VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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U. S. Nuclear Regulatory Commission
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Serial No. 09-581
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Docket Nos. 50-280/281
License Nos. DPR-32/37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED LICENSE AMENDMENT REQUEST
RELOCATION OF CORE OPERATING LIMITS TO THE CORE OPERATING LIMITS REPORT (COLR) AND ADDITION OF COLR REFERENCES

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The proposed changes to the TS are summarized below.

- Analytical Methods Used to Determine the Core Operating Limits

  The license amendment request (LAR) adds two references to the list of NRC approved methodologies contained in TS. Specifically, Westinghouse document WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function," is the NRC-approved methodology that is used to meet the analytical and design bases for the Overtemperature ΔT and Overpressure ΔT setpoint parameters that are being relocated to the Core Operating Limits Report (COLR) as discussed further below. WCAP-8745-P-A will be added as a referenced analytical methodology report in TS 6.2.C, "CORE OPERATING LIMITS REPORT".

  The LAR also requests the inclusion of NRC-approved Dominion Fleet Report DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix B, "Qualification of the Westinghouse WRB-1 CHF Correlation in the Dominion VIPRE-D Computer Code," in TS 6.2.C as a referenced analytical methodology report. The inclusion of DOM-NAF-2-A, including Appendix B, will permit the use of an alternate methodology to perform thermal-hydraulic analyses to predict Critical Heat Flux (CHF) and the Departure from Nucleate Boiling Ratio (DNBR) for Westinghouse 15x15 Upgrade fuel assemblies. Dominion is planning to use Westinghouse 15x15 Upgrade fuel in the Surry Unit 1 and Unit 2 cores as discussed further below. In addition, plant specific application of the methodology requires NRC approval of the Statistical Design Limit (SDL) for the relevant code/correlation pair. Consequently, in addition to the inclusion of Fleet Report DOM-NAF-2-A, including Appendix B, in TS 6.2.C, Dominion also requests NRC review and approval of the implementation of the Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for Westinghouse 15x15

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Upgrade fuel at Surry using the VIPRE-D/WRB-1 code/correlation, as well as the SDL obtained by this implementation. Pursuant to 10 CFR 50.59, the SDL discussed in Attachment 4 requires NRC review and approval since an SDL constitutes a Design Basis Limit for Fission Product Barrier (DBLFPB).

Westinghouse 15x15 Upgrade fuel assemblies have three intermediate flow mixer (IFM) grids that provide Departure from Nucleate Boiling (DNB) margin improvement. These assemblies are a replacement for the resident fuel product, which is a Westinghouse 15x15 VANTAGE+ fuel product, also referred to as Surry Improved Fuel (SIF). SIF does not include IFM grids. The IFM grids in the Westinghouse 15x15 Upgrade fuel assemblies have a grid spacing of 13 inches. Currently, there is a restriction in Appendix B of DOM-NAF-2-A stating that the VIPRE-D/WRB-1 code/correlation will not be used for fuel with less than 13-inch mixing vane grid spacing. Dominion has performed in-house validation of Westinghouse test section data, which allows removal of this restriction. A licensing submittal dated August 28, 2009 (Serial No. 09-528) was provided to the NRC for their review and approval for the removal of the grid spacing restriction. Approval of that request is required to apply the VIPRE-D/WRB-1 code/correlation and to implement the Statistical DNBR Evaluation Methodology for Westinghouse 15x15 Upgrade fuel.

Upon NRC approval of the LAR, the SDL and the removal of the grid spacing restriction, Dominion will be licensed to perform in-house DNB analyses, for the intended uses described in DOM-NAF-2-A using VIPRE-D to support the use of Westinghouse 15x15 Upgrade fuel at Surry Units 1 and 2. Dominion is currently planning to use Westinghouse 15x15 Upgrade fuel in Surry Units 1 and 2 commencing with Surry Unit 1, Cycle 25 (Spring 2012) and Surry Unit 2, Cycle 24 (Spring 2011).

- Relocation of TS Parameters to the COLR

  The LAR also proposes to incorporate changes consistent with Industry/TSTF Standard Technical Specification Change Traveler TSTF-339-A, Revision 2, "Relocate TS Parameters to COLR," into the Surry TS. TSTF-339-A, Revision 2, permits the relocation of cycle-specific parameters including the reactor core safety limits, Overtemperature ΔT and Overpower ΔT setpoints, and DNB parameter limits from the TS to the COLR.

- Additional TS Changes

  Additional TS changes are being implemented to provide consistency with the Improved TS format in NUREG-1431, Revision 3, "Standard Technical Specifications, Westinghouse Plants," where practical, and to delete obsolete TS requirements. TS Basis changes reflecting the proposed changes have also been included for the NRC's information.

A discussion of the proposed changes is provided in Attachment 1. The marked-up and typed proposed TS pages are provided in Attachments 2 and 3, respectively.

Attachment 4 provides 1) the technical basis for adding Dominion Fleet Report DOM-NAF-2-A including Appendix B to the list of NRC approved methodologies for determining core operating limits, 2) the implementation of the Dominion Statistical DNBR Evaluation Methodology for Westinghouse 15x15 Upgrade fuel at Surry using the VIPRE-D/WRB-1 code/correlation, and 3) the SDL obtained by this implementation.

We have evaluated the proposed amendment and have determined that it does not involve a significant hazards consideration as defined in 10CFR50.92. The basis for our determination is included in Attachment 1. We have also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The basis for our determination is also included in Attachment 1. The proposed amendment has been reviewed and approved by the Facility Safety Review Committee.

Approval of the proposed amendments is requested by September 30, 2010. Dominion requests a 60-day implementation period following NRC approval of the requested license amendments.

If you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 16th day of October, 2009.

My Commission Expires: 4/30/13.

Ginger L. Alligood
Notary Public

GINGER L. ALLIGOOD
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

Attachments:
1. Discussion of Change
2. Proposed Technical Specifications Pages (Mark-Up)
3. Proposed Technical Specifications Pages (Typed)
4. Technical Basis for Adding Fleet Report DOM-NAF-2-A to the List of NRC Approved Methodologies for Determining Core Operating Parameters

Commitments made in this letter: None

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ATTACHMENT 1

DISCUSSION OF CHANGE

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

DISCUSSION OF CHANGE

1. SUMMARY DESCRIPTION

Virginia Electric and Power Company (Dominion) is purchasing fuel assemblies from Westinghouse for use at Surry Power Station, Units 1 and 2. These assemblies are scheduled to be inserted in Units 1 and 2, commencing with Cycles 25 and 24 respectively. The fuel assemblies are designated as the 15x15 Upgrade (Reference 20), which have three intermediate flow mixer (IFM) grids for departure from nucleate boiling (DNB) margin improvement. These assemblies are a replacement for the resident fuel product, which is a Westinghouse 15x15 VANTAGE+ fuel product, also referred to as Surry Improved Fuel (SIF), which does not have IFMs.

Dominion proposes the following changes to the Surry Power Station Units 1 and 2 Technical Specifications (TS) pursuant to 10 CFR 50.90. The proposed changes add Topical Reports DOM-NAF-2-A including Appendix B (Reference 10) and WCAP-8745-P-A (Reference 17) to the TS 6.2.C list of NRC-approved methodologies for determining core operating limits. These changes also propose to incorporate Technical Specification Task Force Traveler (TSTF) 339-A Revision 2 (Reference 2) into the Surry TS. TSTF-339-A relocates the cycle-specific limits from TS 2.1.A.1 Reactor Core Safety Limits, TS 2.3.A.2.d Overtemperature ΔT, TS 2.3.A.2.e Overpower ΔT, and TS 3.12.F DNB Parameters to the Core Operating Limits Report (COLR). In addition to the changes listed above, other additions and deletions to the plant TS are included to develop consistency with the Standard (Improved) Technical Specifications (ITS) format in NUREG-1431, Rev. 3 (Reference 1), where practical, and to delete requirements that are obsolete. TS Basis changes reflecting the proposed TS changes have been included for the NRC's information.

Approval of these changes will allow Dominion to use the VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores, using the deterministic design limits (DDLs) documented in Appendix B of the DOM-NAF-2-A Fleet Report and the statistical design limit (SDL) documented in Attachment 4. The DDLs were approved as part of the review and approval of DOM-NAF-2-A Appendix B (Reference 10). Dominion requests the review and approval of the SDL documented herein.

Westinghouse 15x15 Upgrade fuel assemblies include IFM grids which have a grid spacing of 13 inches. There is a restriction in Appendix B of DOM-NAF-2-A (Reference 10) that states that the fuel must have a mixing vane grid spacing greater than 13 inches. Dominion has performed in-house validation of Westinghouse test section data to be able to remove this restriction. A licensing submittal was provided to the NRC for their review and approval by letter dated August 28, 2009 [Serial No. 09-528 (Reference 19)] for the removal of the grid spacing restriction. Approval of the grid spacing restriction removal is required to apply VIPRE-D/WRB-1 and to implement the Statistical DNBR Evaluation Methodology (Reference 6) for 15x15 Upgrade fuel.

The proposed TS change has been reviewed, and it has been determined that no significant hazards consideration exists as defined in 10 CFR 50.92. In addition, it has been determined that the change qualifies for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9); therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed TS change.

2. DETAILED DESCRIPTION

2.1 Proposed Change

The following specific changes to the Surry Units 1 and 2 TS are proposed:

- TS 2.1 Safety Limit, Reactor Core
  - Revise TS 2.1.A.1 to refer to the CORE OPERATING LIMITS REPORT instead of TS Figure 2.1-1, and to add the statement "and the following Safety Limits shall not be exceeded."
  - Revise TS 2.1.A.1 to add TS 2.1.A.1.a and TS 2.1.A.1.b, which support relocating the Reactor Core Safety Limits to the CORE OPERATING LIMITS REPORT.
  - Revise TS 2.1.B to refer to the CORE OPERATING LIMITS REPORT instead of TS Figures 2.1-1, 2.1-2 or 2.1-3.
  - Delete TS Figure 2.1-1 since this change relocates TS Figure 2.1-1 to the CORE OPERATING LIMITS REPORT.
  - Delete obsolete TS 2.1.A.2 and TS 2.1.A.3, and TS Figures 2.1-2, 2.1-3, and 2.1-4.
  - Revise TS 2.1 Basis to reflect TS changes and delete obsolete information.
- TS 2.3 Limiting Safety System Settings, Protective Instrumentation
  - Revise TS 2.3.A.2.d and 2.3.a.2.e to make the format consistent with NUREG-1431, Rev. 3 and relocate the cycle-specific parameters to the CORE OPERATING LIMITS REPORT.
  - Delete TS Figure 2.3-1.
  - Revise TS 2.3 Basis to reflect TS changes.

- <u>TS 3.12.F DNB Parameters</u>

  - Revise TS 3.12.F.1 to remove numerical limits for Reactor Coolant System T_{avg} and pressurizer pressure and replace with reference to CORE OPERATING LIMITS REPORT.

  - Revise TS 3.12.F.1 to include "and ≥ the limit specified in the CORE OPERATING LIMITS REPORT," in addition to the current limit listed for RCS Total Flow Rate.

  - Revise TS 3.12.F.1.a to add Reactor Coolant System Total Flow rate to list of parameters that are to be verified to be within their limits once every 12 hours.

  - Revise TS 3.12.F.1.b to clarify that Reactor Coolant System Total Flow Rate is determined to be within its limit by precision heat balance with the frequency specified in TS Table 4.1-2A.

  - Revise TS 3.12.F.2 to change the time requirement to reduce THERMAL POWER to make it consistent with NUREG-1431, Rev. 3.

  - Revise TS 3.12 Basis to reflect TS requirements.

- <u>Table 4.1-2A Minimum Frequency for Equipment Tests</u>

  - Revise Item 22 to include "and ≥ the limit as specified in the CORE OPERATING LIMITS REPORT."

  - Revise Item 22 to include a note that allows entry into POWER OPERATION, without having performed the Surveillance Requirement (SR), and placement of the unit in the best condition for performing the SR.

  - Revise TS 4.1 Basis to reflect TS requirements.

- <u>TS 6.2.C Core Operating Limits Report</u>

  - Revise TS 6.2.C to include the following Technical Specification Sections: TS 3.12.F - DNB Parameters, TS 2.1.A.1 – Safety Limit, Reactor Core, TS 2.3.A.2.d – Overtemperature ΔT, TS 2.3.A.2.e – Overpower ΔT, and TS Table 4.1-2A Minimum Frequency for Equipment Tests: Item 22 – RCS Flow.

  - Revise TS 6.2.C to include references to DOM-NAF-2-A, including Appendix B, and WCAP-8745-P-A. The COLR references have been renumbered.

3. TECHNICAL EVALUATION

The following discussion provides justification for the proposed change discussed in Section 2.1 above. The proposed change adds DOM-NAF-2-A including Appendix B and WCAP-8745-P-A to the list of NRC-approved methodologies, incorporates TSTF-339-A Rev. 2 and removes obsolete TS. This change is being made to achieve consistency with NUREG-1431 Rev. 3 (Reference 1) with respect to requirements and action statements for Safety Limits, Overtemperature and Overpower ΔT setpoints, and DNB parameters. Additional minor changes are proposed to make terminology corrections between the current Surry TS and ITS wording. Related TS Basis changes reflecting the proposed TS revisions are included for the NRC's information.

__TS 2.1: Safety Limits, Reactor Core__

The proposed changes to TS 2.1, "Safety Limit, Reactor Core," will relocate the Reactor Core Safety Limits figure (TS Figure 2.1-1) to the COLR, add new Safety Limits for departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature (TS 2.1.A.1.a and TS 2.1.A.1.b, respectively), and delete obsolete limits associated with N-1 Loop operation, since two-loop operation is prohibited for Surry (TS 2.1.A.2, TS 2.1.A.3, TS Figure 2.1-2, TS Figure 2.1-3) and fuel densification (TS Figure 2.1-4).

In addition, TS 2.1.A.1 will be revised to read: "Exceed the limits specified in the CORE OPERATING LIMITS REPORT when full flow from three reactor coolant pumps exist, and the following Safety Limits shall not be exceeded:

a. The design limit departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.27 for transients analyzed using the Statistical DNBR Evaluation Methodology and the WRB-1 DNB correlation. For transients analyzed using the deterministic methodology, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit (≥ 1.17 for WRB-1, ≥ 1.30 for W-3).

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

The guidance from TSTF 339-A Rev. 2 and WCAP-14483-A (Reference 3) has been used to relocate the Reactor Core Safety Limits in TS 2.1.A.1 from the TS to the COLR. The proposed change adds TS 2.1.A.1.a and TS 2.1.A.1.b that state the requirements that must be met to ensure that the Safety Limits will not be exceeded. The wording for TS 2.1.A.1.a has been modified from that provided in TSTF 339-A Rev. 2, WCAP-14483-A, and NUREG-1431 Rev. 3. The justification for deviating from the wording in the TSTF is to identify the statistical DNBR limit for the WRB-1 correlation, and the deterministic DNBR limits for the W-3 and WRB-1 DNB correlations. This deviation was previously approved by the NRC for Callaway (References 21 and 22). TS Figure 2.1-1 is being deleted from the TS since it is being relocated to the COLR.

Sections 3.2.3.3 and 3.4 of the Surry Updated Final Safety Analysis Report (UFSAR) describe the reactor core thermal-hydraulic design criteria, evaluation methods, and DNBR limits. UFSAR Chapter 14 includes transients that are analyzed to verify that DNBR limits are met.

DNB analyses for the resident Westinghouse 15x15 SIF product use the NRC-approved COBRA code and the W-3 (Reference 4) or WRB-1 (Reference 5) correlation, depending on the transient. The Dominion Statistical DNBR Evaluation Methodology in Topical Report VEP-NE-2-A (Reference 6) is applied to all statistically-treated events. Reference 7 submitted the analysis to support the implementation of VEP-NE-2-A for the Westinghouse 15x15 SIF product. Reference 7 documented a SDL of 1.27 for transients analyzed using the Statistical DNBR Evaluation Methodology and the WRB-1 DNB correlation, and for non-statistical DNBR transients, the DNBR limit was set to 1.17 for WRB-1 and 1.30 for W-3. The NRC approved the use of the VEP-NE-2-A methodology and the current DNBR limits for Surry in Reference 8. (Note: Reference 9 provided a clarification on the application of the DNBR limit of 1.30 for W-3.) Therefore, the proposed limits for the TS have already received NRC review and approval for application to the Westinghouse 15x15 SIF fuel product.

DNB analyses for the Westinghouse 15x15 Upgrade product will use the NRC-approved VIPRE-D code and the W-3 or WRB-1 correlation, DOM-NAF-2-A (Reference 10), depending on the transient. The Dominion Statistical DNBR Evaluation Methodology in Topical Report VEP-NE-2-A (Reference 6) is applied to all statistically-treated events. The analysis to support the implementation of DOM-NAF-2-A and VEP-NE-2-A for the Westinghouse 15x15 Upgrade product is summarized below. The resulting design limits are 1.27 for transients analyzed using the Statistical DNBR Evaluation Methodology and the WRB-1 DNB correlation, documented herein, and 1.17 for WRB-1 and 1.30 for W-3 for non-statistical DNBR transients (Reference 10). Upon NRC review and approval of this change request, these DNBR design limits will be approved for application to the Westinghouse 15x15 Upgrade fuel product at Surry.

In particular, Attachment 4 provides the technical basis for the USNRC review and approval of the implementation of the Dominion Statistical DNBR Evaluation Methodology (Reference 6) for Westinghouse 15x15 Upgrade fuel at Surry with the VIPRE-D/WRB-1 code/correlation, as well as the SDL obtained by this implementation. Attachment 4 also documents that the existing Reactor Core Safety Limits and protection functions (e.g., OTΔT, OPΔT, fΔI, etc.) remain bounding (i.e., do not require revision) as a consequence of this implementation. The list of UFSAR transients for which the VIPRE-D/WRB-1 code/correlation set will be applied is also included in Table 3.9-1 of Attachment 4. Applicable Chapter 14 analyses were evaluated with the VIPRE-D/WRB-1 code/correlation and the Statistical DNBR Evaluation Methodology, and were demonstrated to have acceptable results. These evaluations accounted for the potential inclusion of a Measurement Uncertainty Recapture (MUR) uprate of 1.7% and will become the Analysis of Record (AOR) for the VIPRE-D implementation concurrent with the transition to 15x15 Upgrade fuel product at Surry.

The proposed limit for peak fuel centerline temperature being added into TS 2.1.A.1.b is from NRC-approved WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report" (Reference 11). This reference is already included in the TS 6.2.C list of approved references.

This change request also proposes to delete TS 2.1.A.2 and TS 2.1.A.3, as well as TS Figures 2.1-2 and 2.1-3 that are referenced by the TS. TS 2.1.A.2 and TS 2.1.A.3 support two-loop operation at Surry Power Station which is not allowed by TS 3.1.A.1.a, which requires that three reactor coolant pumps be in operation in non-isolated loops whenever the reactor is critical. TS 2.1.A.2 and TS 2.1.A.3 and the referenced figures are obsolete and can therefore be deleted.

The proposed change also deletes TS Figure 2.1-4, "Thermal Overpower Limit." TS Figure 2.1-4 was first added to the TS via Change #9, which was approved by the NRC in a letter dated August 9, 1973 (Reference 12). Change #9 provided changes to accommodate the impact of peaking effects due to fuel densification. TS Figure 2.1-4 was referenced in TS 2.1.A.4, TS 2.1.B, and in the TS Basis. Change #15 which was approved by the NRC in a letter dated April 19, 1974 (Reference 13) removed the references to TS Figure 2.1-4 from TS 2.1.A.4, TS 2.1.B, and in the TS Basis, but did not delete TS Figure 2.1-4. Since that time, TS Figure 2.1-4 has been unconnected to any portion of the TS. TS Figure 2.1-4 is obsolete and will be deleted by this proposed change.

TS 2.3: Limiting Safety System Settings, Protective Instrumentation

The proposed change to TS 2.3, "Limiting Safety System Settings, Protective Instrumentation," will relocate the Overtemperature ΔT and Overpower ΔT setpoint parameters [nominal RCS average temperature (Tavg), nominal RCS operating pressure, K values, dynamic compensation time constants, and the breakpoint and slope values for the f(ΔI) penalty function] from TS 2.3.A.2.d and TS 2.3.A.2.e to the COLR. The proposed change will also delete TS Figure 2.3-1, "OPΔT and OTΔT f(ΔI) Function".

TSTF 339-A Rev. 2 and NUREG-1431 Rev. 3 were used to relocate the setpoint terms associated with the Overtemperature and Overpower ΔT protection circuitry, and change the equalities associated with the K terms, and the nominal values for Tavg and pressure with inequalities. The formulas in the TSTF 339-A Rev. 2 have been modified to reflect the current formulas in Surry TS for these functions. The additional time constants in the TSTF 339-A Rev. 2 formulation of the Overtemperature and Overpower ΔT equations have not been included since they do not represent the Surry TS Overtemperature and Overpower ΔT equations. The equalities have been changed to inequalities in a conservative direction consistent with Table 3.3.1-1of NUREG-1431 Rev. 3. Changing the equalities to inequalities is acceptable because requirements continue to ensure that the process variables are maintained consistent with the safety analyses and licensing basis. The value of each constant of the Overtemperature and Overpower ΔT functions is only allowed to vary in the conservative direction for the function. This will ensure their setpoints will not exceed the safety analyses assumptions for these functions.

The proposed change for Surry also corrects two errors regarding units of measure for the equation f(ΔI) used to determine the Overtemperature ΔT function Allowable Value. Specifically, the changes will correct the units in the equation "f(ΔI) = 0% of RATED POWER", whereby the unit designation "% of RATED POWER" will be deleted since the f(ΔI) function is dimensionless, and in the equation for f(ΔI) = (*) { (*) % - (qt - qb)}, whereby the percentage designation "%" will also be deleted. The proposed changes for Surry do not have any impact on equipment operability. The changes only relate to the unit designations for the f(ΔI) equation that were incorrectly assigned. These errors do not affect the value of the variable, or the way in which the plant or equipment is operated. These changes are deviations from Table 3.3.1-1 in both TSTF-339-A, Rev. 2 and NUREG-1431, Rev. 3. Dominion previously requested identical changes to the North Anna TS in a letter dated July 14, 2005 (Reference 14). The NRC reviewed and approved the correction of these errors for North Anna in a letter dated October 25, 2005 (Reference 15).

The proposed change will also delete TS Figure 2.3-1, "OPΔT and OTΔT f(ΔI) Function". The current TS states, "f(ΔI) = function of ΔI, percent of rated core power as shown in TS Figure 2.3-1" for the Overtemperature ΔT function and "f(ΔI) as defined in (d) above" for the Overpower ΔT function. The f(ΔI) specification for the Overtemperature ΔT function will now be provided as algebraic formulas in TS 2.3.A.2.d with the cycle-specific limits values denoted with (*) to be specified in the CORE OPERATING LIMITS REPORT. The f(ΔI) specification for the Overpower ΔT function remains unchanged. Thus, the information in the figure is redundant and will be deleted.

TS 3.12.F: DNB Parameters

The proposed changes to TS 3.12.F, "DNB Parameters," will relocate the limits for RCS Tavg, pressurizer pressure, and RCS total flow rate to the COLR. The proposed changes will also add the requirement to verify that RCS flow is within its limits at least once every 12 hours, and modify the requirement to perform a precision RCS flow measurement with the frequency defined in TS Table 4.1-2A.

The relocation of the DNB parameter limits from TS 3.12.F.1 to the COLR will provide additional flexibility to support future reload cores for Surry. By relocating the limits for the DNB parameters to the COLR, cycle-specific limits for RCS Tavg, pressurizer pressure, and RCS total flow rate can be adjusted to provide additional margin and flexibility to support future plant operation. The relocation of the DNB parameter limits is being done in accordance with NRC guidance provided in Generic Letter 88-16 (Reference 16). The methodology being used for moving the cycle-specific parameter limits from the TS to the COLR is in accordance with TSTF-339-A, Rev. 2 and WCAP-14483-A.

A new requirement is being added to TS 3.12.F.1.a to verify that the RCS flow is within its limits at least once every 12 hours. The 12-hour RCS flow surveillance is used to monitor degradation of the RCS flow during the operating cycle. The required verification of RCS Tavg and pressurizer pressure are contained in the existing TS. This change adds the requirement to verify that the RCS flow is within its limits on the same frequency. The proposed changes to TS 3.12.F.1 will ensure that the core operates within the limits assumed in the safety analysis. This change is consistent with TSTF-339-A, Rev. 2, WCAP-14483-A, and NUREG-1431, Rev. 3.

The proposed change also includes a modification to the limit and the surveillance frequency in TS 3.12.F.1.b for performing a precision RCS flow measurement. The proposed change states that "The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by precision heat balance with the frequency specified in TS Table 4.1-2A." The minimum limit for RCS total flow rate specified in TS 3.12.F.1 is identical to the value used in Reference 7 and approved in Reference 8. The limit in the COLR may be set to a higher value to accommodate future changes to the plant configuration. The specification of the limit is consistent with TSTF-339-A, Rev. 2, WCAP-14483-A, and NUREG-1431, Rev. 3. The proposed change modifies the specified frequency from "by measurement at least once per refueling cycle" to "with the frequency specified in TS Table 4.1-2A". Item 22 of TS Table 4.1-2A currently specifies the frequency as once per 18 months. This frequency meets the intent of the wording in the SR 3.4.1.4 in NUREG-1431 Rev. 3 which requires the specification of a frequency of "[18] months" where the number in the brackets is a site-specific value. Since Surry Units 1 and 2 are refueled every 18 months, the frequency of 18 months is consistent with plant operations and reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

Finally, the proposed change includes a modification to TS 3.12.F.2 to increase the time allowed to reduce thermal power to less than 5% of rated power from 4 hours in the current Surry TS to 6 hours, in the event that any of the DNB parameters in TS 3.12.F.1 has been determined to exceed its limit and cannot be restored to within its limit within 2 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reduce thermal power to 5% from full power conditions in an orderly manner and without challenging plant systems. POWER OPERATION at 5% or less is equivalent to the NUREG-1431 Rev. 3 definition for Mode 2. Surry's TS do not have the ITS mode definitions. Therefore, reducing the power level to 5% or less of RATED POWER meets the intent of the ITS to reduce power and eliminate the potential for violation of the safety analysis. This change is consistent with Required Action B.1 in TS 3.4.1 of NUREG-1431 Rev. 3.

<u>Table 4.1-2A: Minimum Frequency for Equipment Tests</u>

The proposed change to TS 4.1, "Operational Safety Review," modifies Item 22 in TS Table 4.1-2A by adding "and ≥ the limit as specified in the CORE OPERATING LIMITS REPORT" to the flow requirement, and adding a note that allows entry into

POWER OPERATION, without having performed the surveillance requirement (SR), and placement of the unit in the best condition for performing the SR. The first addition is for consistency with the changes proposed for TS 3.12.F. The note states that the SR is "Not required to be performed until 7 days after ≥ 90% RATED POWER." The addition of the note means that the precision flow measurement is required to be made within 7 days after reaching 90% RATED POWER. The power level is consistent with TSTF-339-A, Rev. 2, WCAP-14483-A, and NUREG-1431, Rev. 3. The 7-day time limit of the note is an exception from TSTF-339-A, Rev. 2, WCAP-14483-A, and NUREG-1431, Rev. 3. However, it is less than the 30 days approved for North Anna in Reference 23. The 7-day period after reaching 90% RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results.

TS 6.2.C: Core Operating Limits Report

The proposed change to TS 6.2.C, "Core Operating Limits Report," will add Fleet Report DOM-NAF-2-A including Appendix B (Reference 10) and Westinghouse WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions" (Reference 17) to the list of NRC approved methodologies used to determine core operating limits (i.e., the reference list of the Surry COLR), and will update the list of TS contained in the TS 6.2.C.

The proposed change to TS 2.1 relocates the Reactor Core Safety Limits figure (TS Figure 2.1-1) to the COLR. The Reactor Core Safety Limits 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 Safety Limits are composed of a loci of points of THERMAL POWER, pressurizer pressure, and average RCS temperature for which the minimum DNBR is not less than the safety analysis limit, that fuel 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 hot-leg boiling limits are not true safety limits but act to preclude saturation conditions to ensure that the measured ΔT remains proportional to thermal power. The Reactor Core Safety Limits are used to define the various reactor protection system functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The methodology used to calculate the Reactor Core Safety Limits figure is established in VEP-FRD-42 Rev. 2.1-A, "Reload Nuclear Design Methodology," (Reference 18).

Reference 18 is an NRC-approved reload methodology for Surry Power Station. TS 6.2.C currently cites Reference 18 as one of the analytical methods used to determine the core operating limits. Therefore, the requirement in NRC GL 88-16 (Reference 16) that the NRC-approved methodology used to derive the parameters in the figure be referenced in TS 6.2.C is already satisfied. The acceptability of citing the site-specific approved reload methodology is discussed on page 3 of the NRC Safety Evaluation attached to WCAP-14483-A (Reference 3).

The proposed changes to TS 2.3, "Limiting Safety System Settings, Protective Instrumentation," relocates the Overtemperature ΔT and Overpower ΔT setpoint parameters to the COLR. Westinghouse WCAP-8745-P-A is the NRC reviewed and approved methodology Topical Report that is used to meet the analytical and design basis for the TS 2.3 OTΔT and OPΔT setpoint parameters being relocated to the COLR. Reference 17 will be added as a referenced analytical methodology report in TS 6.2.C. The acceptability of citing this topical report is discussed on page 2 of the NRC Safety Evaluation attached to WCAP-14483-A (Reference 3).

The proposed changes to TS 3.12.F, "DNB Parameters," will relocate the limits for RCS Tavg, pressurizer pressure, and RCS total flow rate to the COLR. References 5, 6, 10 and 18 provide the NRC-approved methodology reports associated with demonstrating that the DNB design basis is met for the 15x15 SIF and 15x15 Upgrade fuel products. TS 6.2.C currently cites References 5, 6 and 18. Citation for Reference 10 will be added to TS 6.2.C. The acceptability of adding this reference to TS 6.2.C is provided by this TSCR.

The proposed changes to TS 6.2.C, “Core Operating Limits Report,” also update the list of TS contained in the Core Operating Limits Report. The following TS are added to the list. This is an administrative change to maintain consistency within the TS.

4. TS 3.12.F – DNB Parameters
5. TS 2.1 – Safety Limit, Reactor Core
6. TS 2.3.A.2.d – Overtemperature ΔT
7. TS 2.3.A.2.e – Overpower ΔT
8. TS Table 4.1-2A – Minimum Frequency for Equipment Tests: Item 22 – RCS Flow

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to develop TS, which are included as a part of the operating license (OL). 10CFR50.36, Technical Specifications, sets forth the content of the TS. This regulation requires the TS to include items in specific categories including, (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements, (4) design features, and administrative controls. 10CFR50.36 does not specify the particular specifications to be included as part of a plant's OL. By letter dated May 9, 1988, (Reference 8) the NRC described results of an NRC staff review to determine which LCOs should be included in the TS. This ultimately resulted in four criteria being developed, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," (Reference 9), which were later codified in 10CFR50.36(c)(2)(ii).

Guidance on the relocation of cycle-specific TS parameters to the COLR is provided to all power reactor licensees and applicants in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 3, 1988. In the GL, the NRC staff stated that license amendments are generally required every refueling outage to update the cycle-specific parameter limits in the TSs; however, there are methodologies developed for the licensee to determine these cycle-specific parameters that have been reviewed and approved by the staff. As a consequence, the NRC staff review of proposed changes to the TSs to update these parameter limits is primarily limited to the confirmation that the updated limits were calculated by the approved methodology and consistent with the appropriate plant-specific safety analysis. The COLR was created to place the NRC-approved methodologies in the TSs and allow licensees to use later revisions of these methodologies to update the parameters without requiring a change to the TSs.

The justification to expand the COLR to relocate additional TS parameters to the COLR is provided in a Westinghouse Topical Report WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," which was approved by the NRC. The NRC staff's letter and safety evaluation (SE) approving the use of the topical report were issued January 19, 1999, and are included in the beginning of the topical report. The topical report addresses the relocation of the (1) reactor core safety limits figure, (2) overtemperature ΔT (OTDT) and overpower ΔT (OPDT) setpoint parameter values for reactor trip instrumentation, and (3) DNB parameter limits in the TSs to the COLR. The NRC staff's SE for the topical report had no conditions specified on the use of the topical report.

The NRC staff has approved WCAP-14483-A as an acceptable method to relocate these TS requirements to the COLR consistent with GL 88-16. The NRC staff incorporated the generic methodology for expanding the COLR of WCAP-14483-A into NUREG-1431, ITS for Westinghouse plants. In TSTF-339-A, Revision 2, "Relocate TS Parameters to COLR," the relocation of the (1) reactor core safety limits figure, (2) OTDT and OPDT setpoint parameter values for reactor trip instrumentation, and (3) DNB parameter limits in the TSs to the COLR in WCAP-14483-A was approved by the NRC staff for NUREG-1431. In TSTF-363, Revision 0, "Relocate Topical Report references in ITS 5.6.5, COLR," the NRC-approved topical reports are listed only by a reference to the report number and title with a note that the COLR provides the complete citation of the report (i.e., report number, title, revision, date, and any supplements). These documents support the changes for TSs 2.1, 2.3, and 3.12.F, 4.1, and 6.2.C that have been proposed herein.

The proposed changes to TS 2.1, "Reactor Core Safety Limits," will add new Safety Limits for DNBR and peak fuel centerline temperature. These limits were developed using NRC-approved methodologies of DOM-NAF-2-A (Reference 10), VEP-NE-2-A (Reference 6), VEP-NE-3-A (Reference 5), and WCAP-12610-P-A (Reference 11), and is consistent with ITS.

Deletion of the obsolete limits associated with N-1 loop operation (TS 2.1.A.2, TS 2.1.A.3, TS Figure 2.1-2, TS Figure 2.1-3) and fuel densification (TS Figure 2.1-4) is acceptable since these limits no longer represent limiting conditions for operation and are not required to be in the TS.

4.2 Determination of No Significant Hazards Consideration

Virginia Electric and Power Company (Dominion) proposes changes to the Surry Power Station Units 1 and 2 TS pursuant to 10 CFR 50.90. The proposed changes add Topical Reports DOM-NAF-2-A and WCAP-8745-P-A to the TS 6.2.C list of NRC-approved methodologies for determining core operating limits. These changes also propose to incorporate TSTF 339-A Revision 2 into the Surry TS. TSTF-339-A relocates the cycle-specific parameter limits, Overtemperature ΔT and Overpower ΔT setpoints, and DNB parameters to the Core Operating Limits Report (COLR). In addition to the changes listed above, other additions and deletions to the plant TS are also included for consistency with the Improved Technical Specifications format, where practical, and to delete requirements that are obsolete. These changes are proposed to enhance the completeness of the Surry TS, to achieve consistency with NUREG-1431 Revision 3 and to maintain in house safety analysis capabilities.

VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs will be used to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores, using the Deterministic Design Limits (DDLs) documented in Appendix B of the DOM-NAF-2-A Fleet Report and the Statistical Design Limit (SDL).

In accordance with the criteria set forth in 10 CFR 50.92, Dominion has evaluated the proposed TS change and determined that the change does not represent a significant hazards consideration. The following is provided in support of this conclusion:

1. <u>Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   Response: No.

   Approval of the proposed changes will allow Dominion to use the VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores, using the DDLs documented in Appendix B of the DOM-NAF-2-A Fleet Report and the SDL. Neither the code/correlation pair nor the Statistical Departure from Nucleate

Boiling Ratio (DNBR) Evaluation Methodology make any contribution to the potential accident initiators and thus cannot increase the probability of any accident. Further, since both the deterministic and statistical DNBR limits meet the required design basis of avoiding Departure from Nucleate Boiling (DNB) with 95% probability at a 95% confidence level, the use of the new code/correlation and the Statistical DNBR Evaluation Methodology do not increase the potential consequences of any accident. Finally, the full core DNB design limit provides increased assurance that the consequences of a postulated accident which includes radioactive release would be minimized because the overall number of rods in DNB would not exceed the 0.1% level. The pertinent evaluations to be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable have been performed and determined to be acceptable. The use of a different code/correlation pair will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. The use of the VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores will not result in a measurable impact on normal operating plant releases and will not increase the predicted radiological consequences of accidents postulated in the UFSAR.

The remaining proposed changes are being made to enhance the completeness of the Surry TS and to achieve consistency with NUREG-1431 Rev. 3. The proposed changes do not add or modify any plant systems, structures or components (SSCs). The proposed changes to relocate TS parameters to the COLR are programmatic and administrative in nature. These changes do not physically alter safety-related systems nor affect the way in which safety-related systems perform their functions. Additional Safety Limits on the DNB design basis and peak fuel centerline temperature are being imposed in TS 2.1, "Safety Limit, Reactor Core," and the Reactor Core Safety Limits figure is being relocated to the COLR. The additional Safety Limits are consistent with the values stated in the UFSAR and those being proposed herein. The proposed changes do not, by themselves, alter any of the relocated parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. TS 6.2.C continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies and that applicable limits of the safety analyses are met. Deletion of the obsolete limits associated with N-1 loop operation (TS 2.1.A.2, TS 2.1.A.3, TS Figure 2.1-2, TS Figure 2.1-3) and fuel densification (TS Figure 2.1-4) is acceptable since these limits no longer represent limiting conditions for operation and are not required to be in the Technical Specifications.

Thus, the proposed changes do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

Response: No.

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed).

The use of VIPRE-D and its applicable fuel design limits for DNBR does not impact any of the applicable design criteria and all pertinent licensing basis criteria will continue to be met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Setpoint safety analysis evaluations have demonstrated that the use of VIPRE-D is acceptable. Design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the VIPRE-D code/correlation or the Statistical DNBR Evaluation Methodology does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

The proposed change adds a new surveillance requirement of RCS Total Flow Rate and requests the addition of an already approved method for determining plant operating limits. The proposed change does not adversely affect accident initiators or precursors, nor does it alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits.

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

3. <u>Does this change involve a significant reduction in a margin of safety?</u>

Response: No.

The proposed changes to relocate TS parameters to the COLR are programmatic and administrative in nature. Additional Safety Limits on the DNB design basis and peak fuel centerline temperature are being imposed in TS 2.1, "Safety Limit, Reactor Core," and the Reactor Core Safety Limits figure is being relocated to the COLR. The additional Safety Limits are consistent with the values stated in the UFSAR and those being proposed herein.

Approval of the proposed changes will allow Dominion to use the VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse 15x15 Upgrade fuel in Surry cores, using the DDLs documented in Appendix B of the DOM-NAF-2-A Fleet Report and the SDL documented herein. The SDL has been developed in accordance with the Statistical DNBR Evaluation Methodology. The DNBR limits meet the design basis of avoiding DNB with 95% probability at a 95% confidence level. The use of the VIPRE-D/WRB-1 code/correlation provides the same margin to safety as the current code/correlation COBRA/WRB-1 used at Surry.

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

Based on the above, Dominion concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 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 implementation of the proposed TS change, (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 ASSESSMENT

The proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(10) as follows:

(i) The proposed change involves no significant hazards consideration.

As described in Section 4.2 above, the proposed change involves no significant hazards consideration.

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

The proposed change does not involve the installation of any new equipment or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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

The proposed change does not involve physical plant changes or introduce any new modes of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Dominion concludes that, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Volume 1, Rev. 3.0, dated June 2004.
2. Industry/TSTF Standard Technical Specification Change Travel, TSTF-339-A, Revision 2, "Relocate TS Parameters to the COLR Consistent with WCAP-14483," May 26, 2000.
3. WCAP-14483-A, Revision 0, "Generic Methodology for Expanded Core Operating Limits Report," January 1999.
4. Topical Report, VEP-FRD-33-A, "Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT Computer Code," October 1983.
5. Topical Report, VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," November 1986.
6. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.
7. Letter from W. L. Stewart (Vepco) to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed Technical Specification Changes, FΔh Increase/Statistical DNBR Methodology," dated July 8, 1991 (Dominion Serial No. 91-374).
8. Letter from B. C. Buckley (USNRC) to W. L. Stewart (Vepco), "Surry Units 1 and 2 – Issuance of Amendments Re: F Delta h Limit and Statistical DNBR Methodology (TAC NOS. M81271 and M82168)," dated June 1, 1992 (Dominion Serial No. 92-405).
9. Letter from W. L. Stewart (Vepco) to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Technical Clarification of NRC Safety Evaluation Report Addressing FΔh Increase and Statistical DNBR Methodology," dated August 13, 1992 (Dominion Serial No. 92-405).
10. Topical Report, DOM-NAF-2-A, Rev. 0.1-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," July 2009.
11. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."
12. Letter from R. C. DeYoung (US Atomic Energy Commission) to S. Ragone (Vepco), "Change No. 9," dated August 9, 1973.
13. Letter from K. R. Goller (US Atomic Energy Commission) to S. Ragone (Vepco), "Change No. 15," dated April 19, 1974.

14. Letter from L. N. Hartz (Vepco) to USNRC, 'Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specification Changes, Correction to Equation for OTΔT Function Allowable Value," dated July 14, 2005 (Dominion Serial No. 05-420).

15. Letter from S. R. Monarque (USNRC) to D. A. Christian (Vepco), "North Anna Power Station, Units 1 and 2 – Issuance of Amendments on Correction to Equation for Over-Temperature Delta T (OTΔT) Function Allowable Value (TAC NOS. MC7612 and MC7613)," dated October 25, 2005 (Dominion Serial No. 05-755).

16. Letter from USNRC to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 3, 1988.

17. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

18. Topical Report, VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.

19. Letter from J. A. Price (Dominion) to USNRC, "Dominion Energy Kewaunee, INC., Dominion Nuclear Connecticut, Inc., Virginia Electric and Power Company, Kewaunee Power Station, Millstone Power Station Units 2 and 3, North Anna Power Station Units 1 and 2; Surry Power Station Units 1 and 2, Removal of Mixing Vane Grid Spacing Restriction in Appendix B to Fleet Report DOM-NAF-2 – Evaluation of 4x4 DNB Test of 15 x 15 VANTAGE+ with IFMs using VIPRE-D/WRB-1," SN 09-528, August 28, 2009.

20. Letter from J. A. Gresham (Westinghouse) to US NRC Document Control Desk, "Fuel Criterion Evaluation Process (FCEP) Notification of the 15x15 Upgrade Design (Proprietary/Non-Proprietary)," ADAMS Accession No. ML040430427 (Westinghouse # LTR-NRC-04-8), February 6, 2004.

21. Letter from K. D. Young (Union Electric Company) to USNRC, "Docket Number 50-483, Callaway Plant, Union Electric Co., Application for Amendment to Facility Operating License NPF-30, Relocation of Technical Specification Cycle-Specific, Parameters to the Core Operating Limits Report, Adoption of Industry Travelers TSTF-339 and TSTF-363," dated August 17, 2006.

22. Letter from J. Donohew (USNRC) to C. D. Naslund (Union Electric Company), Callaway Plant, Unit 1 - Issuance Of Amendment Re: Relocation of Cycle-Specific Parameter Limits to the Core Operating Limits Report (TAC NO. MD2873)," April 2, 2007.

23. Letter from S. Monarque (USNRC) to D. A. Christian (VEPCO), "North Anna Power Station, Units 1 and 2 – Issuance of Amendments Re: Conversion to Improved Technical Specifications (TAC NOS. MB0799 and MB0800)," April 5, 2002 (Dominion Serial No. 02-275).

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS PAGES (MARK-UP)

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

THERMAL POWER

Applies to the limiting combinations of ~~thermal power~~, Reactor Coolant System pressure, coolant temperature and coolant flow when a reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

THERMAL POWER | pressurizer | Reactor Coolant System (RCS) highest loop average temperature

A. The combination of reactor ~~thermal power~~ level, ~~coolant~~ pressure, and ~~coolant temperature~~ shall not:

specified in the CORE OPERATING LIMITS REPORT

1. Exceed the limits ~~shown in TS Figure 2.1-1~~ when full flow from three reactor coolant pumps ~~exist.~~ exists, and the following Safety Limits shall not be exceeded:

~~2. Exceed the limits shown in TS Figure 2.1-2 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are open.~~

~~3. Exceed the limits shown in TS Figure 2.1-3 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are closed.~~

Insert 1

Insert 1

a. The design limit for departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.27 for transients analyzed using the Statistical DNBR Evaluation Methodology and the WRB-1 DNB correlation. For transients analyzed using the deterministic methodology, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit (≥ 1.17 for WRB-1, ≥ 1.30 for W-3).

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

THERMAL POWER

2. ~~4.~~ The reactor ~~thermal power~~ level shall not exceed 118% of rated power.

B. In the event the Safety Limit is violated, the facility shall be placed in at least HOT SHUTDOWN within 1 hour. The safety limit is exceeded if the combination of ~~Reactor Coolant System average temperature and thermal power~~ RCS highest loop average temperature and THERMAL POWER level is at any time above the appropriate pressure line ~~in TS Figures 2.1-1, 2.1-2 or 2.1-3;~~ as specified in the CORE OPERATING LIMITS REPORT; or the core ~~thermal power~~ THERMAL POWER exceeds 118% of the rated power.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, DNB has been correlated to thermal power, reactor coolant temperature and reactor coolant pressure which are observable parameters. This correlation 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 DNB heat flux at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB basis is as follows: there must be at least a 95% probability with 95% confidence 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 based on the entire applicable experimental data set to meet this statistical criterion.(1)

Insert 2

~~The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the calculated DNBR is not less than the design DNBR limit or the average enthalpy at the exit of the vessel is equal to the saturation value~~. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon the design DNBR limit alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The effects of rod bowing are also considered in the DNBR analyses.

Insert 3

Insert 2

The figure provided in the CORE OPERATING LIMITS REPORT shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel 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.

Insert 3

The reactor core Safety Limits 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 Safety Limits 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 Safety Limits will be satisfied during steady state operation, normal operational transients, and AOOs.

~~TS Figure 2.1-1 is based on a 1.55 cosine axial flux shape, and a statistical treatment of key DNBR analysis parameter uncertainties including an enthalpy rise hot channel factor which follows the following functional form: FΔH(N) = 1.56 [1 + 0.3 (1-P)] where P is the fraction of RATED POWER. The limits include margin to accommodate rod bowing.[1] TS Figures 2.1-2 and 2.1-3 are based on an FΔH(N) of 1.55, a deterministic treatment of key DNB analysis parameter uncertainties, and include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor. The FΔH(N) limit presented in the unit- and reload-specific CORE OPERATING LIMITS REPORT is confirmed for each reload to be accommodated by the Reactor Core Safety Limits.~~

~~These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies~~

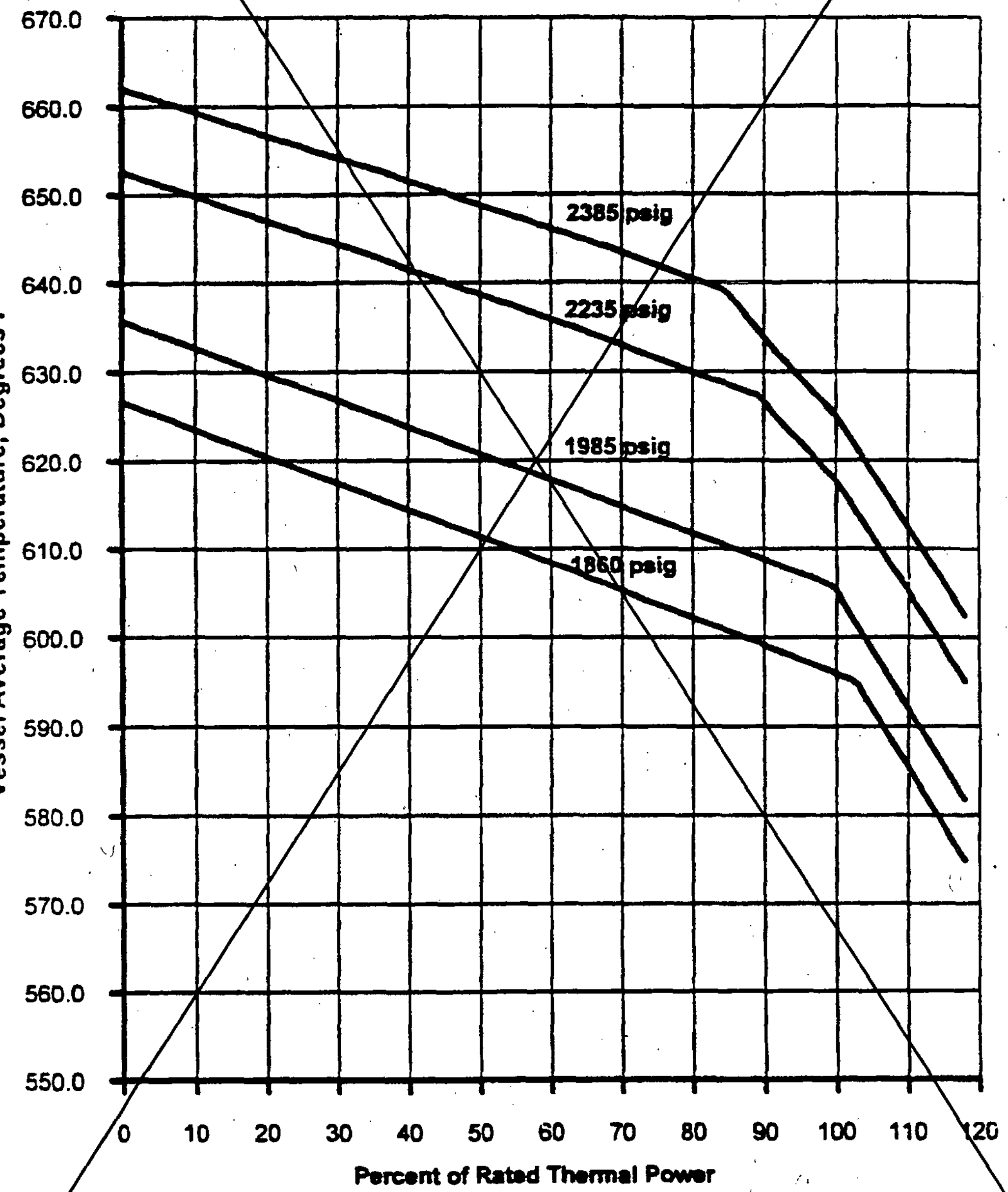
~~fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits as specified in the CORE OPERATION LIMITS REPORT ensure that the DNBR is always greater at partial power than at full power.~~

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 573.0°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

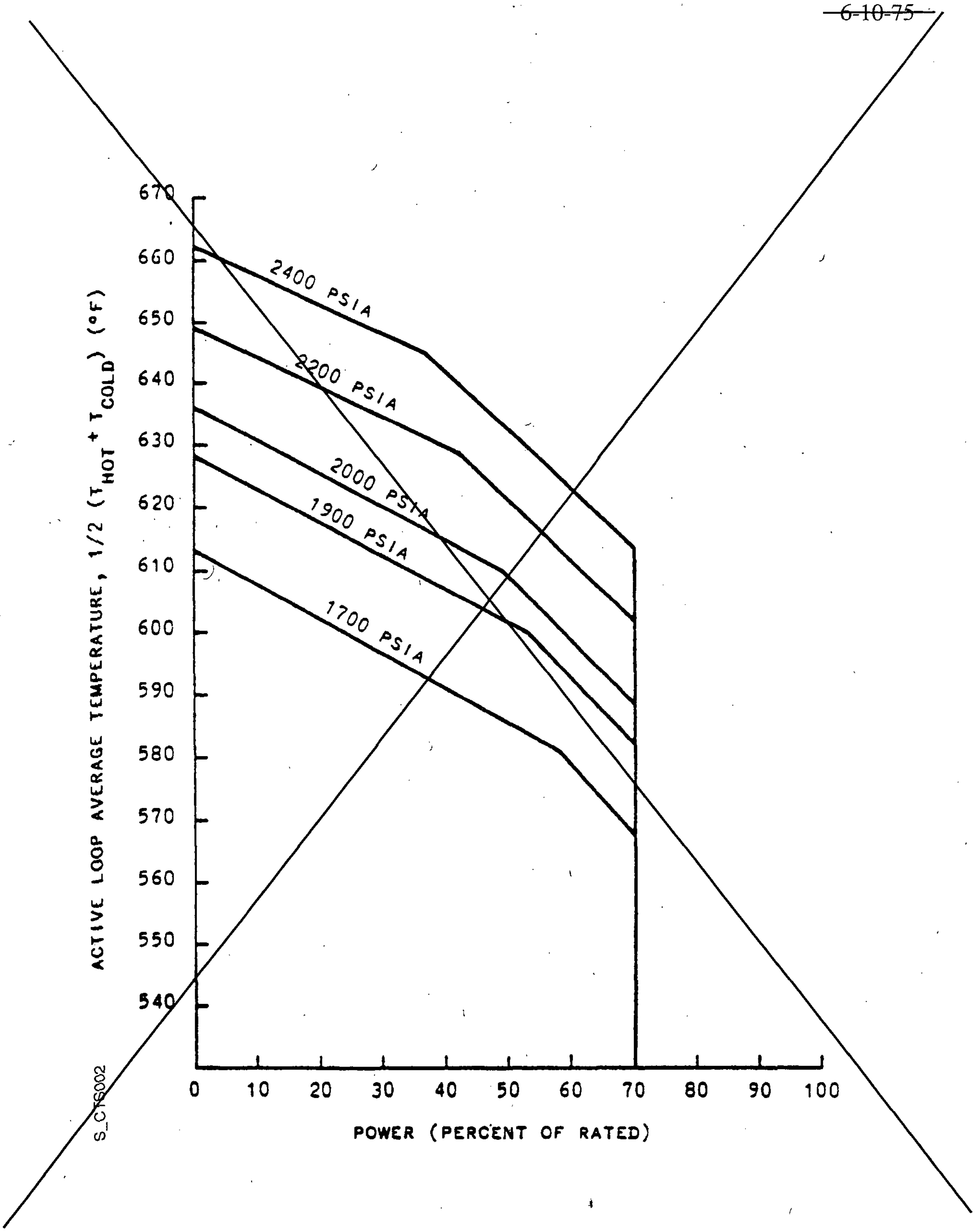
For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also

~~TS FIGURE 2.1-1~~

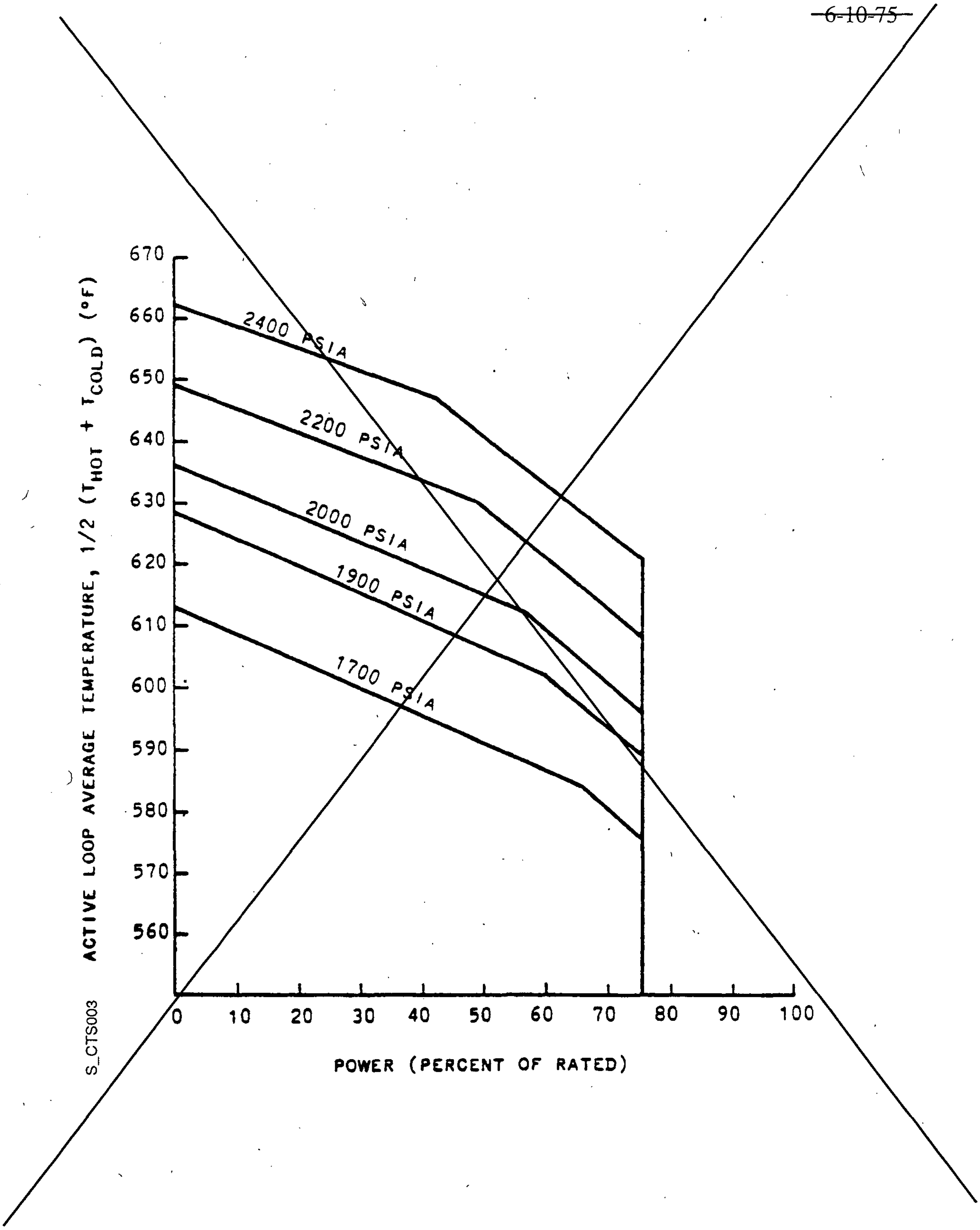
~~REACTOR CORE THERMAL AND~~
~~HYDRAULIC SAFETY LIMITS~~
~~THREE LOOP OPERATION, 100% FLOW~~



S_CTS001

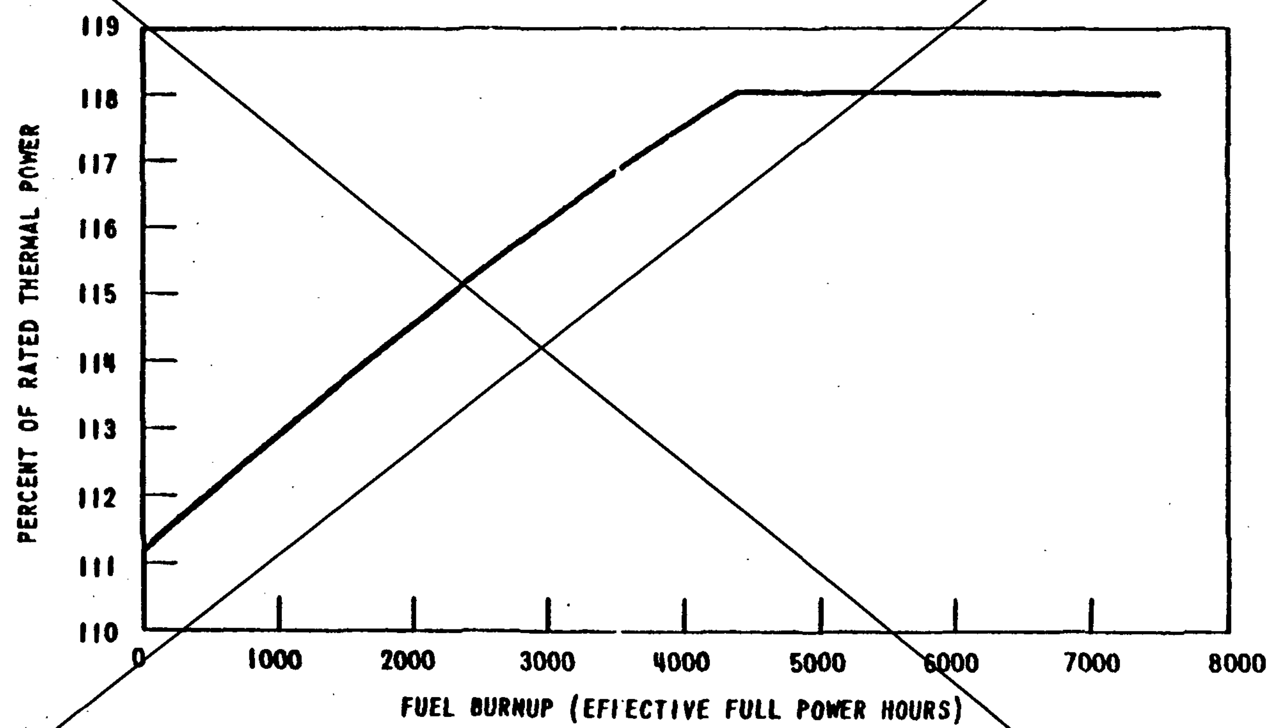


~~FIGURE 2.1-2~~ ~~REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES OPEN~~



~~FIGURE 2.1-3~~ ~~REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES CLOSED~~

~~CHANGE NO. 9~~



~~CHANGE NO. 9~~ S_CTS004

~~Figure 2.1-4 Thermal Overpower Limit~~

(b) High pressurizer pressure - ≤ 2380 psig.

(c) Low pressurizer pressure - ≥ 1875 psig.

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1 + t_1 s}{1 + t_2 s} \right) (T - T') + K_3 (P - P') - f(\Delta I) \right]$$

~~where~~ Insert 4

~~ΔT_0 = Indicated ΔT at rated thermal power, °F~~

~~T = Average coolant temperature, °F~~

~~T' = 573.0°F~~

~~P = Pressurizer pressure, psig~~

~~P' = 2235 psig~~

~~K_1 = 1.135~~

~~K_2 = 0.01072~~

~~K_3 = 0.000566~~

~~$\Delta I = q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power~~

~~$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1~~

~~$t_1 \geq 29.7$ seconds~~

~~$t_2 \leq 4.4$ seconds~~

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of the ΔT span. (Note that 2.0% of the ΔT span is equal to 3.0% ΔT Power.)

(e) Overpower ΔT

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \left(\frac{t_3 s}{1 + t_3 s} \right) T - K_6 (T - T') - f(\Delta I) \right]$$

Insert 4

Where:

ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RATED POWER, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature (T_{avg}), °F.

T' is the nominal T_{avg} at RATED POWER, $\leq$[*] °F.

P is the measured pressurizer pressure, psig.

P' is the nominal RCS operating pressure, $\geq$[*] psig.

$K_1 \leq$[*] $K_2 \geq$[*] / °F $K_3 \geq$[*] / psig

$t_1 \geq$[*] sec $t_2 \leq$[*] sec

$f(\Delta I)$ = [*]{[*] – ($q_t - q_b$)} when $q_t - q_b$ < [*]% RATED POWER

0 when [*]% RATED POWER $\leq q_t - q_b \leq$[*]% RATED POWER

[*]{($q_t - q_b$)-[*]} when $q_t - q_b$ > [*]% RATED POWER

Where q_t and q_b are percent RATED POWER in the upper and lower halves of the core, respectively, and $q_t + q_b$ of the total THERMAL POWER in percent RATED POWER.

The values denoted with [*] are specified in the CORE OPERATING LIMITS REPORT.

~~where~~ Insert 5

~~ΔT_0 = Indicated ΔT at rated thermal power, °F~~

~~T = Average coolant temperature, °F~~

~~T' = Average coolant temperature measured at nominal conditions and rated power, °F~~

~~K_4 = A constant = 1.089~~

~~K_5 = 0 for decreasing average temperature~~

~~A constant, for increasing average temperature 0.02/°F~~

~~K_6 = 0 for T ≤ T'~~

~~= 0.001086 for T > T'~~

~~f(ΔI) as defined in (d) above~~

~~τ_3 ≥ 9.0 seconds~~

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of the ΔT span. (Note that 2.0% of the ΔT span is equal to 3.0% of ΔT Power.)

(f) Low reactor coolant loop flow - ≥ 91% of normal indicated loop flow as measured at elbow taps in each loop

(g) Low reactor coolant pump motor frequency - ≥ 57.5 Hz

(h) Reactor coolant pump under voltage - ≥ 70% of normal voltage

3. Other reactor trip settings

(a) High pressurizer water level - ≤ 89.12% of span

(b) Low-low steam generator water level - ≥ 16% of narrow range instrument span

(c) Low steam generator water level - ≥ 19% of narrow range instrument span in coincidence with steam/feedwater mismatch flow - ≤ 1.0×10^6 lbs/hr

(d) Turbine trip

(e) Safety injection - Trip settings for Safety Injection are detailed in TS Section 3.7.

Insert 5

Where:

ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RATED POWER, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature (T_{avg}), °F.

T' is the nominal T_{avg} at RATED POWER, $\leq$ [*] °F.

$K_4 \leq$ [*]	$K_5 \geq$ [*] / °F for decreasing T_{avg}	$K_6 \geq$ [*] / °F when $T > T'$
	[*] / °F for increasing T_{avg}	[*] / °F when $T \leq T'$

$t_3 \geq$ [*] sec

$f(\Delta I)$ = [*]

The values denoted with [*] are specified in the CORE OPERATING LIMITS REPORT.

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3 seconds), and pressure is within the range between high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,(2) is always below the core safety limit as ~~shown on TS Figure 2.1-1.~~ specified in the CORE OPERATING LIMITS REPORT. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor limit is automatically reduced.(4)(5)

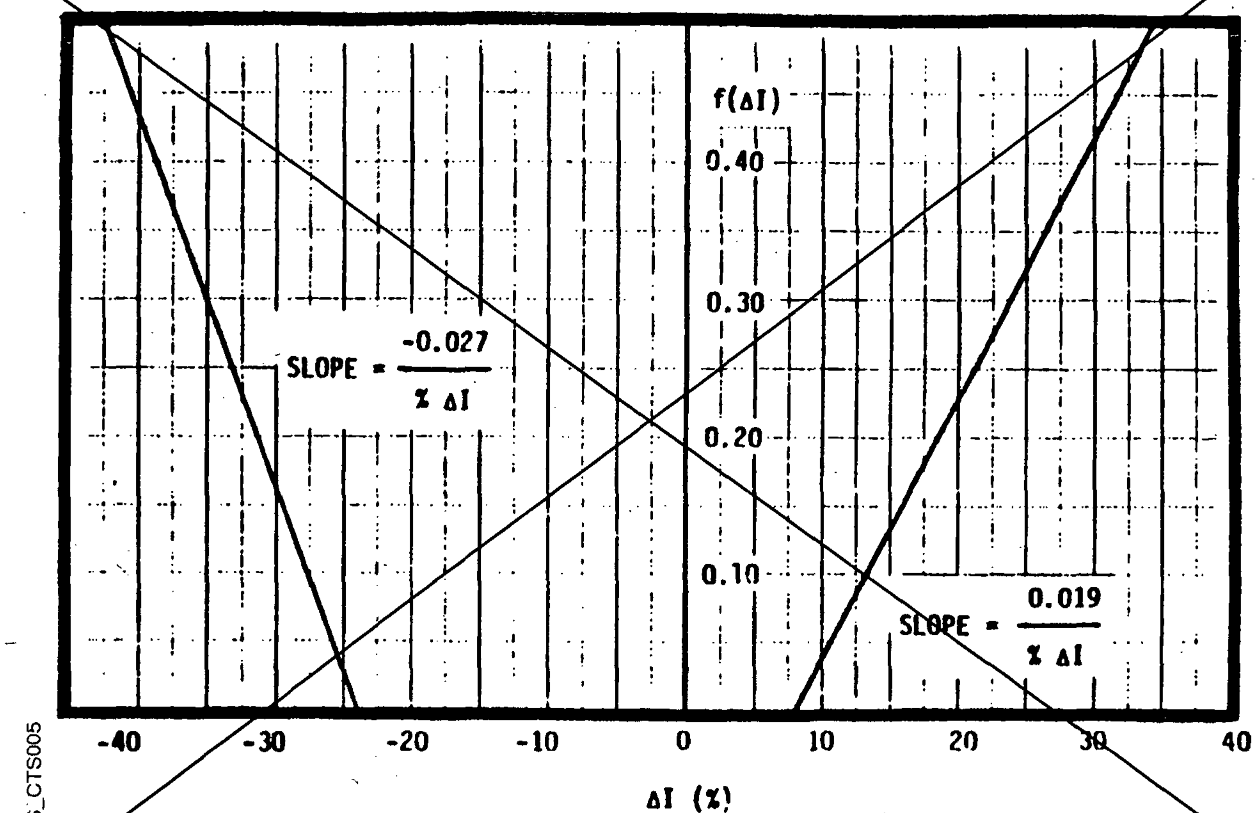
The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification on core safety limits and to apply to 100% of design flow. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as ~~shown in Figures 2.1-1 through 2.1-3.~~ specified in the CORE OPERATING LIMITS REPORT. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The overpower protection system set points include the effects of fuel densification.

~~In order to operate with a reactor coolant loop out of service (two-loop operation) and with the stop valves of the inactive loop either open or closed, the overtemperature ΔT trip setpoint calculation has to be modified by the adjustment of the variable K_1. This adjustment, based on limits of two-loop operation, provides sufficient margin to DNB for the aforementioned transients during two loop operation. The required adjustment and subsequent mandatory calibrations are made in the protective system racks by qualified technicians* in the same manner as adjustments before initial startup and normal calibrations for three-loop operation.~~

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed Section 7 and specified in Section 14.2.2 of the FSAR and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.(2)

Refer to Technical Report EE-0116 for justification of the dynamic limits (time constants) for the Overtemperature ΔT and Overpower ΔT Reactor Trip functions.

~~* As used here, a qualified technician means a technician who meets the requirements of ANS-3. He shall have a minimum of two years of working experience in his speciality and at least one year of related technical training.~~



~~Figure 2.3-1 OPΔT and OTΔT f(ΔI) Function~~

3. If more than one rod position indicator per group is inoperable, place the control rods under manual control immediately, monitor and record RCS T_{avg} once per hour, verify the position of the control rod assemblies indirectly using the movable incore detectors at least once per 8 hours, and restore inoperable position indicators to OPERABLE status such that a maximum of one position indicator per group is inoperable within 24 hours.

4. If one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position, verify the position of the control rod assemblies indirectly using the movable incore detectors within 4 hours or reduce power to less than 50% of RATED POWER within 8 hours.

5. If one group step demand counter per bank for more than one or more banks is inoperable, verify that all rod position indicators for the affected bank(s) are OPERABLE once per 8 hours and verify that the most withdrawn rod and the least withdrawn rod of the affected bank(s) are less than or equal to 12 steps apart once per 8 hours. Alternatively, reduce power to less than 50% of RATED POWER within 8 hours.

6. If the requirements of Specification 3.12.E.2, 3.12.E.3, 3.12.E.4, or 3.12.E.5 are not satisfied, then the unit shall be placed in HOT SHUTDOWN within 6 hours.

F. DNB Parameters

1. The following DNB related parameters shall be maintained within their limits during POWER OPERATION:

the limit specified in the CORE OPERATING LIMITS REPORT

- Reactor Coolant System T_{avg} ≤ ~~577.0°F~~ Insert
- Pressurizer Pressure ≥ ~~2205 psig~~

and ≥ the limit specified in the CORE OPERATING LIMITS REPORT

- Reactor Coolant System Total Flow Rate ≥ 273,000 gpm Insert

a. The Reactor Coolant System T_{avg} ~~and Pressurizer Pressure~~ shall be verified to

, Pressurizer Pressure, and Reactor Coolant System Total Flow Rate

Insert

be within their limits at least once every 12 hours.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by ~~measurement at least once per refueling cycle.~~

Insert

precision heat balance with the frequency specified in TS Table 4.1-2A

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED POWER within the next ~~4~~ Insert 6 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED POWER per minute or a THERMAL POWER step increase in excess of 10% of RATED POWER.

G. Shutdown Margin

1. Whenever the reactor is subcritical, the shutdown margin shall be within the limits specified in the CORE OPERATING LIMITS REPORT. If the shutdown margin is not within limits, within 15 minutes, initiate boration to restore shutdown margin to within limits.

A 2% QUADRANT POWER TILT allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod assembly and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rod assemblies producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The QPTR limit must be maintained during power operation with THERMAL POWER > 50% of RATED POWER to prevent core power distributions from exceeding the design limits.

Applicability during power operation $\leq$ 50% RATED POWER or when shut down is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RATED POWER or lower.

The limits of 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 UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. ~~The measurement of the Reactor Coolant System Total Flow Rate once per refueling cycle is adequate to detect flow degradation.~~

Insert 6

Insert 6

The 12 hour surveillance of RCS total flow rate, by installed flow instrumentation, is sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions. Measurement of RCS total flow rate by performance of a precision calorimetric heat balance specified in TS Table 4.1-2A allows for the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The refueling water storage tank is sampled weekly for Cl^- and/or F^- contaminations. Weekly sampling is adequate to detect any inleakage of contaminated water.

Control Room Bottled Air System

The control room bottled air system is required to establish a positive differential pressure in the control room for one hour following a design basis accident. The ability of the system to meet this requirement is verified by: 1) checking air bottle pressurization, 2) demonstrating the capability to pressurize the control room pressure boundary, 3) functionally testing the pressure control valve(s), and 4) functionally testing the manual and automatic actuation capability. The test requirements and frequency are specified in Table 4.1-2A.

~~Pressurizer PORV, PORV Block Valve, and PORV Backup Air Supply~~

~~The safety-related, seismic PORV backup air supply is relied upon for two functions - mitigation of a design basis steam generator tube rupture accident and low temperature overpressure protection (LTOP) of the reactor vessel during startup and shutdown. The surveillance criteria are based upon the more limiting requirements for the backup air supply (i.e. more PORV cycles potentially required to perform the mitigation function), which are associated with the LTOP function.~~

~~The PORV backup air supply system is provided with a calibrated alarm for low air pressure. The alarm is located in the control room. Failures such as regulator drift and air leaks which result in low pressure can be easily recognized by alarm or annunciator action. A periodic quarterly verification of air pressure against the surveillance limit supplements this type of built-in surveillance. Based on experience in operation, the minimum checking frequencies set forth are deemed adequate.~~

Insert 7

Insert 7

RCS Flow

The frequency of 18 months for RCS flow surveillance reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of the flow resistance. This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of $\geq$ 90% of RATED POWER. The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. The surveillance shall be performed within 7 days after reaching 90% of RATED POWER.

TABLE 4.1-2A(CONTINUED)
MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
19.	Primary Coolant System	Functional	1. Periodic leakage testing(a)(b) on each valve listed in Specification 3.1.C.5.a shall be accomplished prior to entering POWER OPERATION after every time the plant is placed in COLD SHUTDOWN for refueling, after each time the plant is placed in COLD SHUTDOWN for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
20.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval(c)	
21.	Deleted			
22.	RCS Flow	Flow ≥ 273,000 gpm	Once per 18 months $\boxed{\text{(d)}}$ Insert	14
23.	Deleted			

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

* See Specification 4.1.D.

Insert 8

Insert

and ≥ the limit as specified in the CORE OPERATING LIMITS REPORT

Insert 8

(d) Not required to be performed until 7 days after $\geq$ 90% RATED POWER.

6.2 GENERAL NOTIFICATION AND REPORTING REQUIREMENTS

Specification

A. The following action shall be taken for Reportable Events:

A report shall be submitted pursuant to the requirements of Section 50.73 to 10 CFR.

B. Immediate notifications shall be made in accordance with Section 50.72 to 10 CFR.

C. CORE OPERATING LIMITS REPORT

Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. Parameter limits for the following Technical Specifications are defined in the CORE OPERATING LIMITS REPORT:

1. TS 3.1.E - Moderator Temperature Coefficient
2. TS 3.12.A.2 and TS 3.12.A.3 - Control Bank Insertion Limits
3. TS 3.12.B.1 and TS 3.12.B.2 - Power Distribution Limits

$\boxed{\text{Insert 9}}$

Insert 9

4. TS 3.12.F – DNB Parameters
5. TS 2.1 – Safety Limit, Reactor Core
6. TS 2.3.A.2.d – Overtemperature ΔT
7. TS 2.3.A.2.e – Overpower ΔT
8. TS Table 4.1-2A – Minimum Frequency for Equipment Tests: Item 22 – RCS Flow

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The ~~COLR~~ CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the ~~COLR~~ CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements). The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS 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 thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REFERENCES

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology"

2 ~~2a.~~ WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary).

3 ~~2b.~~ WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," (W Proprietary)

4 ~~2c.~~ WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," (W Proprietary)

5 ~~2d.~~ WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary)

6 ~~3a.~~ VEP-NE-2-A, "Statistical DNBR Evaluation Methodology"

7 ~~3b.~~ VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code"

Insert

8. DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix B, "Qualification of the Westinghouse WRB-1 CHF Correlation in the Dominion VIPRE-D Computer Code."

9. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS PAGES (TYPED)

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of THERMAL POWER, Reactor Coolant System pressure, coolant temperature and coolant flow when a reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

A. The combination of reactor THERMAL POWER level, pressurizer pressure, and Reactor Coolant System (RCS) highest loop average temperature shall not:

1. Exceed the limits specified in the CORE OPERATING LIMITS REPORTS when full flow from three reactor coolant pumps exists, and the following Safety Limits shall not be exceeded:

   a. The design limit for departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.27 for transients analyzed using the Statistical DNBR Evaluation Methodology and the WRB-1 DNB correlation. For transients analyzed using the deterministic methodology, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit ($\geq$ 1.17 for WRB-1, $\geq$ 1.30 for W-3).

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

2. The reactor THERMAL POWER level shall not exceed 118% of rated power.

B. In the event the Safety Limit is violated, the facility shall be placed in at least HOT SHUTDOWN within 1 hour. The safety limit is exceeded if the combination of RCS highest loop average temperature and THERMAL POWER level is at any time above the appropriate pressure line as specified in the CORE OPERATING LIMITS REPORT; or the core THERMAL POWER exceeds 118% of the rated power.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, DNB has been correlated to thermal power, reactor coolant temperature and reactor coolant pressure which are observable parameters. This correlation 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 DNB heat flux at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB basis is as follows: there must be at least a 95% probability with 95% confidence 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 based on the entire applicable experimental data set to meet this statistical criterion.(1)

The figure provided in the CORE OPERATING LIMITS REPORT shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel 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 area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon the design DNBR limit alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The effects of rod bowing are also considered in the DNBR analyses.

The reactor core Safety Limits 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 Safety Limits 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 the variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core Safety Limits will be satisfied during steady state operation, normal operational transients, and AOOs.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit[(3)] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 573.0°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also ensures that at least 99.9% of the core avoids the onset of DNB when the limiting rod is at the DNBR limit.

The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

References

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1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

(b) High pressurizer pressure - $\leq$ 2380 psig.

(c) Low pressurizer pressure - $\geq$ 1875 psig.

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_0\left[K_1 - K_2\left(\frac{1 + t_1 s}{1 + t_2 s}\right)(T - T') + K_3(P - P') - f(\Delta I)\right]$$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RATED POWER, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature (T_{avg}), °F.
T' is the nominal T_{avg} at RATED POWER, $\leq$ [*] °F.
P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, $\geq$ [*]psig.
$K_1 \leq$ [*] $K_2 \geq$ [*]/°F $K_3 \geq$ [*]/psig
$t_1 \geq$ [*] sec $t_2 \leq$ [*] sec
$f(\Delta I)$ = [*] {[*] - (q_t - q_b)}   when q_t - q_b < [*]% RATED POWER
0 when [*]% RATED POWER $\leq q_t - q_b \leq$ [*]% RATED POWER
[*]{(q_t - q_b)-[*]} when q_t - q_b > [*]% RATED POWER

Where q_t and q_b are percent RATED POWER in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RATED POWER.

The values denoted with [*] are specified in the CORE OPERATING LIMITS REPORT.

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of the ΔT span. (Note that 2.0% of the ΔT span is equal to 3.0% ΔT Power.)

(e) Overpower ΔT

$$\Delta T \leq \Delta T_0\left[K_4 - K_5\left(\frac{t_3 s}{1 + t_3 s}\right)T - K_6(T - T') - f(\Delta I)\right]$$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RATED POWER, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature (T_{avg}), °F.
T' is the nominal T_{avg} at RATED POWER, $\leq$ [*] °F.

$K_4 \leq$ [*]	$K_5 \geq$ [*]/°F for decreasing T_{avg}	$K_6 \geq$ [*]/°F when T > T'
	[*] / °F for decreasing T_{avg}	[*]/°F when T $\leq$ T'

$t_3 \geq$ [*] sec
$f(\Delta I)$ = [*]

The values denoted with [*] are specified in the CORE OPERATING LIMITS REPORT.

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of the ΔT span. (Note that 2.0% of the ΔT span is equal to 3.0% of ΔT Power.)

(f) Low reactor coolant loop flow - $\geq$ 91% of normal indicated loop flow as measured at elbow taps in each loop

(g) Low reactor coolant pump motor frequency - $\geq$ 57.5 Hz

(h) Reactor coolant pump under voltage - $\geq$ 70% of normal voltage

3. Other reactor trip settings

(a) High pressurizer water level - $\leq$ 89.12% of span

(b) Low-low steam generator water level - $\geq$ 16% of narrow range instrument span

(c) Low steam generator water level - $\geq$ 19% of narrow range instrument span in coincidence with steam/feedwater mismatch flow - $\leq 1.0 \times 10^6$ lbs/hr

(d) Turbine trip

(e) Safety injection - Trip settings for Safety Injection are detailed in TS Section 3.7.

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3 seconds), and pressure is within the range between high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,(2) is always below the core safety limit as specified in the CORE OPERATING LIMITS REPORT. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor limit is automatically reduced.(4)(5)

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification on core safety limits and to apply to 100% of design flow. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as specified in the CORE OPERATING LIMITS REPORT. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The overpower protection system set points include the effects of fuel densification.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed Section 7 and specified in Section 14.2.2 of the FSAR and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.(2)

Refer to Technical Report EE-0116 for justification of the dynamic limits (time constants) for the Overtemperature ΔT and Overpower ΔT Reactor Trip functions.

3. If more than one rod position indicator per group is inoperable, place the control rods under manual control immediately, monitor and record RCS T_{avg} once per hour, verify the position of the control rod assemblies indirectly using the movable incore detectors at least once per 8 hours, and restore inoperable position indicators to OPERABLE status such that a maximum of one position indicator per group is inoperable within 24 hours.

4. If one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position, verify the position of the control rod assemblies indirectly using the movable incore detectors within 4 hours or reduce power to less than 50% of RATED POWER within 8 hours.

5. If one group step demand counter per bank for more than one or more banks is inoperable, verify that all rod position indicators for the affected bank(s) are OPERABLE once per 8 hours and verify that the most withdrawn rod and the least withdrawn rod of the affected bank(s) are less than or equal to 12 steps apart once per 8 hours. Alternatively, reduce power to less than 50% of RATED POWER within 8 hours.

6. If the requirements of Specification 3.12.E.2, 3.12.E.3, 3.12.E.4, or 3.12.E.5 are not satisfied, then the unit shall be placed in HOT SHUTDOWN within 6 hours.

F. <u>DNB Parameters</u>

1. The following DNB related parameters shall be maintained within their limits during POWER OPERATION:

   - Reactor Coolant System $T_{avg} \leq$ the limit specified in the CORE OPERATING LIMITS REPORT
   - Pressurizer Pressure $\geq$ the limit specified in the CORE OPERATING LIMITS REPORT
   - Reactor Coolant System Total Flow Rate $\geq$ 273,000 gpm and $\geq$ the limit specified in the CORE OPERATING LIMITS REPORT

a. The Reactor Coolant System T_{avg}, Pressurizer Pressure, and Reactor Coolant System Total Flow Rate shall be verified to be within their limits at least once every 12 hours.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by precision heat balance with the frequency specified in TS Table 4.1-2A.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED POWER within the next 6 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED POWER per minute or a THERMAL POWER step increase in excess of 10% of RATED POWER.

G. Shutdown Margin

1. Whenever the reactor is subcritical, the shutdown margin shall be within the limits specified in the CORE OPERATING LIMITS REPORT. If the shutdown margin is not within limits, within 15 minutes, initiate boration to restore shutdown margin to within limits.

A 2% QUADRANT POWER TILT allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod assembly and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rod assemblies producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The QPTR limit must be maintained during power operation with THERMAL POWER > 50% of RATED POWER to prevent core power distributions from exceeding the design limits.

Applicability during power operation ≤ 50% RATED POWER or when shut down is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RATED POWER or lower.

The limits of 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 UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The 12 hour surveillance of RCS total flow rate, by installed flow instrumentation, is sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions. Measurement of RCS total flow rate by performance of a precision calorimetric heat balance specified in TS Table 4.1-2A allows for the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The refueling water storage tank is sampled weekly for Cl^- and/or F^- contaminations. Weekly sampling is adequate to detect any inleakage of contaminated water.

Control Room Bottled Air System

The control room bottled air system is required to establish a positive differential pressure in the control room for one hour following a design basis accident. The ability of the system to meet this requirement is verified by: 1) checking air bottle pressurization, 2) demonstrating the capability to pressurize the control room pressure boundary, 3) functionally testing the pressure control valve(s), and 4) functionally testing the manual and automatic actuation capability. The test requirements and frequency are specified in Table 4.1-2A.

Pressurizer PORV, PORV Block Valve, and PORV Backup Air Supply

The safety-related, seismic PORV backup air supply is relied upon for two functions - mitigation of a design basis steam generator tube rupture accident and low temperature overpressure protection (LTOP) of the reactor vessel during startup and shutdown. The surveillance criteria are based upon the more limiting requirements for the backup air supply (i.e. more PORV cycles potentially required to perform the mitigation function), which are associated with the LTOP function.

The PORV backup air supply system is provided with a calibrated alarm for low air pressure. The alarm is located in the control room. Failures such as regulator drift and air leaks which result in low pressure can be easily recognized by alarm or annunciator action. A periodic quarterly verification of air pressure against the surveillance limit supplements this type of built-in surveillance. Based on experience in operation, the minimum checking frequencies set forth are deemed adequate.

RCS Flow

The frequency of 18 months for RCS flow surveillance reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of the flow resistance. This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of $\geq$ 90% of RATED POWER. The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. The surveillance shall be performed within 7 days after reaching 90% of RATED POWER.

TABLE 4.1-2A(CONTINUED)
MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
19.	Primary Coolant System	Functional	1. Periodic leakage testing(a)(b) on each valve listed in Specification 3.1.C.5.a shall be accomplished prior to entering POWER OPERATION after every time the plant is placed in COLD SHUTDOWN for refueling, after each time the plant is placed in COLD SHUTDOWN for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
20.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval(c)	
21.	Deleted			
22.	RCS Flow	Flow $\geq$ 273,000 gpm and $\geq$ the limit as specified in the CORE OPERATING LIMITS REPORT	Once per 18 months (d)	14
23.	Deleted			

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.
(b) Minimum differential test pressure shall not be below 150 psid.
(c) Refer to Section 4.4 for acceptance criteria.
(d) Not required to be performed until 7 days after $\geq$ 90% RATED POWER.
* See Specification 4.1.D.

6.2 GENERAL NOTIFICATION AND REPORTING REQUIREMENTS

Specification

A. The following action shall be taken for Reportable Events:

A report shall be submitted pursuant to the requirements of Section 50.73 to 10 CFR.

B. Immediate notifications shall be made in accordance with Section 50.72 to 10 CFR.

C. CORE OPERATING LIMITS REPORT

Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. Parameter limits for the following Technical Specifications are defined in the CORE OPERATING LIMITS REPORT:

1. TS 3.1.E - Moderator Temperature Coefficient
2. TS 3.12.A.2 and TS 3.12.A.3 - Control Bank Insertion Limits
3. TS 3.12.B.1 and TS 3.12.B.2 - Power Distribution Limits
4. TS 3.12.F - DNB Parameters
5. TS 2.1 - Safety Limit, Reactor Core
6. TS 2.3.A.2.d - Overtemperature ΔT
7. TS 2.3.A.2.e - Overpower ΔT
8. TS Table 4.1-2A - Minimum Frequency for Equipment Tests: Item 22 - RCS Flow

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements). The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS 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 thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

<u>REFERENCES</u>

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology"
2. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary).
3. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," (W Proprietary)
4. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," (W Proprietary)
5. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary).
6. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology"
7. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code"
8. DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix B, "Qualification of the Westinghouse WRB-l CHF Correlation in the Dominion VIPRE-D Computer Code."
9. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

ATTACHMENT 4

TECHNICAL BASIS FOR ADDING FLEET REPORT DOM-NAF-2-A TO THE LIST OF NRC APPROVED METHODOLOGIES FOR DETERMINING CORE OPERATING PARAMETERS

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

TABLE OF CONTENTS

1. Introduction

This report provides the plant specific application for Surry Power Station (SPS) cores containing 15x15 Upgrade fuel assemblies. Specifically, this report supports the application of U.S. Nuclear Regulatory Commission (USNRC) approved Dominion Fleet Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 1) to SPS. It provides the technical basis and documentation required by the USNRC to evaluate the plant specific application of VEP-NE-2-A methods to SPS. This application employs the VIPRE-D code with the Westinghouse WRB-1 Critical Heat Flux (CHF) correlation (VIPRE-D/WRB-1 code/correlation pair) for the thermal-hydraulic analysis of Westinghouse 15x15 Upgrade fuel assemblies at SPS. In particular, Dominion requests the review and approval of the Statistical Design Limit (SDL) documented herein as per 10 CFR 50.59(c)(2)(vii) it constitutes a Design Basis Limit for Fission Products Barrier (DBLFPB).

Dominion is seeking approval for the inclusion of Fleet Report DOM-NAF-2-A (Reference 2) to the Technical Specification 6.2.C list of USNRC-approved methodologies used to determine core operating limits (i.e., the reference list of Surry Core Operating Limits Report (COLR)). This would allow Dominion the use of the VIPRE-D/WRB-1 code/correlation set to perform licensing calculations for Westinghouse 15x15 Upgrade fuel in Surry's cores, using the deterministic design limits (DDLs) qualified in Appendix B of the DOM-NAF-2-A Fleet Report, and the statistical design limit (SDL) identified herein.

Westinghouse 15x15 Upgrade fuel assemblies include intermediate flow mixer grids (IFMs) which have a grid spacing of 13". There is currently a restriction in Appendix B of DOM-NAF-2-A (Reference 2) that states that the fuel must have a mixing vane grid spacing greater than 13". Dominion has performed in-house validation of Westinghouse test section data to be able to remove this restriction. A licensing package was submitted to the NRC for their review and approval by letter dated August 28, 2009 (Serial No. 09-528 (Reference 11)) for the removal of the grid spacing restriction. The approval of the grid spacing restriction removal will be needed to apply VIPRE-D/WRB-1 and to implement the Statistical DNRB Evaluation Methodology for 15x15 Upgrade fuel.

With these approvals, Dominion will be licensed to perform in-house DNB analyses for the intended uses described in DOM-NAF-2-A to support Surry Power Station, Units 1 and 2 with 15x15 Upgrade fuel.

2. Background

Dominion (Virginia Power) is purchasing fuel assemblies from Westinghouse for use at Surry Power Station, Units 1 and 2. These assemblies will be inserted in Units 1 and 2, commencing with Cycles 25 and 24, respectively. The fuel assemblies are designated as the 15x15 Upgrade (Reference 6), which have three IFMs. These assemblies are a replacement for the resident fuel product, which is a Westinghouse 15x15 VANTAGE+ fuel product, also referred to as Surry Improved Fuel (SIF), without IFMs.

The 15x15 Upgrade fuel assemblies will be used at Surry after a Measurement Uncertainty Recapture (MUR) uprate has been implemented. The statistical DNBR evaluation performed herein will be valid for the current rated thermal power of 2546 MWt and an MUR uprate condition of up to 1.7% (i.e., 2589 MWt).

The computer code VIPRE (Versatile Internals and Components Program for Reactors - EPRI) was developed for the Electric Power Research Institute (EPRI) by Battelle Pacific Northwest Laboratories to perform detailed thermal-hydraulic analyses to predict CHF and DNBR of reactor cores. Topical Report VIPRE-01 was approved by the U.S. Nuclear Regulatory Commission (USNRC) in References 15 and 16 for referencing in licensing applications. VIPRE-D is the Dominion version of the VIPRE computer code based upon VIPRE-01, MOD-02.1. VIPRE-D was developed to fit the specific needs of Dominion's nuclear plants and fuel products by adding vendor specific CHF correlations and customizing its input and output. However, Dominion has not made any modifications to the NRC-approved constitutive models and algorithms contained in VIPRE-01.

Dominion's approved Fleet Report DOM-NAF-2-A (including Appendix B, which describes the verification and qualification of the WRB-1 CHF correlation) (Reference 2) has been reviewed and approved by the USNRC. DOM-NAF-2-A provided the necessary documentation to describe Dominion's use of the VIPRE-D code, including modeling and qualification for Pressurized Water Reactors (PWR) thermal-hydraulic design and demonstrated that the VIPRE-D methodology is appropriate for PWR licensing applications. Appendix B qualified the WRB-1 CHF correlation with the VIPRE-D code and listed the deterministic code/correlation DNBR limits. In addition, Section 2.1 of DOM-NAF-2-A listed the information to be provided to the USNRC by Dominion for the review and approval of any plant specific application of the VIPRE-D code:

1) Technical Specifications change request to add DOM-NAF-2-A and relevant Appendixes to the plant's COLR list.

2) Statistical Design Limit(s) for the relevant code/correlation(s) (Section 4.1)

3) Any technical specification changes related to OTΔT, OPΔT, FΔI or other reactor protection function, as well as revised Reactor Core Safety Limits (Section 4.5).

4) List of UFSAR transients for which the code/correlations will be applied (Section 3.9).

This report provides the USNRC the necessary documentation (items 1 through 4 above) to review and approve the application of the VIPRE-D methodology with the WRB-1 CHF correlation for the thermal-hydraulic evaluation of 15x15 Upgrade fuel at Surry Power Station, Units 1 and 2.

In 1985, Virginia Power (Dominion) submitted to the USNRC Topical Report VEP-NE-2-A (Reference 1) describing a proposed methodology for the statistical treatment of key uncertainties in core thermal-hydraulic DNBR analysis. The methodology provided DNBR margin through the use of statistical rather than deterministic uncertainty treatment. The methodology was reviewed and approved by the USNRC in May 1987, and the SER provided by the USNRC listed the following conditions for its use (Reference 8):

1) The selection and justification of the Nominal Statepoints used to perform the plant specific implementation must be included in the submittal (Sections 3.6 and 3.8).

2) Justification of the distribution, mean and standard deviation for all the statistically treated parameters must be included in the submittal (Section 3.2).

3) Justification of the value of model uncertainty must be included in the plant specific submittal (Section 3.4).

4) For the relevant CHF correlations, justification of the 95/95 DNBR limit and the normality of the M/P distribution, its mean and standard deviation must be included in the submission, unless there is an approved Topical Report documenting these (such as Reference 2).

The methodology was first implemented for Surry in a package submitted to the USNRC in July 1991 (Reference 4) and approved in June 1992 (Reference 5) for both Surry units using the COBRA/WRB-1 Topical Report (Reference 7). This is valid for Surry cores containing SIF fuel assemblies. Now that Dominion will be purchasing 15x15 Upgrade fuel for use at Surry Units 1 and 2, the existing USNRC-approved implementation (currently for SIF) of the Statistical DNBR methodology is not applicable to 15x15 Upgrade fuel. The analysis herein is necessary because this fuel product uses different fuel design features (IFMs) and a different thermal-hydraulic code (VIPRE-D).

This report provides the technical basis for the USNRC review and approval of the implementation of the Dominion Statistical DNBR Evaluation Methodology for 15x15 Upgrade fuel at Surry with VIPRE-D/WRB-1 code/correlation pair, as well as the SDLs obtained by this implementation (DOM-NAF-2-A Condition 2). This report also documents that the existing Reactor Core Safety Limits and protection functions (OTΔT, OPΔT, FΔI, etc.) do not require revision as a consequence of this implementation (DOM-NAF-2-A Condition 3). The list of UFSAR transients for which the code/correlations will be applied is also included herein (DOM-NAF-2-A Condition 4).

Section 3 of this report summarizes the implementation of the Dominion Statistical DNBR Evaluation Methodology to Westinghouse 15x15 Upgrade fuel at Surry Power Station with the VIPRE-D/WRB-1 code/correlation pair. Section 4 provides all the necessary information for the plant specific application of the VIPRE-D/WRB-1 code/correlation pair to Surry. This section lists the applicable Deterministic Design Limits (DDLs), Statistical Design Limits (SDLs) and Safety Analysis Limits (SALs), as well as the corresponding Retained Margin. The verification of the existing Reactor Core Safety Limits, Protection Setpoints and Chapter 14 events with the above DNBR limits is also documented in Section 4.

3. Implementation of the Statistical DNBR Evaluation Methodology

3.1 Methodology Review

In Appendix B to Fleet Report DOM-NAF-2-A (Reference 2), Dominion calculated a deterministic DNBR Limit (DDL) for the VIPRE-D/WRB-1 code/correlation pair. The Statistical DNBR Evaluation Methodology (Reference 1) is employed herein to determine a Statistical DNBR limit for SPS. This new limit combines the correlation uncertainty with the DNBR sensitivities to uncertainties in key DNBR analysis input parameters. Even though the new DNBR limit (the Statistical Design Limit or SDL) is larger than the deterministic code/correlation design limit, its use is advantageous as the Statistical DNBR Evaluation Methodology permits the use of nominal values for operating initial conditions instead of requiring the application of evaluated uncertainties to the initial conditions for statepoint and transient analysis.

The SDL is developed by means of a Monte Carlo analysis. The variation of actual operating conditions about nominal statepoints due to parameter measurement and other key DNB uncertainties is modeled through the use of a random number generator. Two thousand random statepoints are generated for each nominal statepoint. The random statepoints are then supplied to the thermal-hydraulics code VIPRE-D, which calculates the minimum DNBR (MDNBR) for each statepoint. Each MDNBR is randomized by a code/correlation uncertainty factor as described in Reference 1 using the upper 95% confidence limit on the VIPRE-D/WRB-1 code/correlation pair measured-to-predicted (M/P) CHF ratio standard deviation (Reference 2). The standard deviation of the resultant randomized DNBR distribution is increased by a small sample correction factor to obtain a 95% upper confidence limit, and is then combined Root-Sum-Square with code and model uncertainties to obtain a total DNBR standard deviation (s_{total}). The SDL is then calculated as:

$$SDL = 1 + 1.645 * s_{total} \qquad \text{[Equation 3.1]}$$

in which the 1.645 multiplier is the z-value for the one-sided 95% probability of a normal distribution. This SDL thus provides peak fuel rod DNB protection at greater than 95/95.

As an additional criterion, the SDL is tested to determine the full core DNB probability when the peak pin reaches the SDL. This process is performed by summing the DNB probability of each rod in the core, using a bounding rod census curve and the DNB sensitivity to rod power. If necessary, the SDL is increased to reduce the full core DNB probability to 0.1% or less.

3.2 Uncertainty Analysis

This section is included herein to satisfy Condition 2 in the SER (Reference 8) of VEP-NE-2-A (Reference 1).

Consistent with VEP-NE-2-A, inlet temperature, pressurizer pressure, core thermal power, reactor vessel flow rate, core bypass flow, the nuclear enthalpy rise factor and the engineering enthalpy rise factor were selected as the statistically treated parameters in the implementation analysis. The magnitudes and functional forms of the uncertainties for the statistically treated parameters were derived in a rigorous analysis of plant hardware and measurement/calibration procedures, and have been summarized in Table 3.2-1.

The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and inlet temperature were quantified using all sensor, rack, and other components of a total uncertainty and combined in a manner consistent with their relative dependence or independence. Total uncertainties were quantified at the 2σ level, corresponding to two-sided 95% probability. Margin was included in these uncertainties to provide additional conservatism, and to allow for future changes in plant hardware or calibration procedures without invalidating the analysis. The standard deviations (σ) were obtained by dividing the total uncertainty by 1.96, which is the z-value for the two-sided 95% probability of a normal distribution.

Dominion has quantified the magnitude and distribution of uncertainty on the pressurizer pressure (system pressure) per the pressurizer pressure control system. The pressurizer pressure uncertainty was quantified as normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 3.468% of span or $\pm$ 27.7 psia. The impact of parameter surveillance was considered. The current parameter surveillance limit for pressurizer pressure of 2205 psig was determined to be acceptable. With this parameter surveillance limit, the pressurizer pressure uncertainty was conservatively defined as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 50.0 psia and a standard deviation (σ) of 25.51 psia. The applied uncertainty is unchanged from that employed in Reference 4 and subsequently approved in Reference 5.

Dominion has quantified the magnitude and distribution of uncertainty on the average temperature (T_{AVG}) per the T_{AVG} rod control system. The average temperature uncertainty was quantified as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 3.637% of span or $\pm$ 3.637°F. The impact of parameter surveillance was considered. The current parameter surveillance limit for average temperature of 577.0°F was determined to be acceptable. With this parameter surveillance limit, the average temperature uncertainty is conservatively defined as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 5.6°F and a standard deviation (σ) of 2.857°F. The applied uncertainty is unchanged from that employed in Reference 4 and subsequently approved in Reference 5.

Dominion has quantified the uncertainty on core power as measured by the secondary side heat balance as 1.604% at the non-uprated power 2546 MWt. The core power uncertainty associated with the MUR uprated power (2589 MWt) is less than 0.4%, but will not be credited. Thus, the power parameter uncertainty is conservatively treated as a normal, two-sided, 95% probability distribution with a magnitude of ±2.0% and a standard deviation (σ) of 1.02%. The applied uncertainty is less than that employed in Reference 4.

Dominion has quantified the uncertainty on the reactor coolant system (RCS) flow as 4.58%. This parameter uncertainty is treated as a normal, two-sided, 95% probability distribution with a magnitude of ±4.58% and a standard deviation (σ) of 2.337%. This uncertainty is larger than what is currently used in Reference 4 and approved in Reference 5.

The two-sided, 95/95 tolerance interval (95% probability, 95% confidence) for the measurement uncertainty of the nuclear enthalpy rise factor, $F_{\Delta H}^{N}$, is 3.32%. Conservatively, the measured $F_{\Delta H}^{N}$ uncertainty was defined as a normal distribution with a 4% tolerance interval for consistency with previous applications.

The magnitude and distribution of uncertainty on the engineering hot channel factor, $F_{\Delta H}^{E}$, was quantified as a normal probability distribution with a magnitude of $\pm$ 3.0%. The Statistical DNBR Evaluation Methodology (Reference 1) treats the $F_{\Delta H}^{E}$ uncertainty as a uniform probability distribution.

The total core bypass flow consists of separate flow paths through the thimble tubes, direct leakage to the outlet nozzle, baffle joint leakage flow, upper head spray flow and core-baffle gap flow. These five components were each quantified based on the current Surry core configuration, their uncertainties conservatively modeled and the flows and uncertainties totaled. The Monte Carlo analysis ultimately used a best estimate bypass flow of 5.0% with an uncertainty of 1.0%. The analysis assumed that the probability was uniformly distributed.

Table 3.2-1: Surry Parameter Uncertainties

PARAMETER	NOMINAL VALUE	STANDARD DEVIATION	UNCERTAINTY	DISTRIBUTION
Pressure [psia]	2250	25.51 psi	±50.0 psi at 2σ	Normal
Temperature [°F]	540.7	2.857°F	±5.6°F at 2σ	Normal
Power [MWt]	2,589	1.02%	±2.0% at 2σ	Normal
Flow [gpm]	273,000	2.337%	±4.58% at 2σ	Normal
$F_{\Delta H}^{N}$	1.635	2.0%	±4.0% at 2σ	Normal
$F_{\Delta H}^{E}$	1.0	N/A	±3.0%	Uniform
Bypass [%]	5.0	N/A	±1.0%	Uniform

3.3 CHF Correlations

The WRB-1/W-3 CHF correlations are used for the calculation of DNBRs in Westinghouse 15x15 Upgrade fuel assemblies. Only WRB-1 is applicable to the operating conditions for which the Statistical DNBR Evaluation Methodology applies. Table 3.3-1 presents the Design Limit correlation data for VIPRE-D/WRB-1 code/correlation pair. The W-3 correlation is only used below the first mixing grid or when the operating conditions are outside of the range of validity of the WRB-1 CHF correlation, such as the main steam-line break evaluation, where there are reduced temperature and pressure. The W-3 CHF correlation is always used deterministically.

Table 3.3-1: CHF Code/Correlation Data (Reference 2)

	WRB-1
Average M/P	1.005
S(M/P)	0.083
n	945
K*	1.03963
K x S(M/P)	0.08629

3.4 Model Uncertainty Term

This section is included herein to satisfy Condition 3 in the SER (Reference 8) of the Statistical DNBR Evaluation Methodology Topical Report (Reference 1).

The VIPRE-D 19-channel production model for Surry 15x15 Upgrade fuel was used in the development of the VIPRE-D/WRB-1 code/correlation pair SDL for Surry. Since this is the production model that Dominion intends to use for all Surry evaluations once the VIPRE-D code is added to the Technical Specifications 6.2.C list of USNRC approved methodologies, there is no additional uncertainty associated with the use of this model. In summary, it is concluded that no correction for model uncertainty is necessary, and the model uncertainty term is set to zero for the calculation of the total DNBR standard deviation.

3.5 Code Uncertainty

The code uncertainty accounts for any differences between Dominion's VIPRE-D and Westinghouse's THINC codes, with which the WRB-1 CHF data were correlated, and any effect

* K is a sample size correction factor that gives a one-sided 95% upper confidence limit on the estimated standard deviation of a given population. It can be calculated as:

$$K = \sqrt{\frac{2 \cdot (n-1)}{\left(\sqrt{2n-3} - 1.645\right)^2}}$$

due to the modeling of a full core with a correlation based upon bundle test data. These uncertainties are clearly independent of the correlation, the model, and parameter induced uncertainties. The code uncertainty was quantified at 5%; consistent with the factors specified for other thermal/hydraulic codes in Reference 1. The basis for this uncertainty is described in detail by USNRC staff in Reference 8. In Reference 8, the USNRC Staff refers to the 5% uncertainty as being a 2σ value. The 5% code uncertainty is certainly conservative in light of the excellent VIPRE-D/VIPRE-W and VIPRE-D/CHF data comparisons. However, the 5% uncertainty serves as a conservative factor that may be shown to be wholly or partially unnecessary at a later time. A one-sided 95% confidence level on the code uncertainty is then 3.04% ($= (5.0\%) / 1.645$). The use of the 1.645 divisor (the one-sided 95% tolerance interval multiplier) is conservative since the USNRC Staff considers the 5% uncertainty to be a 2σ value.

3.6 Monte Carlo Calculations

In order to perform the Monte Carlo analysis, nine Nominal Statepoints covering the full range of normal operation and anticipated transient conditions were selected. These statepoints must span the range of conditions over which the statistical methodology will be applied. Two statepoints were selected at each of the four Reactor Core Safety Limit (RCSL) pressures (2400, 2250, 2000, and 1875 psia). For each of the RCSLs, a high power, 118%, and low power, near the intercept of the DNBR limit line with the vessel exit boiling line, were chosen. In order to apply the methodology to low flow events, a low flow statepoint is also included. The inlet temperature used for each statepoint is calculated by determining the inlet temperature that would result in the desired MDNBR (1.27) for each statepoint. The selected Nominal Statepoints are listed in Tables 3.6-1.

Table 3.6-1: Nominal Statepoints for Westinghouse 15x15 Upgrade Fuel at Surry with VIPRE-D/WRB-1

STATE POINT	PRESSURIZER PRESSURE [psia]	INLET TEMPERATURE [°F]	POWER [%]	FLOW [%]	$F_{\Delta H}^{N}$**	MDNBR
A	2400.0	594.6	118	100	1.635	1.262
B	2400.0	630.9	96	100	1.655	1.262
C	2250.0	585.3	118	100	1.635	1.263
D	2250.0	620.0	97	100	1.650	1.262
E	2000.0	571.5	118	100	1.635	1.263
F	2000.0	604.0	100	100	1.635	1.262
G	1875.0	565.3	118	100	1.635	1.263
H	1875.0	593.2	103	100	1.635	1.263
I	2250.0	563.8	102	62	1.635	1.264

The Monte Carlo analysis itself consisted of 2000 calculations performed around each of the nine Nominal Statepoints. As described in Section 3.1, the DNBR standard deviation at each Nominal

** The part-power multiplier described in the Surry Core Operating Limits Report (COLR) is used for less than 100% power statepoints.

Statepoint was augmented by the code/correlation uncertainty, the small sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation.

The Total s_{Total}, is obtained using the Root-Sum-Square method according to Equation 3.2:

$$s_{TOTAL} = \sqrt{s_{DNBR}^2 \cdot \left(1.0 + \sqrt{\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}^2 + \frac{1}{N}}\right)^2 + F_C^2 + F_M^2}$$

[Equation 3.2]

where

- s_{DNBR} is the standard deviation for the Randomized DNBR distribution.
- The factor $\left\{\frac{\sqrt{n-1}}{\chi^2} - 1.0\right\}$ is the uncertainty in the standard deviation of the 2,000 Monte Carlo simulations, and provides a 95% upper confidence limit on the standard deviation.
- $1/_N$ is the uncertainty in the mean of the correlation. N is the number of degrees of freedom in the original correlation database.
- F_c is the code uncertainty, that has been defined as 5% (2σ value), i.e., $5.0\%/_{1.645} = 3.04\%$ (1σ value). See Section 2.5 in Reference 1.
- F_M is the model uncertainty, which is 0.0 since the Monte Carlo simulation is run with the production model

Note that this equation differs slightly from the equation listed in Reference 1. It has an additional factor applied to the Randomized DNBR s_{DNBR}, the $1/_N$ factor to correct for the uncertainty in the mean of the correlation. This factor has been used in previous implementations of the Statistical DNBR Evaluation Methodology, such as Reference 4.

The limiting peak fuel rod SDL was calculated to be 1.259 for VIPRE-D/WRB-1 code/correlation pair. The Monte Carlo Statepoint analysis is summarized in Table 3.6-2.

Table 3.6-2: Peak Pin SDL Results for Surry 15x15 Upgrade with VIPRE-D/WRB-1

STATEPOINT	Randomized DNB S_{DNBR}	Total DNB S_{TOTAL}	Pin Peak $SDL_{95/95}$
A	0.1420	0.1511	1.248
B	0.1469	0.1561	1.257
C	0.1468	0.1560	1.257
D	0.1483	0.1575	**1.259**
E	0.1452	0.1544	1.254
F	0.1463	0.1555	1.256
G	0.1444	0.1535	1.253
H	0.1423	0.1514	1.249
I	0.1471	0.1563	1.257

3.7 Full Core DNB Probability Summation

After the development of the peak pin 95/95 DNBR limit, the data statistics are used to determine the number of rods expected in DNB. The DNB sensitivity to rod power is estimated as ∂(DNBR)/ ∂(1/FΔh). The specific values of ∂(DNBR)/ ∂(1/FΔh), denoted β, are listed in Table 3.7-1.

To ensure that the calculations are conservative, a one-sided tolerance limit of β is used:

$$\beta^* = \beta - t(\alpha, \nu) \cdot se(\beta)$$

in which:

- β^* is the one-sided tolerance limit on β
- $t(\alpha,\nu)$ is the T-statistic with significance level α and ν degrees of freedom. For 2,000 observations at a 0.05 level of significance t(0.05,2000) = 1.645.
- se(β) is the standard error of β.

The variable 1/FΔh is the most statistically significant independent variable in the linear regression model, yielding R^2 values larger than 99%. The value of the statistic parameter F of 1/FΔh was the largest for all statepoints, which indicates that the variable 1/FΔh accounts for the largest amount of the variation in the DNBR.

Table 3.7-1: ∂(DNBR)/ ∂(1/FΔh) Estimation for WRB-1

STATEPOINT	β	se(β)	β^*	R^2
A	4.43212	0.00424	4.42514	99.9%
B	4.45065	0.00509	4.44228	99.9%
C	4.57055	0.00437	4.56336	99.9%
D	4.65588	0.00474	4.64809	99.9%
E	4.57406	0.00422	4.56712	99.9%
F	4.67252	0.00459	4.66497	99.9%
G	4.50831	0.00410	4.50156	99.9%
H	4.60068	0.00453	4.59323	99.9%
I	5.06755	0.00547	5.05855	99.9%

A representative fuel rod census curve used for the determination of the SDL is listed in Table 3.7-2. The full core DNB probability summation will be reevaluated on a reload basis to verify the applicability of the fuel rod census ($F_{\Delta H}{}^N$ versus % of core with $F_{\Delta H}{}^N$ greater than or equal to a given $F_{\Delta H}$ limit) used in the implementation analysis. The limiting full-core DNB probability summation resulted in a SDL of 1.269. The DNB probability summation for VIPRE-D/WRB-1 code/correlation pair is summarized in Table 3.7-3.

Table 3.7-2: Representative Fuel Rod Census for a Maximum Peaking Factor FΔh = 1.635

MAXIMUM % OF FUEL RODS IN CORE WITH FΔh ≥ to:	FΔh LIMIT
0.0	1.6350
0.1	1.6345
0.2	1.6285
0.3	1.6250
0.4	1.6215
0.5	1.6186
0.6	1.6159
0.7	1.6131
0.8	1.6103
0.9	1.6081
1.0	1.6059
1.5	1.5991
2.0	1.5932
2.5	1.5885
3.0	1.5837
4.0	1.5751
5.0	1.5655
6.0	1.5573
7.0	1.5490
8.0	1.5406
9.0	1.5339
10.0	1.5271
20.0	1.4673
30.0	1.4212
40.0	1.3709
PEAK	**1.6350**

Table 3.7-3: Full Core DNB Probability Summation for 15x15 Upgrade with VIPRE-D/WRB-1

STATEPOINT	S_{TOTAL}	% of Rods in DNB	Full Core $SDL_{99.9}$
A	0.1511	0.09879	1.258
B	0.1561	0.09932	**1.269**
C	0.1560	0.09969	1.266
D	0.1575	0.09927	1.268
E	0.1544	0.09843	1.263
F	0.1555	0.09942	1.263
G	0.1535	0.09902	1.262
H	0.1514	0.09965	1.255
I	0.1563	0.09998	1.257

3.8 Verification of Nominal Statepoints

Condition 1 of the USNRC's safety evaluation report for Reference 1 (Reference 8) requires that the Nominal Statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may potentially be applied.

It is therefore necessary to demonstrate that s_{total} as calculated herein is maximized for any conceivable set of conditions at which the core may approach the SDL. To do so, a regression analysis is performed using the unrandomized DNBR standard deviations at each Nominal Statepoint as the dependent variable (i.e., the raw MDNBR results obtained from the Monte Carlo simulation). The Nominal Statepoint pressures, inlet temperatures, powers and flow rates are used as the independent variable. If no clear trend appears in the plot it can be concluded that the standard deviation has been maximized. If a clear trend is displayed, the regression function is determined. This regression equation is evaluated to determine the values of the independent variable for which the standard deviation would be maximized, and it is verified that the Nominal Statepoints selected bound those conditions. In addition, the residuals of the regression are plotted again against all the independent variables, and it is verified that no trends are discernible.

Table 3.8-1 shows the R^2 coefficients obtained for the verification of the nominal statepoints. The largest linear curve fit R^2 coefficient is 23.36%, thus confirming that there is no dependence.

An evaluation of all the data, linear fits, and R^2 coefficients indicates that there are no discernible trends in the database. Therefore, it was concluded that s_{TOTAL} had been maximized for any conceivable set of conditions at which the core may approach the SDL and that the selected Nominal Statepoints provide a bounding standard deviation for any set of conditions to which the methodology may potentially be applied. Figure 3.8-1 displays a sample regression plot for WRB-1 and clearly shows the trends discussed above.

Table 3.8-1: R^2 Coefficients for the Verification of the Nominal Statepoints for Surry 15x15 Upgrade with VIPRE-D/WRB-1

	R^2 - Linear Regression
Pressure	9.66%
Temperature	7.33%
Flow Rate	7.72%
Power	23.36%

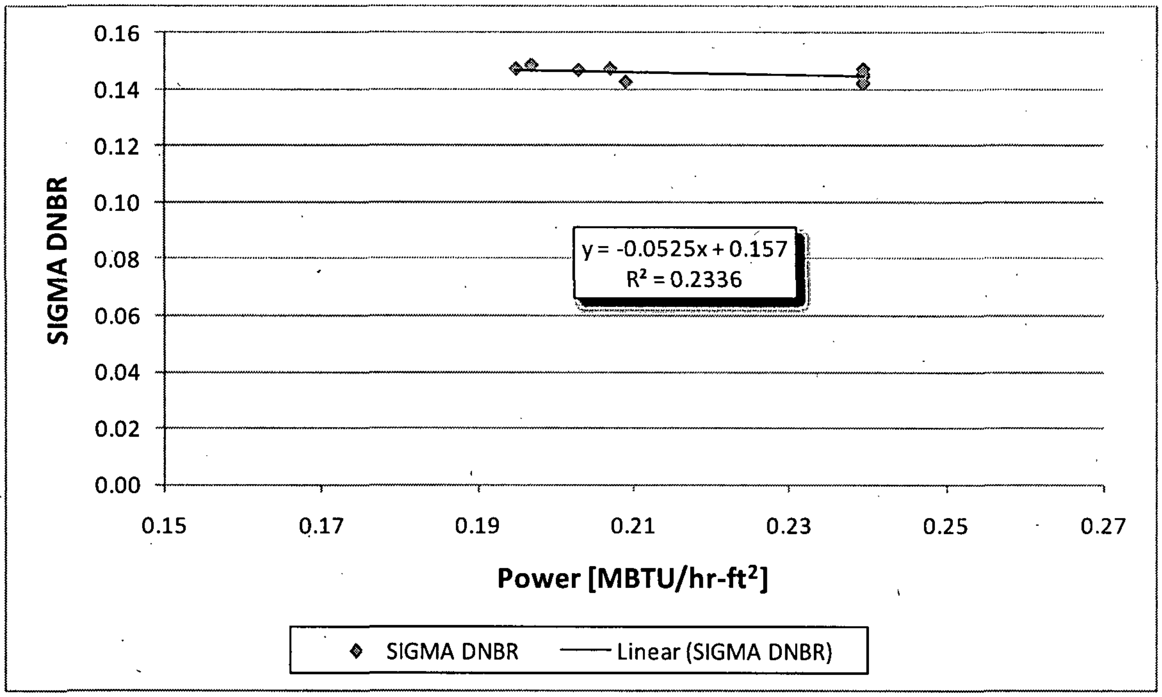

Figure 3.8-1: Variation of the Unrandomized Standard Deviation with Power for the WRB-1 CHF Correlation

3.9 Scope of Applicability

This section is included herein to satisfy Condition 4 in the SER (Reference 8) of VEP-NE-2-A (Reference 1).

The Statistical DNBR Evaluation Methodology may be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical (RWFS) which is initiated from zero power), and to the Loss of Flow analysis and the Locked Rotor Accident. The accidents to which the methodology is applicable are listed in Table 3.9-1. This table corresponds to Table 2.1-1 in Reference 2. The range of application is consistent with previous applications of Dominion Statistical DNBR Evaluation Methodology applications for Surry. This methodology will not be applied to accidents that are initiated from zero power where the parameter uncertainties are higher.

The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and hot channel uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

Table 3.9-1: UFSAR Transients Analyzed with VIPRE-D/WRB-1/W-3 for Surry

ACCIDENT	SPS USAR SECTION	APPLICATION
Uncontrolled Control-Rod Assembly Withdrawal From a Subcritical Condition	14.2.1	DET-DNB
Uncontrolled Control-Rod Assembly Withdrawal at Power	14.2.2	STAT-DNB
Control-Rod Assembly Drop/Misalignment	14.2.4	STAT-DNB
Chemical and Volume Control System Malfunction	14.2.5	Non-DNB
Start-Up of an Inactive Loop (SUIL) Accident Analysis Design Basis	14.2.6	Non-DNB
Excessive Heat Removal Due to Feedwater System Malfunctions	14.2.7	STAT-DNB
Excessive Load Increase Incident	14.2.8	STAT-DNB
Loss of Reactor Coolant Flow	14.2.9.1	STAT-DNB
Locked Rotor Incident	14.2.9.2	STAT-DNB
Loss of External Electrical Load	14.2.10	STAT-DNB
Loss of Normal Feedwater	14.2.11	Non-DNB
Loss of All Alternating Current Power to the Station Auxiliaries	14.2.12	Non-DNB
Rupture of a Main Steam Pipe	14.3.2	DET-DNB

3.10 Summary of Analysis

The steps of the SDL derivation analysis may be summarized as follows:

In accordance with the Statistical DNBR Evaluation Methodology, 2,000 random statepoints are generated about each nominal statepoint and VIPRE-D is then executed to obtain MDNBRs. The standard deviation for the distribution of 2,000 MDNBRs is referred to as the unrandomized standard deviation. At the limiting Nominal Statepoint (D), the standard deviation of the randomized DNBR distributions, which is the unrandomized corrected for CHF correlation uncertainty, was found to be 0.1483. This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation of 0.1575, as listed in Table 3.6-2. The use of 0.1575 in Equation 3.1 yields a peak pin DNBR limit of 1.259 with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core of 1.269, which occurs at Nominal Statepoint (B). Therefore the VIPRE-D/WRB-1 code/correlation pair SDL for Surry 15x15 Upgrade fuel is set to 1.27.

4. Application of VIPRE-D/WRB-1/W-3 to SPS

VIPRE-D/WRB-1 code/correlation pair together with the Statistical DNBR Evaluation Methodology will be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWFS), and to the Complete Loss of Flow event and the Locked Rotor Accident. The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}{}^{N}$ (measurement component) and $F_{\Delta H}{}^{E}$ uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

In addition, there are a few events that will be evaluated with the VIPRE-D/W-3 code/correlation pair and deterministic models because they do not meet the applicability requirements of the Statistical DNBR Evaluation Methodology (see Table 3.9-1, DET-DNB events). These events will be initiated from bounding operating conditions considering the nominal value and the appropriate uncertainty value, and require the application of the bypass flow, $F_{\Delta H}{}^{N}$ (measurement component) and $F_{\Delta H}{}^{E}$ uncertainties. The events modeled deterministically are limited by the deterministic design limit (DDL) stated in DOM-NAF-2-A (Reference 2).

4.1 VIPRE-D/WRB-1 Statistical Design Limit (SDL) for Surry

The Statistical Design Limit for Surry cores containing Westinghouse 15x15 Upgrade fuel assemblies with the VIPRE-D/WRB-1 code/correlation pair was derived in Section 3 of this report. The SDL for VIPRE-D/WRB-1 code/correlation pair is determined to be 1.27. The SDL limit provides a peak fuel rod DNB protection with at least 95% probability at a 95% confidence level and a 99.9% DNB protection for the full core. This SDL is plant specific as it already includes the Surry specific uncertainties for the key parameters accounted for in the application of the Statistical DNBR Evaluation Methodology. Therefore, this limit is applicable to the analysis of statistical DNB events of Westinghouse 15x15 Upgrade fuel in Surry cores with the VIPRE-D/WRB-1 code/correlation pair.

4.2 Safety Analysis Limits (SAL)

In the performance of in-house DNB thermal-hydraulic evaluations, design limits and safety analysis limits are used to define the available retained DNBR margin for each application. The difference between the safety analysis (self-imposed) limit and the design limit is the available retained margin.

For deterministic DNB analyses, the design DNBR limit is set equal to the applicable code/correlation limit and it is termed the deterministic design limit (DDL). For statistical DNB analyses, the design DNBR limit is set equal to the applicable statistical design limit (SDL). These design limits are two of the Design Basis Limits for Fission Product Barriers (DBLFPB) described in Reference 10. The DDLs and SDLs are fixed and any changes to their value require USNRC review and approval. However, the safety analysis limits for deterministic and statistical DNB analyses (SAL_{DET} and SAL_{STAT}, respectively) may be changed without prior USNRC review and approval, provided the changes meet the criteria established in Reference 10.

A deterministic and statistical SAL equal to 1.52 has been selected for 15x15 Upgrade fuel at Surry cores with the VIPRE-D code and the WRB-1 CHF correlation. This SAL is applicable for all deterministic analyses for a maximum peaking factor $F_{\Delta H}^{N}$ equal to 1.62 and for all statistical analyses for a maximum peaking factor $F_{\Delta H}^{N}$ equal to 1.56.

Table 4.2-1: DNBR Limits for WRB-1 and W-3

VIPRE-D/WRB-1	
DDL	1.17
SDL	1.27
SAL	1.52
VIPRE-D/W-3	
DDL (≥1000 psia)	1.30
DDL (<1000 psia)	1.45
SAL (≥1000 psia)	1.44
SAL (<1000 psia)	1.61

4.3 Retained Margin

The difference between the safety analysis (self-imposed) limit and the design limit is the available retained margin:

$$Retained\ Margin\ [\%] = \left(\frac{SAL - DDL}{SAL}\right)$$

The resulting available retained margins are listed in Tables 4.3-1 and 4.3-2.

Table 4.3-1: DNBR Limits and Retained Margin for Deterministic DNB Applications

DETERMINISTIC DNB APPLICATIONS			
DNB CORRELATION	**DDL**	**SAL_{DET}**	**RETAINED MARGIN [%]**
WRB-1	1.17	1.52	23.0
W-3 (< 1000 psia)	1.45	1.61	9.9
W-3 (≥1000 psia)	1.30	1.44	9.7

Table 4.3-2: DNBR Limits and Retained Margin for Statistical DNB Applications

STATISTICAL DNB APPLICATIONS			
DNB CORRELATION	SDL	SAL_{STAT}	RETAINED MARGIN [%]
WRB-1	1.27	1.52	16.4

This method of defining retained DNBR margin allows all the margin to be found in a single, clearly defined location. The retained DNBR margin can be used to offset generic DNBR penalties, such as a transition core penalty.

The reload thermal-hydraulics evaluation prepared as part of the reload safety analysis process presents tables and descriptions of retained margin and applicable penalties. Retained margin is tracked separately for each CHF correlation and for statistical and deterministic analyses.

4.4 Transition Core Penalties

Westinghouse has provided the transition core DNBR penalties for the WRB-1 CHF correlation for application to the 15x15 Upgrade fuel product in mixed-core configurations at Surry. Westinghouse employed their NRC-approve methodology for performing the transition core penalties with WCAP-11837-P-A (Reference 9) and WCAP-14565-P-A (Reference 12). The transition core DNBR analyses were based on VIPRE (i.e., VIPRE-W) and the WRB-1 DNB correlation. Dominion performs the DNBR analyses for Surry Units 1 and 2 using a slightly different version of VIPRE (i.e., VIPRE-D). The WRB-1 correlation is still used to analyze the SIF and 15x15 Upgrade fuel. Since the WRB-1 correlation is applicable with both VIPRE-W and VIPRE-D, then the calculations of the transition core penalty are applicable for Surry analyses. The transition core penalties will be accommodated within the retained margin.

4.5 Verification of Existing Reactor Core Safety Limits, Protection Setpoints and SPS USAR Chapter 14 Events

This section is included herein to satisfy Condition 3 of the plant specific application list in Section 2.1 of DOM-NAF-2-A (Reference 1).

To demonstrate that the DNB performance of the Westinghouse 15x15 Upgrade fuel is acceptable, Dominion performed calculations for full-core configurations of Westinghouse 15x15 Upgrade fuel. The calculations were performed using the VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs and selected statepoints including: the reactor core safety limits (RCSL), axial offset limits (AO), rod withdrawal from subcritical (RWFS), rod withdrawal at power (RWAP), loss of flow (LOFA), locked rotor events (LOCROT), hot zero power steam line break (MSLB), dropped rod limit line (DRLL), and static rod misalignment (SRM). These various statepoints provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power FΔh), pressure and temperature (RCSL); (b) limiting axial flux shapes at several axial offsets (AO); and (c) low flow (LOFA and LOCROT). The statepoints for the RWFS and MSLB were evaluated with deterministic DNB methods. The remaining statepoints were evaluated using statistical DNB methods. The evaluation criterion for these analyses is that the minimum DNBR must be equal to or greater than the applicable safety analysis limit (SAL) listed below.

The results of the calculations demonstrate that the minimum DNBR values are equal to or greater than the applicable safety analysis limit for all of the analyses that are performed to address statepoints of the Reactor Core Safety Limits, the OTΔT, OPΔT and FΔI trip setpoints, as well as all the evaluated Chapter 14 events (including the LOFA and LOCROT) with an FΔh of 1.56 (COLR limit of 1.62 divided by the measurement uncertainty of 1.04 = 1.56) at an MUR uprated power up to 2859 MWt.

5. Conclusions

Dominion's Statistical DNBR Evaluation Methodology has been used to derive a Statistical Design Limit (SDL). This application employs the VIPRE-D code with the Westinghouse WRB-1 Critical Heat Flux (CHF) correlation (VIPRE-D/WRB-1 code/correlation pair) for the thermal-hydraulic analysis of Westinghouse 15x15 Upgrade fuel assemblies at SPS. The existing Reactor Core Safety Limits, OTΔT, OPΔT and FΔI trip setpoints as well as the current analyses of applicable UFSAR Chapter 14 events were shown to be bounding, and will not be changed. In particular, Dominion seeks the review and approval of the Statistical Design Limits (SDL) of 1.27 documented herein as per 10 CFR 50.59(c)(2)(vii) it constitutes a Design Basis Limit for Fission Product Barrier (DBLFPB).

Also Dominion is seeking the approval for the inclusion of Topical Report DOM-NAF-2-A, including Appendix B, to the Technical Specification 6.2.C list of USNRC approved methodologies used to determine core operating limits (i.e., the reference list of the Surry Core Operating Limits Report (COLR)). This would allow Dominion the use of the VIPRE-D/WRB-1 code/correlation pair to perform licensing calculations for Westinghouse 15x15 Upgrade fuel in Surry cores, using the deterministic design limits (DDLs) qualified in Appendix B of Fleet Report DOM-NAF-2-A, and the statistical design limits (SDLs) documented herein. In addition, DOM-NAF-2-A provides justification of the normality of the WRB-1 CHF M/P distributions, their means and standard deviations, as required by the SER to Reference 1.

6. References

1. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," R. C. Anderson, June 1987.

2. Fleet Report, DOM-NAF-2, Rev. 0.1-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," R. S. Brackmann, July 2009.

3. Topical Report, VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.

4. Letter from W. L. Stewart (VEPCO) to US NRC Document Control Desk, "Virginia Electric and Power Company Surry Power Station Units 1 and 2 - Proposed Technical Specification Changes - FΔh Increase/Statistical DNBR Methodology," Serial No. 91-374, July 8, 1991.

5. Letter from B. C. Buckley (NRC) to W. L. Stewart (VEPCO), "Surry Units 1 and 2 – Issuance of Amendments Re: F Delta H Limit and Statistical DNBR Methodology (TAC Nos. M81271 and M82168)," ADAMS Accession No. ML012740396, June 1, 1992.

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