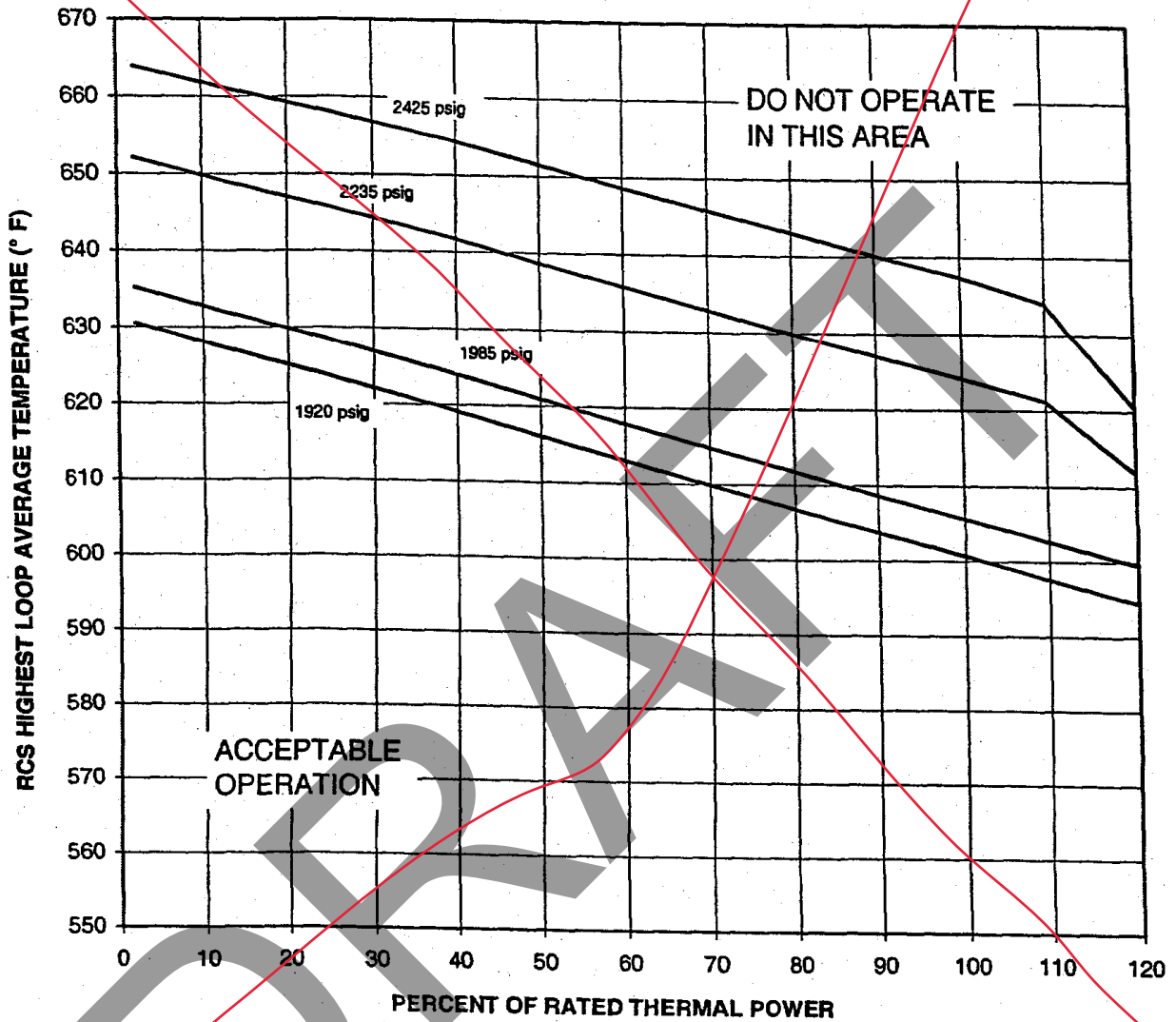
Remove this figure.

Figure 2.1.1-1
Reactor Core Safety Limits

Table 3.3.1-1 (page 7 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature Delta-T

The Allowable Value of each input to the Overtemperature Delta-T function as defined by the equation below shall not exceed its as-left value by more than the following:

(1)   0.5% ΔT span for the ΔT channel
(2)   0.5% ΔT span for the T_{avg} channel
(3)   0.5% ΔT span for the pressurizer pressure channel
(4)   0.5% ΔT span for the $f_1(AFD)$ channel

$$\left[100\frac{\Delta T}{\Delta T_0}\frac{\{1+\tau_1 s\}}{\{1+\tau_2 s\}}\frac{1}{\{1+\tau_3 s\}}\right] \le \left[K_1 - K_2\frac{\{1+\tau_4 s\}}{\{1+\tau_5 s\}}\left[T\frac{1}{\{1+\tau_6 s\}}-T'\right]^{(p)} - K_3\{P'-P\} - f_1(AFD)\right]$$

Where: ΔT measured loop specific RCS differential temperature, degrees F

ΔT_0 indicated loop specific RCS differential at RTP, degrees F

$\dfrac{1+\tau_1 s}{1+\tau_2 s}$ lead-lag compensator on measured differential temperature

τ_1, τ_2 time constants utilized in lead-lag compensator for differential temperature: $\tau_1 = 0$ seconds, $\tau_2 = 0$ seconds   *   *

$\dfrac{1}{1+\tau_3 s}$ lag compensator on measured differential temperature

τ_3 time constant utilized in lag compensator for differential temperature, ≤ 6 seconds   *   *

K_1 fundamental setpoint, ≤ 114.9% RTP

K_2 modifier for temperature, $= 2.24$% RTP per degree F   *

$\dfrac{1+\tau_4 s}{1+\tau_5 s}$ lead-lag compensator on dynamic temperature compensation

τ_4, τ_5 time constants utilized in lead-lag compensator for temperature compensation: $\tau_4 \ge 28$ seconds, $\tau_5 \le 4$ seconds   *   *

T measured loop specific RCS average temperature, degrees F

$\dfrac{1}{1+\tau_6 s}$ lag compensator on measured average temperature

τ_6 time constant utilized in lag compensator for average temperature, ≤ 6 seconds   *

T' indicated loop specific RCS average temperature at RTP, ≤ 588.4 degrees F   *

K_3 modifier for pressure, $= 0.177$% RTP per psig   *

P measured RCS pressurizer pressure, psig

P' reference pressure, ≥ 2235 psig   *

s Laplace transform variable, inverse seconds

Table 3.3.1-1 (page 8 of 9)
Reactor Trip System Instrumentation

Note 1: Overtemperature Delta-T (continued)

f_1(AFD)   modifier for Axial Flux Difference (AFD):

1.   for AFD between -23% and +10%, = 0% RTP [*][*][*]

2.   for each % AFD is below -23%, the trip setpoint shall be reduced by 3.3% RTP [*][*]

3.   for each % AFD is above +10%, the trip setpoint shall be reduced by 1.95% RTP [*][*]

(p) The compensated temperature difference $\dfrac{\{1 + \tau_4 s\}}{\{1 + \tau_5 s\}}\left[T\dfrac{1}{\{1 + \tau_6 s\}} - T' \right]$ shall be no more negative than 3 degrees F. [*]

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

Note 2: Overpower Delta-T

The Allowable Value of each input to the Overpower Delta-T function as defined by the equation below shall not exceed its as-left value by more than the following:

(1)   0.5% ΔT span for the ΔT channel
(2)   0.5% ΔT span for the T_{avg} channel

$$\left[100\,\frac{\Delta T}{\Delta T_0}\,\frac{\{1 + \tau_1 s\}}{\{1 + \tau_2 s\}}\,\frac{1}{\{1 + \tau_3 s\}} \right] \le \left[K_4 - \left[K_5\,\frac{\{\tau_7 s\}}{\{1 + \tau_7 s\}}\,\frac{1}{\{1 + \tau_6 s\}}\,T \right] - K_6\left[T\,\frac{1}{\{1 + \tau_6\}} - T'' \right] - f_2(AFD) \right]$$

Where: ΔT measured loop specific RCS differential temperature, degrees F

ΔT_0 indicated loop specific RCS differential at RTP, degrees F

$\dfrac{1 + \tau_1 s}{1 + \tau_2 s}$ lead-lag compensator on measured differential temperature

τ_1, τ_2 time constants utilized in lead-lag compensator for differential temperature: $\tau_1 = 0$ seconds, [*]
$\tau_2 = 0$ seconds [*]

$\dfrac{1}{1 + \tau_3 s}$ lag compensator on measured differential temperature

τ_3 time constant utilized in lag compensator for differential temperature, ≤ 6 seconds [*]

K_4 fundamental setpoint, ≤ 110% RTP [*]

K_5 modifier for temperature change: ≥ 2% RTP per degree F for increasing temperature, ≥ 0% RTP per degree F for decreasing temperature [*]

$\dfrac{\tau_7 s}{1 + \tau_7 s}$ rate-lag compensator on dynamic temperature compensation

τ_7 time constant utilized in rate-lag compensator for temperature compensation, ≥ 10 seconds [*]

T measured loop specific RCS average temperature, degrees F

$\dfrac{1}{1 + \tau_6 s}$ lag compensator on measured average temperature

Table 3.3.1-1 (page 9 of 9)
Reactor Trip System Instrumentation

Note 2: Overpower Delta-T (continued)

τ_6 time constant utilized in lag compensator for average temperature, ≤ 6 seconds   [*]

K_6 modifier for temperature: $\geq$ 0.244% RTP per degree F for T > T″, = 0% RTP for $T \leq T''$ [*]   [*]

T″   indicated loop specific RCS average temperature at RTP, $\leq$ 588.4 degrees F   [*]

s   Laplace transform variable, inverse seconds

$f_2(AFD)$ modifier for Axial Flux Difference (AFD), = 0% RTP for all AFD   [*]

The values denoted with [*] are specified in the COLR.
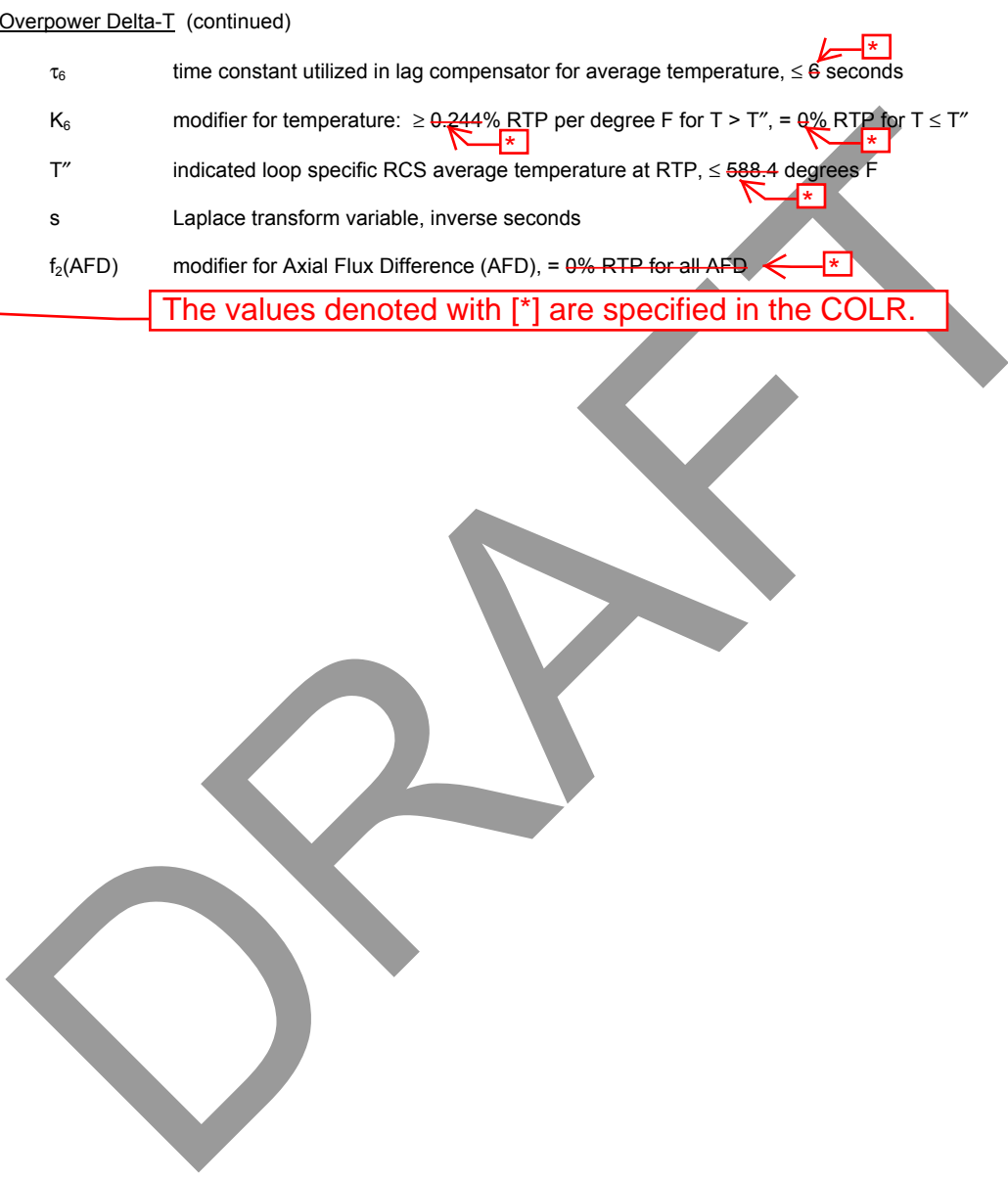
3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1   RCS DNB parameters for pressurizer pressure, RCS average temperature,
                    and RCS total flow rate shall be within the limits specified ~~below:~~ in the COLR.

                    a.   ~~Pressurizer pressure ≥ 2199 psig;~~

                    b.   ~~RCS average temperature ≤ 592.5°F; and~~

                    c.   ~~RCS total flow rate ≥ 384,509 gpm.~~

APPLICABILITY: MODE 1.

                  -------------------------------------NOTE---
                  Pressurizer pressure limit does not apply during:

                  a.   THERMAL POWER ramp > 5% RTP per minute; or

                  b.   THERMAL POWER step > 10% RTP.

ACTIONS

	CONDITION		REQUIRED ACTION	COMPLETION TIME
A.	One or more RCS DNB parameters not within limits.	A.1	Restore RCS DNB parameter(s) to within limit.	2 hours
B.	RCS total flow rate degraded.	B.1.	Perform SR 3.4.1.4.	7 days
C.	Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 Verify pressurizer pressure is ~~≥ 2199 psig.~~ within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2 Verify RCS average temperature is ~~≤ 592.5°F.~~ within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3 Monitor RCS total flow rate for degradation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.4 ------------------------NOTE------------------------- Not required to be performed until 7 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ~~≥ 384,509 gpm.~~ within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

5.6 Reporting Requirements (continued)

5.6.5   Core Operating Limits Report (COLR)

a.   Core operating limits shall be established prior to each reload cycle, or
        prior to any remaining portion of a reload cycle, and shall be
        documented in the COLR for the following:

        LCO 3.1.1 "SHUTDOWN MARGIN"   ← Insert 5.6.5-1
        LCO 3.1.3 "Moderator Temperature Coefficient"
        LCO 3.1.5 "Shutdown Bank Insertion Limits"
        LCO 3.1.6 "Control Bank Insertion Limits"
        LCO 3.2.1 "Heat Flux Hot Channel Factor"
        LCO 3.2.2 "Nuclear Enthalpy Rise Hot Channel Factor"
        LCO 3.2.3 "Axial Flux Difference"   ← Insert 5.6.5-2
        LCO 3.9.1 "Boron Concentration"

b.   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:

        WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION
        METHODOLOGY," July 1985 ($\underline{W}$ Proprietary). (Methodology for
        Moderator Temperature Coefficient, Shutdown Bank Insertion Limit,
        Control Bank Insertion Limits, and Nuclear Enthalpy Rise Hot Channel
        Factor.)   ← Insert 5.6.5-3

        WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL
        OFFSET CONTROL FQ SURVEILLANCE TECHNICAL
        SPECIFICATION," February, 1994 ($\underline{W}$ Proprietary). (Methodology for
        Axial Flux Difference (Relaxed Axial Offset Control) and Heat Flux Hot
        Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

        WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse
        ECCS Evaluation Model Using the BASH Code," March 1987.
        (W Proprietary) (Methodology for Axial Flux Difference (Relaxed Axial
        Offset Control) and Heat Flux Hot Channel Factor (W(Z) surveillance
        requirements for F_Q Methodology).)

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

        WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport
        Code PARAGON," August 2004 (Methodology for Moderator
        Temperature Coefficient.)

        WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear
        Data Methodology," August 2007 (Methodology for Moderator
        Temperature Coefficient.)

                                                                    (continued)

INSERT 5.6.5-1

LCO 2.1.1 "Reactor Core Safety Limits"

INSERT 5.6.5-2

LCO 3.3.1 "Reactor Trip System Instrumentation (RTS)"

LCO 3.4.1 "Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

INSERT 5.6.5-3

, Reactor Trip System Instrumentation, and Reactor Coolant System Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits.)

5.6 Reporting Requirements (continued)

5.6.5   Core Operating Limits Report (COLR) (continued)

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary). (Methodology for Axial Flux Difference (Relaxed Axial Offset Control) and Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO^{TM}," July 2006 (Westinghouse Proprietary). (Methodology for Axial Flux Difference (Relaxed Axial Offset Control) and Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

Insert 5.6.5-4

c.   The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d.   The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6   Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a.   RCS pressure and temperature limits for heatup, cooldown, operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3 "RCS Pressure and Temperature (P/T) Limits"

b.   The power operated relief valve lift settings required to support the Cold Overpressure Protection Systems (COPS) and the COPS arming temperature shall be established and documented in the PTLR for the following:

LCO 3.4.12 "Cold Overpressure Protection Systems"

c.   The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1.   WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

(continued)

INSERT 5.6.5-4

WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 ($\underline{W}$ Proprietary). (Methodology for Reactor Trip System Instrumentation.)

WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 ($\underline{W}$ Proprietary). (Methodology for Reactor Core Safety Limits and RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

Attachment 2

Vogtle Electric Generating Plant 1&2 Revised TS Pages

(## total pages including cover page)

DRAFT

Attachment 3

**Vogtle Electric Generating Plant 1&2 Marked-up TS Bases Pages
(Information only)**

(16 total pages including cover page)

DRAFT

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur on the limiting fuel rods and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

BACKGROUND
   (continued)
The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE
SAFETY ANALYSES
The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. The hot fuel pellet in the core must not experience centerline fuel melting; and

b. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit DNBR values that satisfy the DNB design criterion. The Vantage 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical and thimble cells, respectively. The Lopar fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNB limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio

RCS flow, Axial Flux Difference

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	(DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the ~~following functions:~~ appropriate operation of the RPS and steam generator safety valves.

a. ~~High pressurizer pressure trip;~~

b. ~~Low pressurizer pressure trip;~~

c. ~~Overtemperature ΔT trip;~~

d. ~~Overpower ΔT trip;~~

e. ~~Power Range Neutron Flux trip;~~

f. ~~Reactor Coolant Flow trips (including undervoltage and underfrequency of the reactor coolant pump buses); and~~

g. ~~Main steam safety valves.~~

~~The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.~~

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS	~~The curves provided in Figure B 2.1.1-1 show the loci of points of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A), RCS Pressure (PI-0455A, B, and C, PI-0456, PI-0456A, PI-0457, PI-0457A, PI-0458, and PI-0458A), and average temperature (TI-0412, TI-0422, TI-0432, TI-0442) for which the minimum DNBR is not less than the safety analyses limit, that fuel~~

Bases Insert 2.0-1

(continued)

BASES INSERT 2.0-1

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% 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 a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various 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 AT 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 (NI-0041, NI-0042, NI-0043, NI-0044, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A), RCS Pressure (PI-0455A, B, and C, PI-0456, PI-0456A, PI-0457, PI-0457A, PI-0458, and PI-0458A), RCS average temperature (TI-0412, TI-0422, TI-0432, TI-0442), RCS flow rate (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446), and AXIAL FLUX DIFFERENCE (AFD) ~~AI~~ that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

BASES

| SAFETY LIMITS (continued) | ~~centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.~~ |

~~The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

| APPLICABILITY | SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER. |

| SAFETY LIMIT VIOLATIONS | Section 2.2, SL Violations, provides the Required Actions to be taken in response to a violation of Safety Limits. The bases for the Required Actions of Section 2.2 applicable to a violation of the reactor core SLs are discussed below. |

(continued)

BASES

SAFETY LIMIT
VIOLATIONS
(continued)

2.2.1

If the reactor core SL 2.1.1 is violated, the requirement to go to
MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of
bringing the unit to a MODE of operation where this SL is not
applicable, and reduces the probability of fuel damage.

(continued)

BASES (continued)

REFERENCES   1.   10 CFR 50, Appendix A, GDC 10.

2.   FSAR, Section 7.2.

3.   ~~WCAP-8746-A, March 1977.~~

4.   ~~WCAP-9272-P-A, July 1985.~~

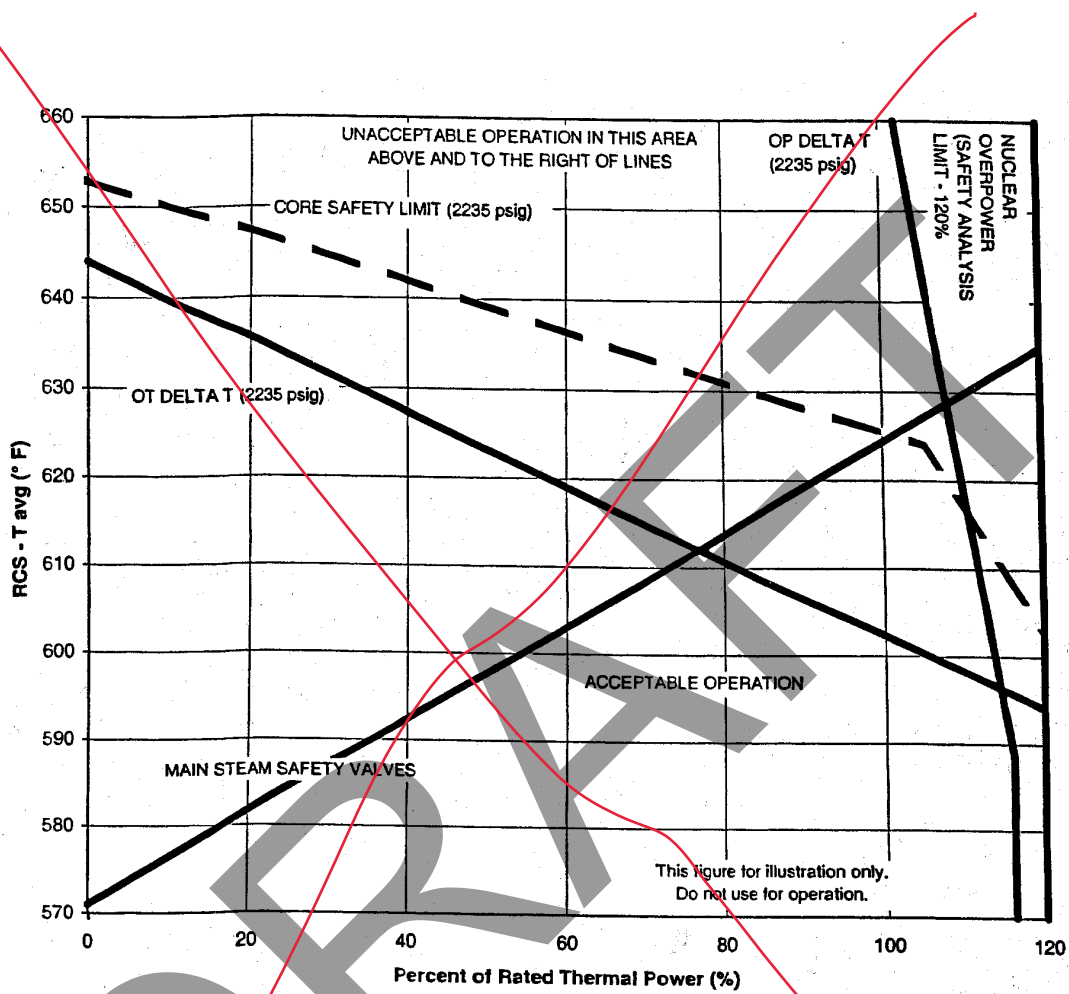Remove this figure.

Figure B 2.1.1-1 (page 1 of 1)
REACTOR CORE SAFETY LIMITS VS. BOUNDARY OF PROTECTION

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling
(DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the DNB design criterion will be met for each of the transients analyzed.

The design method employed to meet the DNB design criterion for the VANTAGE 5 and LOPAR fuel assemblies is the Revised Thermal Design Procedure (RTDP). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined in order to meet the DNB design criterion.

The RTDP design limit DNBR values are 1.24 and 1.23 for the typical and thimble cells, respectively, for the VANTAGE 5 fuel analyses with the WRB-2 correlation. For the LOPAR fuel analyses, the RTDP design limit DNBR values are 1.23 and 1.22 for the typical and thimble cells, respectively.

Additional DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the safety analyses input values to give the lowest minimum DNBR. The design

(continued)

BASES

BACKGROUND (continued)	limit DNBR for STDP is the 95/95 limit for the appropriate DNB correlation. Again, additional DNBR margin is maintained in the safety analyses to offset the applicable DNBR penalties.

For both the WRB-1 and the WRB-2 correlations, the 95/95 DNBR
correlation limit is 1.17. The W-3 DNB correlation is used for both fuel
types where the primary DNBR correlations are not applicable. The
WRB-1 and WRB-2 correlations were developed based on mixing
vane data and therefore are only applicable in the heated rod spans
above the first mixing vane grid. The W-3 correlation, which does not
take credit for mixing vane grids, is used to calculate DNBR values in
the heated region below the first mixing vane grid. In addition, the
W-3 correlation is applied in the analysis of accident conditions where
the system pressure is below the range of the primary correlations.
For system pressures in the range of 500 to 1000 psia, the W-3
correlation limit is 1.45. For system pressures greater than 1000 psia,
the W-3 correlation limit is 1.30.

The RCS pressure and average temperature limits are consistent with
operation within the nominal operational envelope. A lower pressure
and/or higher average temperature will cause the reactor core to
approach DNB limits.

The RCS flow rate normally remains constant during an operational
fuel cycle with all pumps running. The minimum RCS flow limit
corresponds to that assumed for DNB analyses. A lower RCS flow
will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits
increases the likelihood of a fuel cladding failure in a DNB limited
event.

APPLICABLE SAFETY ANALYSES	The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNB design criterion. Changes to the unit that could impact these parameters must be assessed for their impact on the DNB design criterion. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	transients analyzed include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit ~~of 2199 psig~~ and the RCS average temperature limit ~~of 592.5°F~~ correspond to ~~the~~ analytical limits ~~of 2185 psig and 594.4°F~~ used in the safety analyses, with allowance for measurement uncertainty.

[specified in the COLR]

The indicated RCS flow value of 384,509 gpm corresponds to an analytical value of 374,400 gpm with allowance for measurement and indication uncertainties.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables — pressurizer pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, and PI-0458 & PI-0458A), RCS average temperature (TI-0412, TI-0422, TI-0432, and TI-0442), and RCS total flow rate (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) — to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

[These variables are contained int he COLR to provide operating and analysis flexibility from cycle to cycle.]

RCS total flow rate contains a measurement error ~~of 2.7%~~ based on performing a precision heat balance above 90% RTP and using the result to calibrate the RCS flow rate indicators. This measurement uncertainty includes a ~~0.1%~~ penalty to account for potential fouling of the feedwater venturi, which might not be detected and could bias the result from the precision heat balance in a nonconservative manner.

Any fouling that might bias the flow rate measurement greater than ~~0.1%~~ can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and

[the penalty for undetected fouling of the feedwater venturi]

(continued)

BASES

LCO (continued)	compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNB design criterion will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit

~~Another set of limits on DNB related parameters~~ is provided in SL 2.1.1, "Reactor Core SLs." ~~Those limits~~ are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action.

The conditions which define the DNBR limit

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS flow rate exhibits degradation and the actual total RCS flow rate is below the LCO limit as determined by precision heat balance, power must be reduced, as required by Required Action C.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

(continued)

BASES

ACTIONS
A.1 (continued)

The 2 hour Completion Time for restoration of the parameters is based on plant operating experience and provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits.

B.1

If degradation in RCS total flow rate is detected via the flow rate indicators, a precision calorimetric heat balance must be performed within 7 days of detection of the degradation. The precision heat balance will positively verify actual RCS total flow rate. The 7-day Completion Time is adequate to allow for the setup necessary for this measurement and is acceptable since the RCS low flow reactor trips will protect the reactor against actual low flow conditions.

C.1

If Required Actions A.1 or B.1 are not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE
REQUIREMENTS
SR 3.4.1.1

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

This surveillance demonstrates that the pressurizer pressure remains greater than of equal to the limit specified in the COLR.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.1.2

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

This surveillance demonstrates that the average
RCS termperature remains less than or equal to the
limit specified in the COLR.

SR 3.4.1.3

The RCS flow instrumentation indicates from 0% to 120% as opposed
to actual flow in gallons per minute. Therefore, the flow
instrumentation is used to detect degradation in flow rather than as a
comparison against the actual limit in gallons per minute.
Degradation is defined as a change in indicated percent flow which is
greater than the instrument channel inaccuracies and parallax errors.
The Surveillance Frequency is controlled under the Surveillance
Frequency Control Program.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision
calorimetric heat balance allows the installed RCS flow
instrumentation to be correlated with the precision flow measurement
and verifies the actual RCS flow rate is greater than or equal to the
minimum required RCS flow rate. In addition, in order to ensure that
the measurement uncertainty assumed in the limit for RCS total flow
rate is maintained, the instrumentation used for the precision
calorimetric heat balance will be calibrated within 30 days prior to the
precision calorimetric.

specified in the COLR.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.4 (continued)

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

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 7 days after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 7 days after reaching 90% RTP.

REFERENCES

1.   FSAR, Chapter 15.

Attachment 4

**Vogtle Electric Generating Plant Units 1 and 2 Markups of
Core Operating Limits Report (COLR) Pages
(typical, information only)**

(17 total pages including cover page)

DRAFT

1.0   CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for VEGP Unit 1 Cycle 24 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The Technical Requirement affected by this report is listed below:

    13.1.1   SHUTDOWN MARGIN - MODES 1 and 2

The Technical Specifications affected by this report are listed below:

3.1.1	SHUTDOWN MARGIN - MODES 3, 4 and 5
3.1.3	Moderator Temperature Coefficient
3.1.5	Shutdown Bank Insertion Limits
3.1.6	Control Bank Insertion Limits
3.2.1	Heat Flux Hot Channel Factor - $F_Q(Z)$
3.2.2	Nuclear Enthalpy Rise Hot Channnel Factor - $F^N_{\Delta H}$
3.2.3	Axial Flux Difference
3.9.1	Boron Concentration

Insert 1

Insert 2

COLR Section 1.0 INSERT 1

2.1.1   Reactor Core Safety Limits for THERMAL POWER

COLR Section 1.0 INSERT 2

3.3.1   Reactor Trip System Instrumentation

3.4.1   Reactor Coolant System Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits

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

   2.7.1 $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} \cdot \left(1 + PF_{\Delta H} \cdot \left(1 - P \right) \right)$

      where: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

   2.7.2 $F^{RTP}_{\Delta H} = 1.65$

   2.7.3 $PF_{\Delta H} = 0.3$

2.8   Axial Flux Difference (Specification 3.2.3)

   2.8.1   The Axial Flux Difference (AFD) Acceptable Operation Limits are provided in Figure 5.

2.9   Boron Concentration (Specification 3.9.1)

   2.9.1   The boron concentration shall be greater than or equal to 2000 ppm.[2]

Insert 3

---

[2]   This concentration bounds the condition of $k_{eff} \leq 0.95$ (all rods in less the most reactive rod) and subcriticality (all rods out) over the entire cycle. This concentration includes additional boron to address uncertainties and B^{10} depletion.

INSERT 3 for new COLR Sections 2.10, 2.11 and 2.12

2.10   Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)

      2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the safety limits specified in Figure 7.

2.11   Reactor Trip System Instrumentation (Specification 3.3.1)

      2.11.1 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for TS Table 3.3.1-1 are listed in COLR Tables 3 and 4.

2.12   Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits (Specification 3.4.1)

      2.12.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

            a. Pressurizer pressure ≥ 2199 psig;

            b. RCS average temperature ≤ 592.5°F; and

            c. The minimum RCS total flow rate shall be ≥ 384,509 GPM when using the precision heat balance method.

Table 2
RAOC W(Z)

	Axial Point	Elevation (feet)	150 MWD/MTU	4000 MWD/MTU	8000 MWD/MTU	12000 MWD/MTU	16000 MWD/MTU	20000 MWD/MTU
*	1-5	12.072 – 11.267	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
	6	11.066	1.3733	1.4042	1.2807	1.2524	1.2279	1.2288
	7	10.865	1.3592	1.3892	1.2714	1.2511	1.2252	1.2237
	8	10.664	1.3419	1.3696	1.2569	1.2461	1.2119	1.2118
	9	10.462	1.3243	1.3477	1.2420	1.2392	1.1946	1.2012
	10	10.261	1.3032	1.3235	1.2331	1.2382	1.1967	1.1895
	11	10.060	1.2799	1.2969	1.2245	1.2395	1.2029	1.1869
	12	9.859	1.2544	1.2649	1.2132	1.2404	1.2071	1.1850
	13	9.658	1.2448	1.2309	1.2046	1.2287	1.2063	1.1876
	14	9.456	1.2433	1.2035	1.1971	1.2238	1.2078	1.1907
	15	9.255	1.2406	1.1870	1.1866	1.2224	1.2059	1.1962
	16	9.054	1.2314	1.1714	1.1819	1.2178	1.2049	1.2024
	17	8.853	1.2250	1.1656	1.1769	1.2031	1.2024	1.2048
	18	8.652	1.2208	1.1660	1.1767	1.1947	1.1963	1.2063
	19	8.450	1.2213	1.1704	1.1829	1.1974	1.1982	1.2120
	20	8.249	1.2174	1.1728	1.1860	1.2010	1.2074	1.2250
	21	8.048	1.2228	1.1734	1.1872	1.2093	1.2212	1.2399
	22	7.847	1.2252	1.1725	1.1868	1.2150	1.2324	1.2523
	23	7.646	1.2247	1.1694	1.1842	1.2185	1.2412	1.2623
	24	7.444	1.2221	1.1653	1.1805	1.2204	1.2483	1.2710
	25	7.243	1.2164	1.1589	1.1740	1.2184	1.2510	1.2747
	26	7.042	1.2087	1.1509	1.1659	1.2138	1.2505	1.2750
	27	6.841	1.1995	1.1434	1.1568	1.2080	1.2483	1.2738
	28	6.640	1.1883	1.1364	1.1493	1.2003	1.2439	1.2705
	29	6.438	1.1766	1.1299	1.1417	1.1910	1.2372	1.2676
	30	6.237	1.1653	1.1232	1.1343	1.1814	1.2294	1.2627
	31	6.036	1.1534	1.1207	1.1260	1.1709	1.2204	1.2567
	32	5.835	1.1452	1.1179	1.1215	1.1597	1.2100	1.2491
	33	5.634	1.1390	1.1197	1.1196	1.1583	1.2034	1.2398
	34	5.432	1.1386	1.1326	1.1259	1.1642	1.2035	1.2355
	35	5.231	1.1448	1.1440	1.1366	1.1686	1.2082	1.2367
	36	5.030	1.1521	1.1550	1.1463	1.1727	1.2118	1.2391
	37	4.829	1.1589	1.1653	1.1557	1.1777	1.2138	1.2395
	38	4.628	1.1651	1.1751	1.1643	1.1813	1.2141	1.2379
	39	4.426	1.1706	1.1842	1.1724	1.1839	1.2128	1.2342
	40	4.225	1.1750	1.1922	1.1794	1.1851	1.2100	1.2289
	41	4.024	1.1782	1.1994	1.1858	1.1863	1.2060	1.2220
	42	3.823	1.1823	1.2056	1.1910	1.1870	1.1995	1.2120
	43	3.622	1.1888	1.2108	1.1949	1.1857	1.1912	1.1995
	44	3.420	1.1950	1.2151	1.1982	1.1841	1.1822	1.1864
	45	3.219	1.1992	1.2187	1.2002	1.1816	1.1741	1.1738
	46	3.018	1.2056	1.2318	1.2057	1.1867	1.1750	1.1743
	47	2.817	1.2141	1.2489	1.2211	1.1969	1.1846	1.1842
	48	2.616	1.2276	1.2646	1.2408	1.2060	1.1952	1.1955
	49	2.414	1.2436	1.2847	1.2606	1.2150	1.2057	1.2074
	50	2.213	1.2608	1.3090	1.2806	1.2266	1.2162	1.2192
	51	2.012	1.2778	1.3332	1.2999	1.2377	1.2253	1.2293
	52	1.811	1.2946	1.3570	1.3186	1.2484	1.2340	1.2388
	53	1.610	1.3103	1.3795	1.3366	1.2589	1.2431	1.2490
	54	1.408	1.3253	1.4006	1.3535	1.2693	1.2525	1.2598
	55	1.207	1.3395	1.4202	1.3693	1.2792	1.2619	1.2708
	56	1.006	1.3526	1.4379	1.3837	1.2888	1.2714	1.2823
*	57-61	0.805 – 0.000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000

*   **Top and Bottom 5 Points Excluded per Technical Specification B3.2.1.**

These W(Z) values are consistent with Figure 5, and are valid over the HFP T_{avg} temperature range from 580.0 to 587.0°F.

Insert 4. Remaining pages to be renumbered.

INSERT 4 for new COLR Table 3 and Table 4

Table 3

Reactor Trip System Instrumentation - Overtemperature ΔT (OTΔT) Setpoint Parameter Values

Parameter	Value
time constants utilized in lead-lag compensator for differential temperature	$\tau_1 = 0$ seconds, $\tau_2 = 0$ seconds
time constant utilized in lag compensator for differential temperature	$\tau_3 \leq 6$ seconds
fundamental setpoint	$K_1 \leq 114.9\%$ RTP
modifier for temperature	$K_2 = 2.24\%$ RTP per °F
time constants utilized in lead-lag compensator for temperature compensation	$\tau_4 \geq 28$ seconds, $\tau_5 \leq 4$ seconds
time constant utilized in lag compensator for average temperature	$\tau_6 \leq 6$ seconds
indicated loop specific RCS average temperature at RTP	$T' \leq 588.4$ °F
modifier for pressure	$K_3 = 0.177\%$ RTP per psig
reference pressure	$P' \geq 2235$ psig

f_1(AFD) modifier for Axial Flux Difference (AFD):

1. for AFD between -23% and +10%, = 0% RTP

2. for each % AFD is below -23%, the trip setpoint shall be reduced by 3.3% RTP

3. for each % AFD is above +10%, the trip setpoint shall be reduced by 1.95% RTP

(p) The compensated temperature difference shall be no more negative than 3 degrees F.

Table 4

**Reactor Trip System Instrumentation - Overpower ΔT (OPΔT)
Setpoint Parameter Values**

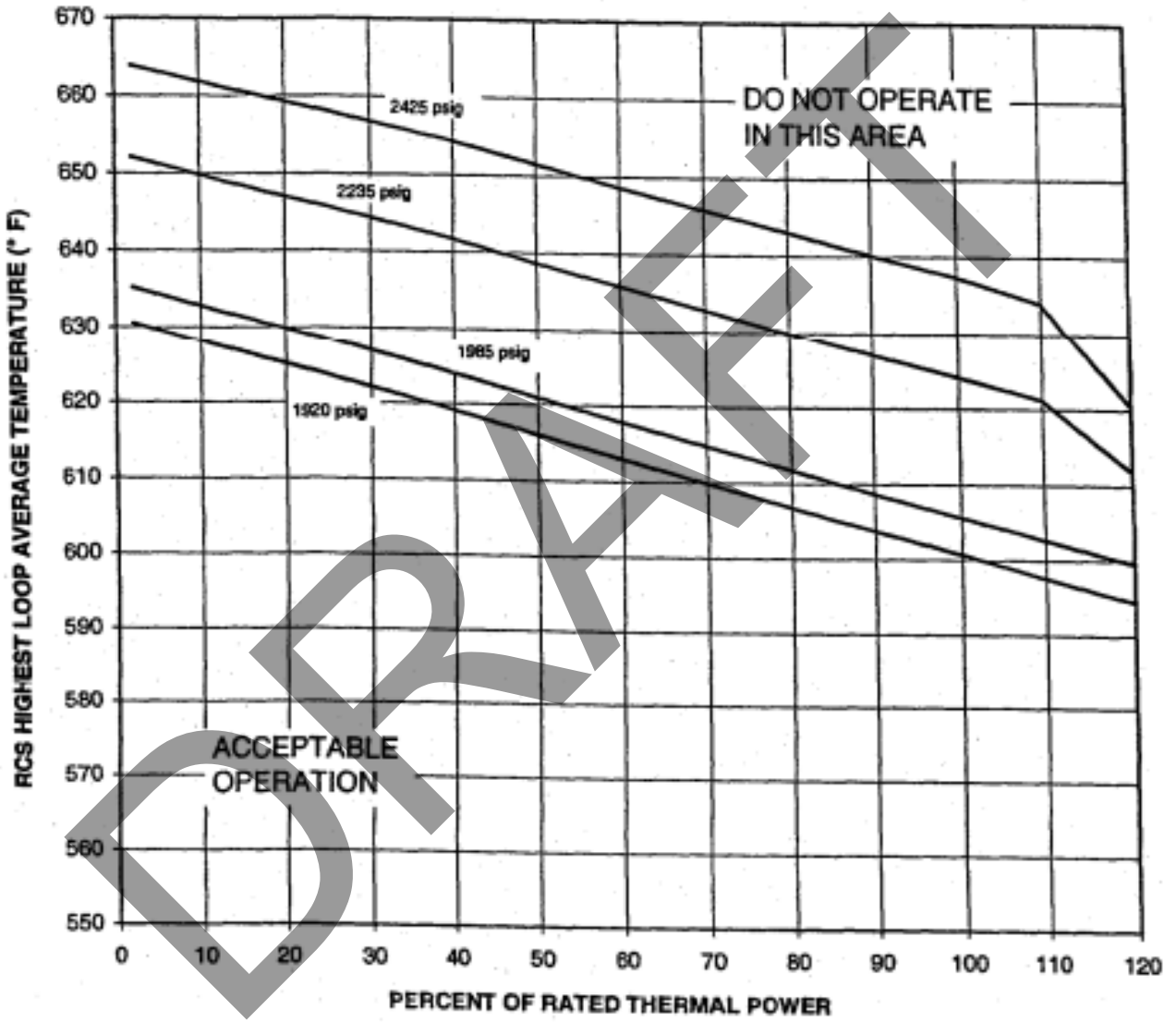| **Parameter**	**Value**
time constants utilized in lead-lag compensator for differential temperature	$\tau_1 = 0$ seconds, $\tau_2 = 0$ seconds
time constant utilized in lag compensator for differential temperature	$\tau_3 \leq 6$ seconds
fundamental setpoint	$K_4 \leq 110\%$ RTP
modifier for temperature change, for <u>increasing</u> temperature	$K_5 \geq 2\%$ RTP per °F
modifier for temperature change, for <u>decreasing</u> temperature	$K_5 \geq 0\%$ RTP per °F
time constants utilized in rate-lag compensator for temperature compensation	$\tau_7 \geq 10$ seconds
time constant utilized in lag compensator for average temperature	$\tau_6 \leq 6$ seconds
modifier for temperature, for $T > T''$	$K_6 \geq 0.244\%$ RTP per °F
modifier for temperature, for $T \leq T''$	$K_6 = 0\%$ RTP per °F
modifier for Axial Flux Difference (AFD), for all AFD	$f_2(AFD) = 0\%$ RTP

INSERT 5 for new COLR Figure 7

FIGURE 7
REACTOR CORE SAFETY LIMITS

1.0   CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for VEGP Unit 2 Cycle 23 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The Technical Requirement affected by this report is listed below:

13.1.1   SHUTDOWN MARGIN - MODES 1 and 2

The Technical Specifications affected by this report are listed below:

Insert 1

3.1.1	SHUTDOWN MARGIN - MODES 3, 4 and 5	
3.1.3	Moderator Temperature Coefficient	
3.1.5	Shutdown Bank Insertion Limits	
3.1.6	Control Bank Insertion Limits	
3.2.1	Heat Flux Hot Channel Factor - $F_Q(Z)$	
3.2.2	Nuclear Enthalpy Rise Hot Channnel Factor - $F^N_{\Delta H}$	
3.2.3	Axial Flux Difference	
3.9.1	Boron Concentration	

Insert 2

COLR Section 1.0 INSERT 1

2.1.1   Reactor Core Safety Limits for THERMAL POWER

COLR Section 1.0 INSERT 2

3.3.1   Reactor Trip System Instrumentation

3.4.1   Reactor Coolant System Pressure, Temperature, and Flow Departure from Nucleate
               Boiling Limits

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

   2.7.1 $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} \cdot \left(1 + PF_{\Delta H} \cdot \left(1 - P \right) \right)$

        where: $P = \dfrac{THERMAL POWER}{RATED THERMAL POWER}$

   2.7.2 $F^{RTP}_{\Delta H} = 1.65$

   2.7.3 $PF_{\Delta H} = 0.3$

2.8   Axial Flux Difference (Specification 3.2.3)

   2.8.1   The Axial Flux Difference (AFD) Acceptable Operation Limits are provided in Figure 5.

2.9   Boron Concentration (Specification 3.9.1)

   2.9.1   The boron concentration shall be greater than or equal to 2000 ppm.[2]

Insert 3

---

[2]   This concentration bounds the condition of $k_{eff} \leq 0.95$ (all rods in less the most reactive rod) and subcriticality (all rods out) over the entire cycle. This concentration includes additional boron to address uncertainties and B^{10} depletion.

INSERT 3 for new COLR Sections 2.10, 2.11 and 2.12

2.10   Reactor Core Safety Limits for THERMAL POWER (Specification 2.1.1)

        2.10.1 In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the safety limits specified in Figure 7.

2.11   Reactor Trip System Instrumentation (Specification 3.3.1)

        2.11.1 Reactor Trip System Instrumentation Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) setpoint parameter values for TS Table 3.3.1-1 are listed in COLR Tables 3 and 4.

2.12   Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits (Specification 3.4.1)

        2.12.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

            a. Pressurizer pressure ≥ 2199 psig;

            b. RCS average temperature ≤ 592.5°F; and

            c. The minimum RCS total flow rate shall be ≥ 384,509 GPM when using the precision heat balance method.

Table 2
RAOC W(Z)

Axial Point	Elevation (feet)	150 MWD/MTU	3000 MWD/MTU	8000 MWD/MTU	12000 MWD/MTU	16000 MWD/MTU	20000 MWD/MTU
* 1-5	12.072 – 11.267	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
6	11.066	1.3208	1.3905	1.3305	1.2998	1.2263	1.2382
7	10.865	1.2951	1.3779	1.3249	1.2954	1.2225	1.2320
8	10.664	1.2800	1.3569	1.3138	1.2868	1.2108	1.2224
9	10.462	1.2661	1.3333	1.3005	1.2779	1.1996	1.2102
10	10.261	1.2426	1.3096	1.2863	1.2663	1.2031	1.2052
11	10.060	1.2217	1.2811	1.2698	1.2537	1.2091	1.2090
12	9.859	1.2105	1.2500	1.2509	1.2390	1.2157	1.2101
13	9.658	1.2105	1.2183	1.2364	1.2340	1.2135	1.2090
14	9.456	1.2078	1.1994	1.2231	1.2303	1.2142	1.2085
15	9.255	1.1975	1.1986	1.2246	1.2222	1.2099	1.2061
16	9.054	1.1917	1.2004	1.2286	1.2136	1.2047	1.2114
17	8.853	1.1992	1.1980	1.2283	1.2037	1.2020	1.2282
18	8.652	1.2022	1.1964	1.2264	1.1964	1.2090	1.2405
19	8.450	1.2059	1.1916	1.2213	1.1988	1.2197	1.2490
20	8.249	1.2077	1.1870	1.2177	1.2000	1.2314	1.2585
21	8.048	1.2106	1.1833	1.2142	1.2079	1.2412	1.2672
22	7.847	1.2142	1.1804	1.2111	1.2133	1.2488	1.2753
23	7.646	1.2138	1.1778	1.2122	1.2166	1.2540	1.2876
24	7.444	1.2117	1.1736	1.2104	1.2182	1.2578	1.2966
25	7.243	1.2066	1.1671	1.2050	1.2159	1.2572	1.3002
26	7.042	1.1998	1.1591	1.1972	1.2111	1.2537	1.2997
27	6.841	1.1915	1.1516	1.1881	1.2050	1.2487	1.2971
28	6.640	1.1813	1.1445	1.1772	1.1970	1.2425	1.2918
29	6.438	1.1707	1.1379	1.1649	1.1874	1.2366	1.2837
30	6.237	1.1620	1.1336	1.1514	1.1777	1.2295	1.2728
31	6.036	1.1558	1.1310	1.1371	1.1671	1.2213	1.2599
32	5.835	1.1488	1.1277	1.1262	1.1593	1.2116	1.2486
33	5.634	1.1432	1.1258	1.1237	1.1572	1.2025	1.2401
34	5.432	1.1396	1.1341	1.1255	1.1631	1.2003	1.2352
35	5.231	1.1397	1.1447	1.1357	1.1672	1.2024	1.2351
36	5.030	1.1473	1.1543	1.1445	1.1709	1.2032	1.2343
37	4.829	1.1543	1.1635	1.1530	1.1754	1.2045	1.2335
38	4.628	1.1609	1.1720	1.1609	1.1789	1.2045	1.2316
39	4.426	1.1668	1.1799	1.1681	1.1812	1.2030	1.2277
40	4.225	1.1716	1.1867	1.1744	1.1825	1.2000	1.2222
41	4.024	1.1757	1.1928	1.1800	1.1827	1.1968	1.2166
42	3.823	1.1786	1.1975	1.1846	1.1834	1.1940	1.2090
43	3.622	1.1833	1.2043	1.1878	1.1842	1.1900	1.1987
44	3.420	1.1894	1.2129	1.1918	1.1838	1.1853	1.1875
45	3.219	1.1935	1.2203	1.1959	1.1831	1.1800	1.1773
46	3.018	1.2016	1.2304	1.2029	1.1818	1.1762	1.1734
47	2.817	1.2120	1.2504	1.2109	1.1901	1.1815	1.1808
48	2.616	1.2235	1.2731	1.2224	1.2037	1.1924	1.1934
49	2.414	1.2401	1.2947	1.2402	1.2163	1.2029	1.2054
50	2.213	1.2579	1.3167	1.2594	1.2292	1.2136	1.2176
51	2.012	1.2753	1.3380	1.2776	1.2409	1.2228	1.2279
52	1.811	1.2924	1.3590	1.2955	1.2522	1.2316	1.2378
53	1.610	1.3086	1.3786	1.3125	1.2634	1.2408	1.2483
54	1.408	1.3238	1.3969	1.3287	1.2743	1.2503	1.2593
55	1.207	1.3382	1.4139	1.3438	1.2847	1.2597	1.2704
56	1.006	1.3515	1.4291	1.3576	1.2948	1.2692	1.2820
* 57-61	0.805 – 0.000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000

*   **Top and Bottom 5 Points Excluded per Technical Specification B3.2.1.**

These W(Z) values are consistent with Figure 5, and are valid over the HFP T_{avg} temperature range from 582.0 to 587.0°F.

Insert 4. Remaining pages to be renumbered.

INSERT 4 for new COLR Table 3 and Table 4

Table 3

**Reactor Trip System Instrumentation - Overtemperature ΔT (OTΔT)
Setpoint Parameter Values**

Parameter	**Value**
time constants utilized in lead-lag compensator for differential temperature	$\tau_1 = 0$ seconds, $\tau_2 = 0$ seconds
time constant utilized in lag compensator for differential temperature	$\tau_3 \leq 6$ seconds
fundamental setpoint	$K_1 \leq 114.9\%$ RTP
modifier for temperature	$K_2 = 2.24\%$ RTP per °F
time constants utilized in lead-lag compensator for temperature compensation	$\tau_4 \geq 28$ seconds, $\tau_5 \leq 4$ seconds
time constant utilized in lag compensator for average temperature	$\tau_6 \leq 6$ seconds
indicated loop specific RCS average temperature at RTP	$T' \leq 588.4$ °F
modifier for pressure	$K_3 = 0.177\%$ RTP per psig
reference pressure	$P' \geq 2235$ psig

f_1(AFD) modifier for Axial Flux Difference (AFD):

1. for AFD between -23% and +10%, = 0% RTP

2. for each % AFD is below -23%, the trip setpoint shall be reduced by 3.3% RTP

3. for each % AFD is above +10%, the trip setpoint shall be reduced by 1.95% RTP

(p) The compensated temperature difference shall be no more negative than 3 degrees F.

Table 4

Reactor Trip System Instrumentation - Overpower ΔT (OPΔT)
Setpoint Parameter Values

Parameter	**Value**
time constants utilized in lead-lag compensator for differential temperature	$\tau_1 = 0$ seconds, $\tau_2 = 0$ seconds
time constant utilized in lag compensator for differential temperature	$\tau_3 \leq 6$ seconds
fundamental setpoint	$K_4 \leq 110\%$ RTP
modifier for temperature change, for <u>increasing</u> temperature	$K_5 \geq 2\%$ RTP per °F
modifier for temperature change, for <u>decreasing</u> temperature	$K_5 \geq 0\%$ RTP per °F
time constants utilized in rate-lag compensator for temperature compensation	$\tau_7 \geq 10$ seconds
time constant utilized in lag compensator for average temperature	$\tau_6 \leq 6$ seconds
modifier for temperature, for $T > T''$	$K_6 \geq 0.244\%$ RTP per °F
modifier for temperature, for $T \leq T''$	$K_6 = 0\%$ RTP per °F
modifier for Axial Flux Difference (AFD), for all AFD	$f_2(AFD) = 0\%$ RTP

FIGURE 7
REACTOR CORE SAFETY LIMITS