



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 12, 2014

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA AND SURRY POWER STATIONS UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ADDITION OF AN ANALYTICAL METHODOLOGY TO THE NORTH ANNA AND SURRY CORE OPERATING LIMITS REPORTS AND AN INCREASE TO THE SURRY MINIMUM TEMPERATURE FOR CRITICALITY (TAC NOS. MF2364, MF2365, MF2366, AND MF2367)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 271 and 253 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS), Units 1 and 2, and Amendment Nos. 283 and 283 to Renewed Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station (SPS), Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated June 26, 2013, as supplemented January 23, 2014.

These amendments revise NAPS TS 5.6.5.b, and Surry TS 6.2.C "Core Operating Limits Report (COLR)," to include "Appendix D, Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code" in the list of approved methodologies to determine the core operating limits, and revise Surry TS 2.1, "Safety Limit, Reactor Core," and TS 3.1, "Reactor Coolant System." The amendments authorize the plant-specific application of Appendix D to DOM-NAF-2-A to North Anna and Surry Power Stations (in accordance with Section 2.1 of DOM-NAF-2-A) as well as design limit for ABB-NV in Surry TS 2.1, an increase in the Surry Power Station Minimum Temperature for Criticality specified in TS 3.1.4, and references to that limit in TS 3.1.B.

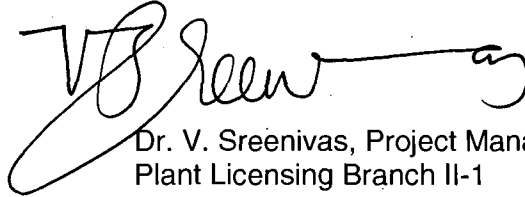
The U.S. Nuclear Regulatory Commission staff has reviewed the submitted Appendix D to the approved Fleet Report, DOM-NAF-2-A, "Reactor Core Thermal Hydraulics Using VIPRE-D Computer Code." Appendix D documents the qualification of the W-3 Alternate CHF correlations, ABB-NV and WLOP in VIPRE-D code for improved predictive capabilities for the thermal hydraulic performance of 17x17 fuel products within NAPS cores and 15x15 fuel products within the Surry cores. The safety evaluation finds the generic application of Appendix D, "Qualification of the ABB-NV and WLOP Critical Heat Flux (CHF) Correlations in the Dominion VIPRE-D Computer Code," to Fleet Report DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," acceptable within the range of validity stated.

D. Heacock

- 2 -

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

A handwritten signature in black ink, appearing to read "V. Sreenivas", with a long horizontal flourish extending to the right.

Dr. V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339
Docket Nos. 50-280 and 50-281

Enclosures:

1. Amendment No. 271 to NPF-4
2. Amendment No. 253 to NPF-7
3. Amendment No. 283 to DPR-32
4. Amendment No. 283 to DPR-37
5. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 271
Renewed License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated June 26, 2013, as supplemented January 23, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-4 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 271, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

 for

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-4
and the Technical Specifications

Date of Issuance: August 12, 2014



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 253
Renewed License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated June 26, 2013, as supplemented January 23, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 253, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "Robert J. Pascarelli" followed by "for".

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-7
and the Technical Specifications

Date of Issuance: August 12, 2014



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ATTACHMENT

TO LICENSE AMENDMENT NO. 271

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 253

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Licenses and the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment numbers and contain marginal lines indicating the areas of change.

Remove Pages

Licenses

License No. NPF-4, page 3
License No. NPF-7, page 3

TSs

5.6-4

Insert Pages

Licenses

License No. NPF-4, page 3
License No. NPF-7, page 3

TSs

5.6-4

- (2) Pursuant to the Act and 10 CFR Part 70, VEPCO to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report;
 - (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
 - (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or component; and
 - (5) Pursuant to the Act and 10 CFR Parts 30 and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.
- C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Maximum Power Level
VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2940 megawatts (thermal).
 - (2) Technical Specifications
The Technical Specifications contained in Appendix A, as revised through Amendment No. 271 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

- (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
 - (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
 - (5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations as set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 253, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the condition or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission:

- a. If VEPCO plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Power Station, the

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

13. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."
 14. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."
 15. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).
 16. DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix A, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."
 17. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO" (Westinghouse Proprietary).
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.



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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 283
Renewed License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 26, 2013, as supplemented January 23, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 283, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: August 12, 2014



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 283
Renewed License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 26, 2013, as supplemented January 23, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 283, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink that reads "Robert J. Pascarelli for". The signature is written in a cursive style.

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance August 12, 2014

ATTACHMENT

TO LICENSE AMENDMENT NO. 283

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 283

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages

License

License No. DPR-32, page 3
License No. DPR-37, page 3

TSs

TS 2.1-1
TS 3.1-6
TS 3.1-8
TS 3.1-18
TS 3.1-19
TS 6.2-2

Insert Pages

License

License No. DPR-32, page 3
License No. DPR-37, page 3

TSs

TS 2.1-1
TS 3.1-6
TS 3.1-8
TS 3.1-18
TS 3.1-19
TS 6.2-2

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 283, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 65

F. Deleted by Amendment 71

G. Deleted by Amendment 227

H. Deleted by Amendment 227

I. Fire Protection

The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.283 are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

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 REPORT 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 ≥ 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, ≥ 1.14 for ABB-NV).
 - b. The peak fuel centerline temperature shall be maintained $< 5080^{\circ}\text{F}$, 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. HEATUP AND COOLDOWN

Specification

1. Unit 1 and Unit 2 reactor coolant temperature and pressure and the system heatup and cooldown (with the exception of the pressurizer) shall be limited in accordance with TS Figures 3.1-1 and 3.1-2.

Heatup:

Figure 3.1-1 may be used for heatup rates of up to 60°F/hr.

Cooldown:

Allowable combinations of pressure and temperature for specific cooldown rates are below and to the right of the limit lines as shown in TS Figure 3.1-2. This rate shall not exceed 100°F/hr. Cooldown rates between those shown can be obtained by interpolation between the curves on Figure 3.1-2.

Core Operation:

During operation where the reactor core is in a critical condition (except for low level physics tests), vessel metal and fluid temperature shall be maintained above the reactor core criticality limits specified in 10 CFR 50 Appendix G. The reactor shall not be made critical when the reactor coolant temperature is below the Minimum Temperature for Criticality specified in T.S. 3.1.E.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

- 4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. and 200°F/hr. respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

- 5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI according to the leak test limit line shown in Figure 3.1-1.
- 6) The reactor shall not be made critical when the reactor coolant temperature is below the Minimum Temperature for Criticality specified in Technical Specification 3.1.E.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

E. Minimum Temperature for Criticality

Specifications

1. Except during LOW POWER PHYSICS TESTS, the reactor shall not be made critical at any Reactor Coolant System temperature above which the moderator temperature coefficient is more positive than the limit specified in the CORE OPERATING LIMITS REPORT. The maximum upper limit for the moderator temperature coefficient shall be:
 - a. + 6 pcm/°F at less than 50% of RATED POWER, or
 - b. + 6 pcm/°F at 50% of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER.
2. In no case shall the reactor be made critical with the Reactor Coolant System temperature below the limiting value of $RT_{NDT} + 10^{\circ}\text{F}$, where the limiting value of RT_{NDT} is as determined in Part B of this specification.
3. When the Reactor Coolant System temperature is below the minimum temperature as specified in E-2 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization.
4. The reactor shall not be made critical when the Reactor Coolant System temperature is below 538°F.

Basis

During the early part of a fuel cycle, the moderator temperature coefficient may be calculated to be slightly positive at coolant temperatures in the power operating range. The moderator temperature coefficient will be most positive near the beginning of cycle life, generally corresponding to when the boron concentration in the coolant is the greatest. Later in the cycle, the boron concentration in the coolant will generally be lower and the moderator temperature coefficient will be less positive or will be negative in the power operating range. At the beginning of cycle life, during pre-operational physics tests, measurements are made to determine that the moderator temperature coefficient is less than the limit specified in the CORE OPERATING LIMITS REPORT.

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the low power limit specified in the CORE OPERATING LIMITS REPORT has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during LOW POWER PHYSICS TESTS to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operation precautions will be taken. In addition, the strong negative Doppler coefficient⁽²⁾⁽³⁾ and the small integrated Delta k/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below the limiting value of $RT_{NDT} + 10^{\circ}\text{F}$ provides increased assurance that the proper relationship between Reactor Coolant System pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility transition temperature range. Heatup to this temperature is accomplished by operating the reactor coolant pumps.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below 538°F provides added assurance that the assumptions made in the safety analyses remain bounding by maintaining the moderator temperature within the range of those analyses.

If a specified shutdown reactivity margin is maintained (TS Section 3.12), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.

- (1) UFSAR Figure 3.3-8
- (2) UFSAR Table 3.3-1
- (3) UFSAR Figure 3.3-9

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.

REFERENCES

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-1 CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations 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"
10. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO," (Westinghouse Proprietary)



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 271 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AMENDMENT NO. 253 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

AMENDMENT NO. 283 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 283 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated June 26, 2013 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML13179A014), supplemented by additional information dated January 23, 2014 (ADAMS Accession No. ML14031A120), Virginia Electric and Power Company (Dominion or licensee) submitted a license amendment request (LAR) asking to revise the Technical Specifications (TSs) for North Anna Power Station (NAPS) Units 1 and 2 and Surry Power Station (SPS) Units 1 and 2. The proposed license amendment requests the approval of (1) generic application of Appendix D, "Qualification of the ABB-NV and WLOP Critical Heat Flux (CHF) Correlations in the Dominion VIPRE-D Computer Code," to Fleet Report DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," (2) the plant-specific application of Appendix D to DOM-NAF-2-A to North Anna and Surry Power Stations (in accordance with Section 2.1 of DOM-NAF-2-A), and (3) an increase in the Surry Power Station TS Minimum Temperature for Criticality.

Dominion has submitted Appendix D to the Fleet Report, DOM-NAF-2-A, Rev.0.2-A "Reactor Core Thermal Hydraulics Using VIPRE-D Computer Code," dated June 20, 2013 (LAR Attachment 6). Appendix D documents Dominion's qualification of the W-3 Alternate CHF correlations in VIPRE-D code performed for 17x17 fuel products within North Anna's cores and 15x15 fuel products within Surry's cores. Dominion requests the inclusion of Appendix D of the Dominion Fleet Report in the NAPS TS 5.6.5.b and Surry TS 6.2.C list of approved methodologies to determine the core operating limits. This change would allow Dominion to use

the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs to perform licensing calculations for North Anna and Surry, using the DDLs qualified in Appendix D of DOM-NAF-2.

Dominion also asks to increase the Surry TS 3.1.E.4 Minimum Temperature for Criticality limit from 522° F to 538° F and to add a departure from nucleate boiling (DNB) correlation limit for ABB-NV in Surry TS 2.1. Dominion expects that the proposed increase in the TS Minimum Temperature for Criticality will provide margin in verifying that a given reload cycle design meets the most-positive moderator temperature coefficient (MTC). Dominion also proposes to delete each specific minimum temperature criticality limit in TS 3.1.B and replace it with a cross reference to the limit specified in TS 3.1.E.

The supplement dated January 23, 2014 (ADAMS Accession No. ML14031A120), provided additional information that clarified the applications, did not expand the scope of the applications as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determinations as published in the *Federal Register* on September 3, 2013 (78 FR 54292).

2.0 REGULATORY EVALUATION

VIPRE-01 is the Electric Power Research Institute's (EPRI's) Versatile Internals and Component Program for Reactors and is used by U.S. pressurized water reactor (PWR) commercial nuclear utilities for detailed fuel safety limit analysis for both steady-state and transient applications. NRC safety evaluations, dated April 4, 2006 (ADAMS Accession No. ML060790496), June 23, 2006 (ADAMS Accession No. ML061740212), April 22, 2009 (ADAMS Accession No. ML091030639), and June 21, 2010 (ADAMS Accession No. ML101620034) which are incorporated in Fleet Report DOM-NAF-2-A, Rev.0.2-A, Appendices A, B, and C, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," August 31, 2010 (ADAMS Accession No. ML102390419), provide the approved use of VIPRE-D for PWRs applications.

In PWRs, the heat energy generated inside fuel pellets by the fission process results in heat flux at the surface of the pellet and this energy is removed by the reactor coolant flow. The most efficient heat transfer regime is nucleate boiling. Film boiling regime where a continuous layer of steam (film) starts to blanket the pellet surface is less efficient for heat transfer than during nucleate boiling and can result in significant increase of fuel rod temperature and may lead to failure of fuel rod cladding. The CHF is the heat flux (expressed in watts/cm², or Btu/hr-ft²) at which the steam film starts to form or at the point of DNB. DNB Ratio (DNBR) which is the ratio of the predicted heat flux to the actual local heat flux is a measure of thermal margin; the greater the DNBR value the greater the thermal margin. The predicted heat flux is calculated from empirical correlations that are developed through experiments and are functions of pressure, mass flux, and fuel diameter.

The NRC staff approved the ABB-NV correlation in "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-14565-P, Addendum 2, Revision 0, Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP [Westinghouse Low Pressure] for PWR Low Pressure Applications," dated February 14, 2008 (ADAMS Accession No. ML080360381) (WCAP SE), for application for fuel designs with NV grids for combustion engineering (CE) PWR with VIPRE-01 code. The Appendix D is a modified ABB-NV correlation that was developed based on rod bundle data at

low pressure and low flow conditions. This modified low pressure, low flow correlation is designated as Westinghouse Low Pressure (WLOP) correlation.

Minimum Temperature for Criticality ensures that the plant moderator temperature coefficient (MTC) will be in the range from slightly positive to negative for transient and accident analyses.

2.1 Proposed Technical Specifications (TS) Changes

North Anna TS 5.6.5.b Core Operating Limits Report (COLR) and Surry COLR TS 6.2.C

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. The cycle specific parameter limits must be determined for each reload cycle in accordance with TS 5.6.5 for NAPS and TS 6.2.C for Surry. Plant operation within these limits is addressed in individual TSs.

Dominion proposes to modify the reference to DOM-NAF-2-A by adding Appendix D, as follows:

North Anna TS 5.6.5.b.16 COLR:

DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix A, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."

Surry TS 6.2.C COLR:

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," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."

The added text would authorize the Dominion-specific analytical application of the ABB-NV and WLOP CHF correlations that are proposed to be used in determining core limits.

Surry TS 2.1 Safety Limit, Reactor Core

General Design Criteria (GDC) of 10 CFR 50 Appendix A, requires that specified acceptable fuel design limits (SAFDL) are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a DNB design basis that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature in terms of a safety limit (SL). The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant.

The licensee requests revision of the TS 2.1 Safety Limit, Reactor Core as follows:

Revise TS 2.1.A.1.a to include " ≥ 1.14 for ABB-NV". The change adds the ABB-NV correlation and the corresponding VIPRE-D Deterministic Design Limit.

Surry TS 3.1 Reactor Coolant System

The minimum temperature for criticality specified in TS helps ensure that the plant MTC will be in the range from slightly positive to negative for transient and accident analyses. The MTC relates a change in core reactivity to a change in coolant (Moderator) temperature. Positive MTC means reactivity increases with increasing moderator temperature and conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. A coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value, i.e., the temperature increase will be self-limiting, and will result in stable power operation.

The licensee proposed that TS 3.1 be revised as follows:

Revise TS 3.1.E.4 to reflect the proposed Minimum Temperature for Criticality limit that is being increased from 522° F to 538° F.

Revise TS 3.1.B.1 to reference the applicable specification for the Minimum Temperature for Criticality, TS 3.1.E, instead of a specific temperature limit (currently 522° F).

2.2 Regulatory Evaluation

Dominion is proposing to add Appendix D of Fleet Report DOM-NAF-2 that documents the qualification of W-3 Alternate CHF correlation in VIPRE-D and lists the deterministic design limits (DDLs) for each alternate correlation. The W-3 Alternate CHF correlations are the ABB-NV and WLOP CHF correlations. The W-3 CHF correlation has been used to predict departure from nucleate boiling ratio (DNBR) margin in the non-mixing vane grid (NMVG) region to supplement the primary DNB correlation used for Westinghouse and CE PWR fuel designs with mixing vane grids (MVG) and for low flow, low pressure conditions of Westinghouse and CE PWR fuel designs. The DNBR is the ratio of the predicted CHF to the local heat flux under a given set of conditions. Therefore, DNBR is a measure of thermal margin to film boiling. In order to avoid fuel damage, the margin to film boiling in terms of SL must be maintained at all times during normal operations as well as AOOs and accidents.

10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 10, states, "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

Section 4.4, "Thermal and Hydraulic Design," of the Standard Review Plan (SRP, NUREG-0800) provides the DNB acceptance criteria. The objectives of the review are to confirm that the thermal and hydraulic design of the core and the reactor coolant system (RCS),

- uses acceptable analytical methods,
- is equivalent to or is a justified extrapolation from proven designs,
- provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and
- is not susceptible to thermal-hydraulic instability.

Dominion expects that the proposed increase in the minimum temperature for criticality to 538° F from current value of 522° F along with the use of the ABB-NV and WLOP CHF correlation at the Surry units will increase the available retained DNBR thermal margin. Retained DNBR margin is quantified as the percent difference between the Safety Analysis Limit (SAL) and deterministic design limit (DDL) and will be described in detail in Section 3.0 of this safety evaluation.

The requirements for information that must be included in TSs are set forth in 10 CFR 50.36(c). The requested TS revisions are related to safety limits, limiting conditions of operation (minimum temperature for criticality), and administrative controls (COLR).

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the LAR to evaluate the applicability, confirm that the use of this methodology is within NRC approved ranges of applicability, and verify that the results of the analyses are in compliance with the requirements specified in Appendix A to 10 CFR Part 50.

3.1 Qualification of the ABB-NV and WLOP CHF Correlations in VIPRE-D

The NRC staff's technical evaluation of the requests for use of the ABB-NV and WLOP CHF correlations is based on the NRC staff's WCAP SE dated February 14, 2008 (ADAMS Accession No. ML080360381) which approved WCAP-14565-P (ADAMS Accession No. ML062780167) (Proprietary, publicly withheld)[ADAMS Accession No. ML062780173, Addendum 2 to WCAP-15306-NP-A "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications"]. Attachment 6 of the LAR documents Dominion's qualification of the ABB-NV and the WLOP correlations with the VIPRE-D code (ADAMS Accession No. ML13179A014). Dominion's qualification of the ABB-NV and WLOP correlations was performed against data from the Columbia University Heat Transfer Research Facility (HTRF). The same data were used by Westinghouse in the qualification of the ABB-NV and WLOP correlations with the VIPRE-01 code. These data evaluations were performed to qualify the VIPRE-D/ABB-NV and VIPRE-D/WLOP code correlation pairs and to develop the DDL for each correlation in VIPRE-D.

VIPRE-D ABB-NV Correlation

ABB-NV correlation which is based on a linear relationship between CHF and local quality includes the following variables: pressure, local mass velocity, local equilibrium quality, distance from grid to CHF location, heated length from inlet to CHF location, and heated hydraulic diameter of the subchannel. The ABB-NV correlation has been used with the TORC code and the VIPRE code for CE-PWR licensing applications. The ABB-NV correlation predicts the DNBR at normal operating conditions of pressure and flow for Westinghouse and CE PWR fuel designs. All test sections used in the development, validation, and qualification of the ABB-NV CHF correlations contained only NMVGs, specifically the correlation is used below the first MVG to supplement the first DNB correlation used in the analysis. Two supplemental tests were selected to demonstrate that the ABB-NV correlation is applicable for the region of Westinghouse PWR fuel designs below mixing vane grids at the core inlet. One rod bundle test, identified as Test 190, was performed on a brazed Inconel NV grid design with 21.5 inch grid spacing at Columbia University's Heat Transfer Research Facility in the 1980's. The second test was performed on a NV grid design in a hexagonal rod bundle with similar rod and hydraulic

diameter dimensions as the Westinghouse PWR with 0.360-inch diameter rods and 0.496-inch rod pitch.

The Dominion qualification of ABB-NV in VIPRE-D was performed against the same test data from Columbia University HTRF database used by CE and Westinghouse to develop the ABB-NV correlation. Table D.4.1-1 of LAR provides a summary of the ABB-NV CHF experimental database for the Dominion ABB-NV correlation development.

VIPRE-D Results as Compared to Westinghouse Results

Dominion used the statistical tests that are described in NRC staff-approved topical report (TR) WCAP-14565-P (ADAMS Accession No. ML062780167) for its ABB-NV correlation to determine 95/95 DNBR limit for the adoption of ABB-NV and WLOP correlations for PWR fuel designs. The W and D Tests for normality at the 95% confidence level were performed on the correlation data sets to determine the proper statistical methods to be used for the data. Statistical tests were then performed to determine if all or selected data groups belong to the same population, in order to be combined for the evaluation of the 95/95 DNBR limit. Using the Bartlett's test, the homogeneity of variance for normally distributed groups was examined. The analysis of variance (ANOVA) F-test was applied to multiple groups that passed the normality tests. For groups that did not pass the normality test, the Wilcoxon-Mann-Whitney test or the Kruskal-Wallis One-Way Analysis of Variance by Ranks test is used to test the null hypotheses that the medians, or averages, of the tests or groups are the same.

The mathematical model for each ABB-NV CHF test section is described in NRC staff-approved TR WCAP-14565-P (ADAMS Accession No. ML062780167). The rod bundles and cell geometry, the rod radial peaking values, the rod axial flux shapes, the spacer grid type and their locations, and the thermocouple locations incorporating many of the tests are included in the ABB-NV correlation. Westinghouse used the test data set and the VIPRE-01 thermal-hydraulics computer code to calculate a DDL of 1.13 for the ABB-NV correlation. Dominion used these experimental data to develop the VIPRE-D/ABB-NV code/correlation pair DDL, as described in Section D.5.1 of LAR.

Modeling analysis of the appropriate test sections was performed with the VIPRE-D thermal-hydraulic code using the methodology described in Section 4 of the NRC-approved generic Fleet Report (ADAMS Accession No. ML102390419). VIPRE-D produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc.) at every axial node along the heated length of the test section. The ratio of measured-to-predicted CHF (M/P) is the variable that is normally used to evaluate the thermal-hydraulic performance of a code/correlation pair. The measured CHF is the local heat flux at a given location, while the predicted CHF is calculated by the code using the ABB-NV CHF correlation. The ratio of these two values (M/P) is the inverse of the DNB ratio and is frequently used to validate the CHF correlations because their distribution is usually a normal distribution and it simplifies the statistical analyses. Table D.5.1-1 of LAR lists the average M/P, standard deviation, maximum M/P, and minimum M/P for the ABB-NV test data. The results of the statistical tests performed are listed in Tables D.5.1-2, D.5.1-3, and D.5.1-4 LAR. The results include those from normality and poolability of database and other additional statistical tests which were performed on groups that did not pass the normality test.

For normally distributed data, one-sided tolerance theory is used for the calculation of the VIPRE- D/ABB/NV code/correlation pair DDL. The results indicate that the overall distribution for the M/P ratios is a normal distribution. The D' value for the normality tests for the selected number of data points with a 95% confidence level is found to be within the range of acceptability. The DNBR DDL for the subset is calculated from:

$$DNBR_{DDL} = \frac{1.0}{\frac{M}{P} - K * \sigma \frac{M}{P}} \quad (1) \text{ and}$$

$$K = \frac{1.645 + 1.645 \left[1 - \frac{(2.706)}{2(N)} * \left(1 - \frac{1}{n} \right) \right]^{\frac{1}{2}}}{1 - \frac{2.706}{2(N)}} \quad (2)$$

Where,

M/P = average measured to predicted CHF ratio

$\Sigma_{M/P}$ = standard deviation of the measured to predicted CHF ratios of the database (practically equivalent to Owen's table)

K = 95/95 confidence multiplier (practically equivalent to Owen's Factor)

n = number of data points

N = degrees of freedom

The parametric DDL for the VIPRE-D ABB-NV code/correlation pair is calculated to be 1.1357 per Table D.5.1-5 of Attachment 6 (ADAMS Accession No. ML13179A014).

The DDL for the non-parametric data that fails the D' normality tests, the DDL is calculated using a distribution free approach. When comparing the nonparametric data results in a calculated DDL of 1.1313 per Table D.5.1-6 of Attachment 6 (ADAMS Accession No. ML13179A014) to the parametric DDL of 1.14, the conservative DDL is 1.14.

Figures D.5.1-1 through D.5.1-8 of Attachment 6 (ADAMS Accession No. ML13179A014), display the performance of the M/P ratio, and its distributions as a function of the pressure, mass velocity, quality, heated hydraulic diameter ratio, matrix (typical channel) heated hydraulic diameter, heated length, and distance from the grid. These plots show that there are no biases in the M/P ratio distribution, and that the performance of the ABB-NV correlation is independent of the independent variables of interest.

Therefore, the NRC staff concludes based on the non-parametric data results in a DDL of 1.13, that the conservative DDL is 1.14. This value is a conservative number compared to the value approved by NRC staff in the WCAP SE dated February 14, 2008 (ADAMS Accession No. ML080360381). The NRC staff concludes that the DDL value of minimum DNBR 1.14 is conservative within the range of applicability identified in the NRC staff WCAP SE. Therefore, Dominion request for generic application of Appendix D, qualification of the ABB-NV CHF

correlations in the Dominion VIPRE-D Code to Fleet Report DOM-NAF-2-A is acceptable within that range.

Table 1 of the NRC staff WCAP SE dated February 14, 2008 (ADAMS Accession No. ML080360381) summarizes the ranges of validity of the VIPRE-D/ABB-NV code/correlation pair for generic Westinghouse PWR fuel design application.

Table 1: Range of Validity for ABB-NV Correlation for VIPRE-D

Parameter	VIPRE-D
Pressure (psia)	1750 to 2415
Mass Velocity (Mlbm/hr-ft ²)	0.8 to 3.16
Thermodynamic Quality at CHF	≤ 0.22
Heated Hydraulic Diameter	0.679 to 1.08
Heated Length (in)	48.0* to 150.0
Distance from Grid (in)	7.3 to 24

*Though the heated length below the first mixing grid is below 48 inches, the minimum heated length used in the correlation is conservatively set to 48 inches.

3.1.2 VIPRE-D WLOP Correlation

The WLOP CHF correlation was developed by Westinghouse to replace the W-3 and Macbeth CHF correlations used to predict the DNBR at low pressure/low flow conditions for PWR fuel designs. WLOP is a modification of the ABB-NV correlation and was developed based on CHF data of rod bundles obtained at the HTRF for PWR 14x14, 16x16, and 17x17 fuel designs containing structural NMVGs. WLOP CHF correlation was developed for low pressure conditions and extended flow range to cover low flow conditions that are not within the approved range of applicability of the primary DNB correlation used in analysis. The axial shape correction factor for non-uniform power shapes used with WLOP CHF correlation is the same as that used with the primary CHF correlation.

VIPRE-D WLOP Results

The mathematical model for each separate WLOP CHF test section is described in NRC staff TR WCAP-14565-P (ADAMS Accession No. ML080360381) by utilizing the bundle and cell geometry, the rod radial peaking values, the rod axial flux shapes, the spacer grid information (i.e., type, axial locations and form losses) and the thermocouple locations. Westinghouse used the test data set and the VIPRE-01 thermal-hydraulics computer code to calculate a DDL of 1.18 for the WLOP correlation. Dominion used these experimental data, as described in Section 3.1.1 to develop the VIPRE-D/WLOP code/correlation pair DDL. Modeling analysis with VIPRE-D code was performed using the methodology described in Fleet Report (ADAMS Accession No. ML102390421). VIPRE-D produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc.) at every axial node along the heated length of the test section. The ratio of measured CHF at a given location of the fuel rod to the predicted CHF that is calculated by the code using the WLOP CHF correlation (M/P ratio) is the inverse of the DNB ratio and is used to validate CHF correlations instead of using the DNB ratios because

their distribution is usually a normal distribution. Table D.5.2-1 of Attachment 6 provides the average M/P, standard deviation, maximum M/P, and minimum M/P for the WLOP test data.

Table D.5.2-2 of Attachment 6 lists the results of W and D' normality tests. The results of the statistical tests for determining normality and poolability of the datasets are presented in Tables D.5.2-2 and D.5.2-3 of Attachment 6. The results from additional statistical tests performed on groups that did not pass the normality test are listed in Table D.5.2-4 of Attachment 6. The statistical tests presented in these tables are consistent with the statistical tests presented in Addendum 2 to WCAP-15306-NP-A Extended Application of ABB-NV Correlation and Modified BB-NV Correlation WLOP for PWR Low Pressure Applications, dated September 2006 (ADAMS Accession No. ML062780167).

The VIPRE-D/WLOP code/correlation pair DDL is calculated using both parametric and non-parametric statistical techniques. The most limiting calculated DDL is then applied as the DDL for the entire VIPRE-D/WLOP core/correlation pair DDL. The parametric DDL is calculated using the equations as presented in Table D.5.2-5, "Statistical Analysis of VIPRE-D/WLOP DDL." The parametric data results in a DDL of 1.22.

The DDL for a non-parametric data set is calculated using a distribution free approach as approved in the NRC staff WCAP SE dated February 14, 2008 (ADAMS Accession No. ML080360381).

This distribution-free approach is applied to data groups that do not pass the D' normality test and is known as non-parametric data. The non-parametric DDLs are listed in Table D.5.2-6 of Attachment 6. The conservative DDL is 1.22 when the non-parametric data results of DDL of 1.21 is compared with the parametric DDL of 1.22.

Figures D.5.2-1 through D.5.2-8 of Attachment 6 display the performance of the M/P ratio, and its distributions as a function of the pressure, mass velocity, quality, heated hydraulic diameter ratio, matrix (typical channel) heated hydraulic diameter, heated length, and grid spacing term. The plots show that there are no biases of the M/P ratio distribution, and that the performance of the WLOP correction is independent of the independent variables of interest.

Therefore, the NRC staff concludes, based on the comparison of non-parametric DDL of 1.21 listed in Table D.5.2-6 to the parametric DDL of 1.22 calculated using the equations as presented in Table D.5.2-5, that a DDL of 1.22 value is a conservative number compared to the value 1.21 in NRC staff WCAP SE dated February 14, 2008 (ADAMS Accession No. ML080360381). The NRC staff concludes that the DDL value of minimum DNBR 1.22 is conservative within the range of applicability identified in the NRC staff WCAP SE. Therefore, the Dominion request for generic application of Appendix D, "Qualification of the ABB-NV CHF correlations in the Dominion VIPRE-D Code, to Fleet Report DOM-NAF-2-A is acceptable within that range.

Table 2 of the NRC staff WCAP SE dated February 14, 2008 (ADAMS Accession No. ML080360381) summarizes the ranges of validity for the VIPRE-D/WLOP correlation. These ranges are identical to those submitted by Westinghouse. A minimum value of 48 inches for the heated length is used in the calculation of the WLOP predicted CHF value.

Table 2: Range of Validity for WLOP Correlation

Parameter	VIPRE-D
Pressure (psia)	185 to 1800
Mass Velocity (Mlbm/hr-ft ²)	0.23 to 3.07
Thermodynamic Quality at CHF	≤0.75
Matrix Heated Hydraulic Diameter (in)	0.4635 to 0.5334
Heated Hydraulic Diameter Ratio	0.679 to 1.00
Heated Length (in)	48* to 168
Grid Spacing Term	27 to 115

*Though the heated length below the first mixing grid is below 48 inches, the minimum heated length used in the correlation is conservatively set to 48 inches.

3.2 Application of Appendix D of DOM-NAF-2 to North Anna Units

Consistent with NRC staff safety evaluation dated April 4, 2006 (ADAMS Accession No. ML060790496), the licensee intends to implement Appendix D of DOM-NAF-2 in its plant-specific applications through the following methods: (1) TS changes to add applicable appendices (Appendix D) to DOM-NAF-2 to the Core Operating Limits Report (COLR) list (see section 2.1 above), (2) changes to the statistical design limit (SDL) for the relevant code and correlation(s), (3) identification of any TS changes to over temperature delta T (OTΔT), over pressure delta T (OPΔT), enthalpy rise factor (FΔH), or reactor protection function, as well as any changes to Reactor Core Safety Limits (RCSLs), and (4) the identification of any changes to the list of updated final safety analysis report (UFSAR) transients for which the code and correlations apply.

Licensee will not use the W-3 alternate CHF correlations statistically in licensing bases calculations at North Anna Power Station. Instead, W-3 alternate CHF correlations will be applied deterministically for licensing basis calculations for North Anna, Units 1 and 2. The applicable DDLs for the W-3 Alternate CHF correlations are documented in Appendix D of DOM-NAF-2 and are 1.14 for ABB-NV and 1.22 for WLOP.

The licensee has incorporated a deterministic SAL for both the ABB-NV and WLOP CHF correlations at North Anna for use in the analysis of Westinghouse RFA-2 fuel design. The retained departure from nucleate boiling ratio (MDNBR) margin is calculated as the difference between SAL and DDL as:

$$\text{Retained DNBR Margin (\%)} = ((\text{SAL}-\text{DDL})/\text{SAL}) * 100$$

This retained margin in DNBR is used to offset cycle-specific DNB penalties such as rod bow penalty or a penalty for transition core for mixed cores.

The OT Δ T, OP Δ T and f Δ H, and reactor protection functions are established to ensure bounding protection for the core thermal limits. Attachment 6 (Appendix D) states that the ABB-NV correlation is applied to UFSAR transients where the limiting location of DNBR occurs below the first MVG, and the WLOP CHF correlation is applied to UFSAR transients when the conditions occur outside the range of applicability of the primary CHF correlation, specifically low pressure or low flow conditions. The licensee has demonstrated acceptable DNB performance of the Westinghouse RFA-2 fuel at North Anna with the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs using the DDLs qualified in Attachment 6. The MDNBR values are equal or greater than the applicable SAL for all of the reload analyses. These calculations support the current RCSLs; the OT Δ T and OP Δ T trip functions (including F Δ I reset functions); F Δ H limits; and the evaluated UFSAR Chapter 15 events. Accordingly, there are no changes to the OT Δ T and OP Δ T trip functions and F Δ H limits (ADAMS Accession No. ML14031A120).

Rod withdrawal from subcritical (RWSC) and Main Steam Line Break (MSLB) are two UFSAR Chapter 15 accidents where the W-3 Alternate CHF correlation is limiting with respect to DNB performance in licensing basis analyses. Since the RWSC event occurs within the applicable range of the primary CHF correlation with the exception of the limiting location of DNBR potentially being below the first MVG, the ABB-NV CHF correlation is applied to the RWSC transient. The limiting DNBR for the MSLB event occurs under low pressure conditions and is outside the applicable range of the primary CHF correlation. Thus, the WLOP CHF correlation is applied to the MSLB event. The other events in Chapter 15, the list of UFSAR transients for which the code and correlations apply, as listed in Table 1, UFSAR Transients Analyzed with VIPRE-D, in NRC staff safety evaluation dated April 4, 2006 (ADAMS Accession No. ML060790496) are limited within the applicable range of the primary CHF correlation and therefore, not affected by the implementation of W-3 alternate CHF correlation.

NRC staff concludes that the addition of Appendix D to COLR TS will ensure that cycle specific parameters for operation of the units will be determined consistent with plant safety analyses and applicable safety limits. Based on the thermal performance evaluation in Section 3.1, the licensee has shown that the current reactor protection setpoints will continue to provide bounding protection for both units. Licensee has also identified the UFSAR events (RWSC and MSLB) where the new correlations will be applied to calculate the thermal margin performance. Therefore, the NRC staff concludes that the application of Appendix D is acceptable for North Anna.

3.3 Application of Appendix D of DOM-NAF-2 to Surry Units

Consistent with NRC staff safety evaluation dated April 4, 2006 (ADAMS Accession No. ML060790496), the licensees intend to implement Appendix D of DOM-NAF-2 in its plant-specific applications through the following methods: (1) TS changes to add applicable appendices (Appendix D) to DOM-NAF-2 to the Core Operating Limits Report (COLR) list (see section 2.1 above), (2) changes to the statistical design limit (SDL) for the relevant code and correlation(s), (3) identification of any TS changes to over temperature delta T (OT Δ T), over pressure delta T (OP Δ T), enthalpy rise factor (F Δ H), or reactor protection function, as well as any changes to Reactor Core Safety Limits (RCSLs), and (4) the identification of any changes to the list of UFSAR transients for which the code and correlations apply. The LAR has addressed all these methods.

The licensee will not use the W-3 alternate CHF correlations statistically in licensing basis calculations at SPS. Instead, W-3 alternate CHF correlations will be applied deterministically for licensing basis calculations for Surry, Units 1 and 2. The applicable DDLs for the W-3 Alternate CHF correlations are documented in Appendix D of DOM-NAF-2, and are 1.14 for ABB-NV and 1.22 for WLOP.

The licensee has incorporated a deterministic safety analysis limit (SAL) for both the ABB-NV and WLOP CHF correlations at SPS for use in the analysis of Westinghouse 15x15 fuel design. The retained departure from nucleate boiling ratio (MDNBR) margin is calculated as the difference between SAL and DDL as:

$$\text{Retained DNBR Margin (\%)} = ((\text{SAL}-\text{DDL})/\text{SAL}) * 100$$

This retained margin in DNBR is used to offset cycle-specific DNB penalties such as rod bow penalty or a penalty for transition core for mixed cores.

The OTΔT, OPΔT and fΔH, and reactor protection functions are established to ensure bounding protection for the core thermal limits. Attachment 6 (Appendix D) states that the ABB-NV correlation is applied to UFSAR transients where the limiting location of DNBR occurs below the first MVG, and the WLOP CHF correlation is applied to UFSAR transients when the conditions occur outside the range of applicability of the primary CHF correlation, specifically low pressure or low flow conditions. The licensee has demonstrated acceptable DNB performance of the Westinghouse RFA-2 fuel at Surry with the VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs using the DDLs qualified in Attachment 6. The MDNBR values are equal or greater than the applicable SAL for all of the reload analyses. These calculations support the current RCSLs; the OTΔT and OPΔT trip functions (including FAI reset functions); FΔH limits; and the evaluated UFSAR Chapter 14 events. Accordingly, there are no changes to the OTΔT and OPΔT trip functions and FAH limits (ADAMS Accession No. ML14031A120).

RWSC and MSLB are two UFSAR Chapter 14 accidents where the W-3 Alternate CHF correlation is limiting with respect to DNB performance in licensing basis analyses. Since the RWSC event occurs within the applicable range of the primary CHF correlation with the exception of the limiting location of DNBR potentially being below the first MVG, the ABB-NV CHF correlation is applied to the RWSC transient. The limiting DNBR for the MSLB event occurs under low pressure conditions and is outside the applicable range of the primary CHF correlation. Thus, the WLOP CHF correlation is applied to the MSLB event. The other events in Chapter 14, the list of UFSAR transients for which the code and correlations apply, as listed in Table 1, UFSAR Transients Analyzed with VIPRE-D, in NRC staff safety evaluation dated April 4, 2006 (ADAMS Accession No. ML060790496) are limited within the applicable range of the primary CHF correlation and therefore, not affected by the implementation of W-3 alternate CHF correlation.

The NRC staff concludes that the addition of Appendix D to COLR TS will ensure that cycle specific parameters for operation of the units will be determined consistent with plant safety analyses and applicable safety limits. Based on the thermal performance evaluation in Section 3.1, the licensee has shown that the current reactor protection setpoints will continue to provide bounding protection for both units. Licensee has also identified the UFSAR events (RWSC and MSLB) where the new correlations will be applied to calculate the thermal margin performance. Therefore, the NRC staff concludes that the application of Appendix D is acceptable for Surry.

3.4 Surry Safety Limit, Reactor Core

The licensee proposed to revise Surry TS 2.1 Safety Limit, Reactor Core. The request is to add " ≥ 1.14 for ABB-NV" to TS 2.1.A.1. The DDL value of minimum DNBR 1.14 is conservative within the range of applicability identified in the NRC staff WCAP SE. DNB will not occur by requiring that fuel centerline temperature stays below the melting temperature in terms of a safety limit. Therefore, the proposed TS change to Surry, Units 1 and 2, is acceptable. The change adds the ABB-NV correlation and the corresponding VIPRE-D Deterministic Design Limit.

3.5 Increase in the Surry Minimum Temperature for Criticality

Dominion proposes to increase the Surry TS 3.1.E.4 Minimum Temperature for Criticality limit from 522° F to 538° F. Dominion also proposed modifying Surry TS 3.1.B.1 Reactor Coolant System heatup and cooldown to reflect the revised limit by removal of the current temperature and a cross reference to TS 3.1.E.

There are four requirements in Surry TS 3.1.E Minimum Temperature for Criticality that must be met prior to taking units at Surry critical. They are:

1. The maximum positive Moderator Temperature Coefficient limit is met,
2. The reactor is not made critical below the temperature corresponding to the non-ductile failure of the reactor vessel,
3. The reactor is subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization when the RCS is below the temperature corresponding to non-ductile failure of the reactor vessel, and
4. The reactor is not taken critical below a given RCS temperature (currently 522° F).

The proposed increase in the minimum temperature for criticality does not affect the plants capability to satisfy the first three requirements listed above. Since the increase in Minimum Temperature for criticality is conservative with regard to non-ductile failure, the increase provides additional margin to non-ductile failure. The proposed change does not affect the allowable RCS heatup and cooldown rates.

The increased Minimum Temperature for Criticality will be verified against the assumptions in the safety analyses on a reload basis and does not impact the NRC-approved analytical methods used to determine the core operating limits such as the moderator temperature coefficient. The change to TS 3.1.B.1, which add cross references to the Minimum Temperature of Criticality specified in TS 3.1.E, appropriately reflect the increased temperature and will ensure that an acceptable plant MTC will be maintained. Therefore, the NRC staff finds the proposed increase in Minimum Temperature for Criticality for Surry, Units 1 and 2, is acceptable.

3.6 Summary and Conclusion

The NRC staff has reviewed the LAR, in conjunction with the supplemental information the responses to the NRC staff's requests for additional information to evaluate the acceptability of the license requests. Specifically the NRC staff concludes that:

The W-3 Alternate CHF correlation has been used to predict DNBR margin. The calculated DDL value of minimum DNBR 1.14 for ABB-NV is conservative within the range of applicability for ABB-NV CHF correlations identified in the NRC staff WCAP SE. A DDL value of minimum DNBR 1.22 for WLOP is conservative within the range of applicability identified in the NRC staff WCAP SE. Therefore, the Dominion request for generic application of Appendix D, qualification of the ABB-NV and WLOP CHF correlations in the Dominion VIPRE-D Code, to Fleet Report DOM is acceptable within the range stated.

The W-3 Alternate CHF correlations, the ABB-NV and the WLOP CHF correlations, are approved for thermal hydraulic performance analyses of Westinghouse 17x17 fuel products within North Anna, Units 1 and 2, cores within the range of applicability noted in Tables 1 and 2 of this SE.

The W-3 Alternate CHF correlations, the ABB-NV and the WLOP CHF correlations, are approved for thermal hydraulic performance analyses of Westinghouse 15x15 fuel products within Surry, Units 1 and 2, cores within range of applicability noted in Tables 1 and 2 of the SE.

The proposed TS changes that add Appendix D to Fleet Report DOM-NAF-2-A, to NAPS TS 5.6.5.b and Surry TS 6.2.C will ensure that safety analyses are conducted to ensure that operation of the units remain consistent with applicable safety limits.

The DDL value of minimum DNBR 1.14 is conservative within the range of applicability identified in the NRC staff WCAP SE. DNB will not occur by requiring that fuel centerline temperature stays below the melting temperature in terms of a safety limit. Therefore, the proposed change to include " ≥ 1.14 for ABB-NV" in Surry TS 2.1 is acceptable. The change adds the ABB-NV correlation and the corresponding VIPRE-D Deterministic Design Limit.

The proposed increase in the Surry, Units 1 and 2, Minimum Temperature for Criticality provides margin in verification that cycle-specific core design meets the maximum-positive MTC. The increased Minimum Temperature for Criticality is verified against the assumptions in the safety analyses on a reload basis and does not impact the NRC-approved analytical methods used to determine the core operating limits such as the MTC. The change to TS 3.1.B.1 (to reference minimum temperature for criticality in TS 3.1.E instead of the specific temperature limit), appropriately reflect the increased temperature and will ensure that an acceptable plant MTC will be maintained. Therefore, the NRC staff finds the proposed increase in Minimum Temperature for Criticality for Surry, Units 1 and 2, and the modified cross reference to MTC acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration,

and there has been no public comment on such finding (78 FR 54292). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributor: Mathew M. Panicker, NRR

Date: August 12, 2014

D. Heacock

- 2 -

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Dr. V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339
Docket Nos. 50-280 and 50-281

Enclosures:

1. Amendment No. 271 to NPF-4
2. Amendment No. 253 to NPF-7
3. Amendment No. 283 to DPR-32
4. Amendment No. 283 to DPR-37
5. Safety Evaluation

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