July 18, 2006

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

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS ON CHANGES TO ANALYTICAL METHODOLOGY AND CORE OPERATING LIMITS REPORT (TAC NOS. MC7526 AND MC7527)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 247 and 227 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2 (North Anna 1 and 2). The amendments change the Technical Specifications (TSs) in response to your application dated July 5, 2005, as supplemented by letters dated March 30, April 13, and May 11, 2006.

These amendments add a reference to Technical Specification 5.6.5.b, "Core Operating Limits Report (COLR)," to permit the use of an alternate methodology to perform thermal-hydraulic analysis to predict the Critical Heat Flux (CHF) and Departure from Nucleate Boiling Ratio (DNBR) for the Advanced Mark-BW Fuel at North Anna 1 and 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/

Stephen Monarque, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 247 to NPF-4
2. Amendment No. 227 to NPF-7
3. Safety Evaluation

cc w/encls: See next page

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Package No. ML Tech Spec No. ML Amendment No. ML 062020005

OFFICE	NRR/LPL2-1/PM	NRR/LPD2-1/LA	NRR/SPWB/BC	OGC	NRR/LPL2-1/BC
NAME	SMonarque	MO’Brien	JNakoski	MYoung	EMarinos
DATE	6/12 /2006	6/20/2006	6/14/2006	6/29/2006	7/18/2006

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
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 July 5, 2005, as supplemented by letters dated March 30, April 13, and May 11, 2006, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 247, 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

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: 247
Changes to License No. NPF-4
and the Technical Specifications

Date of Issuance: July 18, 2006

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 227
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 July 5, 2005, as supplemented by letters dated March 30, April 13, and May 11, 2006, 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. 227, 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

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: 227
Changes to License No. NPF-7
and the Technical Specifications

Date of Issuance: July 18, 2006

ATTACHMENT TO

LICENSE AMENDMENT NO. 247 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

LICENSE AMENDMENT NO. 227 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NOS. 50-338 AND 50-339

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

Remove Pages	Insert Pages
Licenses	Licenses
License No. NPF-4, page 3	License No. NPF-4, page 3
License No. NPF-7, page 3	License No. NPF-7, page 3
TSs	TSs
5.6-4	5.6-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 247 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 227 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated July 5, 2005 (Reference 1) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051890034), as supplemented by letters dated March 30, April 13, and May 11, 2006 (References 2, 3, and 4) (ADAMS Accession Nos. ML060900631, ML061040062, and ML061310495, respectively), Virginia Electric and Power Company (the licensee) requested an amendment to the Technical Specifications (TSs) for North Anna Power Station Units 1 and 2 (North Anna 1 and 2). The licensee's proposed changes would add Topical Report DOM-NAF-2-A, "Reactor Core Thermal-Hydraulic Using the VIPRE-D Computer Code," including Appendix A, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," (Reference 5) to TS 5.6.5.b, "Core Operating Limits Report (COLR)." In addition, the licensee requested the Nuclear Regulatory Commission (NRC) staff's approval of a code-specific Statistical Design Limits (SDL), related to its implementation of an NRC staff approved Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," to allow the use of AREVA Advanced Mark-BW (AMBW) fuel at North Anna 1 and 2.

Currently, the licensee uses the LYNXT code and the BWU critical heat flux (CHF) correlations for the departure from nucleate boiling (DNB) analysis of the AMBW fuel in the North Anna 1 and 2 reactor cores. The approval of these amendments would permit the licensee to use the VIPRE-D code to perform the DNB analyses, the BWU CHF correlations, and the statistical core design methodology for the AMBW fuel.

The supplements dated March 30, April 13, and May 11, 2006, contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.36(c)(2)(ii) requires that TS limiting conditions for operation (LCOs) be established for process variables, design features, and operating restrictions for which a value is assumed as an initial condition of a design-basis accident in the licensee's safety analyses. As such, license amendments are generally required for each fuel cycle to update the values of cycle-specific parameter limits in the TSs. To eliminate the need for a license amendment to update the cycle-specific parameter limits for each fuel cycle while meeting 10 CFR 50.36(c)(2)(ii) requirements, the NRC has allowed licensees to use an alternative to incorporate the cycle-specific parameter limits in the COLR. Generic Letter (GL) 88-16 provides the COLR-implementation guidance that includes the requirement to list in the TSs the NRC-approved analytical methods used to determine the core operating limits. The analytical methods referenced in the TSs would identify the topical report(s) by number, title, and date, or identify the NRC staff's safety evaluation report for a plant-specific methodology by NRC letter and date. Also, to eliminate the need for a TS change for every approved revision or supplement to a listed topical report, TS Task Force (TSTF) Traveler TSTF-363 allows for listing of only the report numbers and titles in the TSs, with the detailed identification of the report revisions, supplement numbers and approval dates specified in the COLR.

3.0 TECHNICAL EVALUATION

3.1 Addition of Topical Report DOM-NAF-2-A and Appendix A to TS 5.6.5.b

Topical Report DOM-NAF-2 and its Appendix A, respectively, describe the VIPRE-D computer code and the qualifications of the BWU CHF correlations. VIPRE-D is the licensee's version of the VIPRE-01 code, which is an NRC staff-approved finite volume subchannel thermal-hydraulic code with specific capabilities to model a three-dimensional core and other component geometries. Using appropriate boundary conditions from a system code such as RETRAN, VIPRE-01 computes the flow, void, pressure, and temperature distribution of the fluid through the core to ultimately compute the minimum DNBR for steady-state and transient conditions. VIPRE-D has been enhanced by the addition of several vendor-specific CHF correlations.

For the calculations of DNBRs for the AMBW fuel assemblies, the licensee intends to utilize the VIPRE-D code and the BWU-N, BWU-Z, and BWU-ZM correlations. BWU-N is used in the non-mixing grid span from the beginning of the heated length to the leading edge of the first structural mixing grid. BWU-Z is used in the span from the leading edge of the first structural mixing grid to the leading edge of the second structural mixing grid, and in the uppermost span beyond the last mixing grid. BWU-ZM is an enhanced form of BWU-Z with a multiplicative performance factor, and this correlation is used from the leading edge of the second structural mixing grid to the leading edge of the last structural mixing grid. The use of these BWU CHF correlations with the VIPRE-D code and the qualification of VIPRE-D/BWU pair are described in Appendix A to Topical Report DOM-NAF-2.

The qualification of the BWU-N, BWU-Z and BWU-ZM correlations with the VIPRE-D code was performed against the same CHF experimental database used by Framatome-ANP to develop and license the correlations. Appendix A to Topical Report DOM-NAF-2 summarized the data evaluations that were performed to qualify the VIPRE-D/BWU code/correlation pair, and to develop the corresponding deterministic DNBR design limits for each correlation. Table A.5-1 of the licensee's submittal dated September 30, 2004, summarized the VIPRE-D DNBR limits for the BWU-Z, BWU-ZM and BWU-N correlations, and Table A.5-2 of this same submittal specified the range of validity for these correlations. In its letter dated April 4, 2006, the NRC staff approved of Topical Report DOM-NAF-02 and Appendix A for licensing applications (Reference 6). The licensee's proposed TS change would include the approved Topical Report DOM-NAF-02-A and its Appendix A in TS 5.6.5.b, "Core Operating Limits Report (COLR)." The NRC staff finds this proposed TS change acceptable because it follows the guidance of GL 88-16 and TSTF-363 in that only the topical report numbers and titles will be listed in the TSs, along with the detailed identification of the report revisions, supplement numbers, and approval dates specified in the COLR.

3.2 Implementation of the Statistical DNBR Evaluation Methodology

The licensee uses the NRC staff-approved Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" for the safety analyses of many anticipated transient events listed in Table 3.1.9-1 of the July 5, 2005, submittal (Reference 1). The statistical DNBR limits, referred to as the SDL, are associated with the use of the VIPRE-D code; therefore, the BWU CHF correlations must be determined.

The statistical methodology derives SDLs that statistically account for the uncertainties associated with the safety analysis input parameters, the thermal-hydraulic code and modeling, and the CHF correlations. For each of the selected statepoints, a DNBR standard deviation is developed using a Monte Carlo process to combine the CHF correlation uncertainty with the DNBR sensitivities to uncertainties in key DNBR analysis input parameters. The standard deviation of the resultant randomized DNBR distribution is then combined statistically using the "square-root-of-sum-of the squares" (RSS) method with thermal-hydraulic code and analysis model uncertainties to obtain a total standard deviation. The SDL is then derived from the total standard deviation with a one-sided tolerance limit factor to provide 95/95 hot fuel rod protection from DNB. As an additional criterion, a core-wide DNB probability analysis is performed to find an SDL that resulted in an expected number of rods in DNB to be less than 0.1 percent of the total in the core. The NRC staff finds that the final SDL is the more restrictive of the SDLs derived from the peak rod DNB probability and the core-wide DNB probability.

3.2.1 Input Parameter Uncertainty Analysis

The input parameters whose uncertainties are treated statistically in the SDL determination are inlet temperature, pressurizer pressure, core thermal power, reactor pressure vessel flow rate, core bypass flow, nuclear enthalpy rise factor, and engineering enthalpy rise factor. Table 3.1.2-1 of the licensee's submittal dated July 5, 2005 (Reference 1), lists the nominal values, standard deviations, uncertainties, and probability distributions for these parameters. In response to the NRC staff's requests for additional information (RAI) (References 2 and 3), the licensee provided detailed analyses that derived the uncertainty values.

The measurement uncertainties of a parameter include errors, allowances, or drift of various components within the measurement channel, such as process measurement accuracy, sensors, processing equipment, readout devices, etc. These uncertainties are then combined statistically using the RSS methodology for the those error components that are statistically independent. The errors that are interactive are added arithmetically into groups to form an independent group that is statistically combined.

Systematic errors, or bias, are combined arithmetically.

The licensee's calculations showed that uncertainties of the temperature, pressure, core power and RCS flow are lower than those listed in Table 3.1.2-1. For the remaining statistically treated parameters in Table 3.1.2-1, i.e., nuclear enthalpy rise factor, engineering enthalpy rise factor, local and bundle spacing, and core flow bypass, their uncertainties and probability distributions are either consistent with previous applications or are more conservative than the calculated values. Therefore, the NRC staff determines that the standard deviations and uncertainties of Table 3.1.2-1 used in the SDL calculations are acceptable.

3.2.2 Code Uncertainty

The thermal-hydraulic code uncertainty is applied to account for the effect of analyzing a full core with a correlation, which was based only upon steady-state test bundle data, and the effect of performing analyses with the VIPRE-D and LYNXT codes, which the BWU CHF data were correlated. This code uncertainty was quantified at 5 percent. The 5 percent uncertainty is consistent with the values specified for other thermal-hydraulics codes that the NRC staff approved in Topical Report VEP-NE-2-A.

3.2.3 Model Uncertainty

A model uncertainty accounts for differences between the simple core thermal-hydraulics model with which the correlation DNBR limit was derived and the sophisticated model which is normally used for production calculations. Since the same VIPRE-D 14-channel production model used for the safety analyses for North Anna was used in the development of the NRC staff approved VIPRE/BWU DNBR limits, and the licensee is not adding additional uncertainty to the model, the NRC staff finds this acceptable.

3.2.4 CHF Correlation Uncertainty

As discussed in Section 3.1 of this safety evaluation, the BWU correlation consists of three correlations for different types of spacer grids: BWU-N, which is applicable in the presence of non-mixing vane grid; BWU-Z, which is the enhanced mixing vane correlation and is applicable to the DNB analysis of the fuel assembly in the mixing grid region; and BWU-ZM, which is BWU-Z with a multiplication enhancement factor and is applicable in the presence of mid-span mixing grids. Appendix A to Topical Report DOM-NAF-2 documented the licensee's qualification of the BWU-N, BWU-Z and BWU-ZM CHF correlations with the VIPRE-D code. Since a CHF correlation is an analytical fit to experimental data, it has an associated uncertainty that is quantified in a DNBR design limit. Tables A.4.1-2, A.4.2-2, and A.4.3-2 of the licensee's submittal dated September 30, 2004, summarized the standard deviations of the measured-to-predicted (M/P) CHF ratios and the design DNBR limits for the BWU-Z, BWU-ZM and BWU-N correlations. It should be noted that the DNBR limits for these correlations are deterministic design DNBR limits based solely on the M/P distributions from the VIPRE-D/BWU code/correlation set. Table 3.2.1-1 of the licensee's submittal dated July 5, 2005 (Reference 1), summarizes the DNBR limits of the three BWU correlations at respective pressure ranges. These DNBR limits are the deterministic design limits (DDL) for the deterministic DNBR analyses of the anticipated transient events.

3.2.5 Statistical Design DNBR Limit

BWU-Z and BWU-ZM are exactly the same correlations, except for the multiplicative performance factor applied to the BWU-ZM to correct for the thermal-hydraulic performance improvement due to the mid-span mixing grids. As such, the licensee combined these two correlations into one group, BWU-Z/ZM, to obtain an SDL applicable to them. In its letter dated May 11, 2006 (Reference 4), the licensee stated that the BWU-ZM correlation uncertainties were used to obtain the BWU-Z/ZM SDL because the BWU-ZM correlations were slightly more conservative than the uncertainties for BWU-Z. As a consequence, two SDLs were calculated, one for BWU-N, and the other for BWU-Z/ZM.

The Monte Carlo analysis was performed for nine selected nominal statepoints for the BWU-Z/ZM correlations and sixteen selected nominal statepoints for the BWU-N correlation covering the full range of the normal operation and anticipated transient conditions. As listed in Tables 3.1.6-1 and 3.1.6-2 of the licensee's submittal dated July 5, 2005, these selected conditions spanned the pressure range between the high and low trip setpoints, inlet temperatures between normal operation and maximum heatup, power range up to 118 percent of the overpower limit, and a bounding low-flow event.

The variation of actual operating conditions about nominal statepoints due to parameter measurement uncertainties is modeled with the assistance of a random number generator. For each nominal statepoint, two thousand random statepoints are generated and then supplied to the thermal-hydraulic code VIPRE-D, which calculates the minimum DNBR for each one of them. Each minimum DNBR is randomized by a CHF correlation uncertainty factor using the upper 95 percent confidence limit on the VIPRE-D/BWU M/P CHF ratio standard deviation. The standard deviation of the resultant randomized DNBR distribution is increased by a small sample correction factor to obtain a 95-percent upper-confidence limit, and is then combined statistically using the RSS method with thermal-hydraulic code and analysis model uncertainties to obtain a total standard deviation. The SDL is then derived from the total standard deviation with a multiplier factor (1.645) for one-sided tolerance limits at 95-percent probability of a normal distribution to provide 95/95 hot fuel rod protection from DNB. Tables 3.1.6-3 and 3.1.6-4 of the licensee's submittal dated July 5, 2005, summarize the Monte Carlo statepoint analysis results of the randomized DNBR standard deviation, the total standard deviation, and the pin peak SDL for each statepoint for the BWU-Z/ZM and BWU-N correlations. The 95/95 SDLs are 1.307 and 1.315, for the BWU-Z/ZM and BWU-N correlations, respectively. These SDLs were chosen from the highest pin peak SDL for all statepoints shown in Tables 3.1.6-3 and 3.1.6-4.

The licensee, in Section 3.1.8 of its submittal dated July 5, 2005, demonstrated that the selected nominal statepoints provided bounding DNBR standard deviations. For the BWU-Z/ZM and the BWU-N correlations, a regression analysis was performed for the unrandomized DNBR standard deviation using each statepoint as the dependent variable and using pressure, inlet temperature, power, and flow rate as the independent variables. A correlation coefficient, R^2, was calculated for the curve fit for each independent variable. As shown in the response to the NRC staff's RAI (Reference 2), all the regression analyses performed for each independent variable showed extremely low correlation coefficients, indicating that the unrandomized DNBR standard deviations are not related to the independent variables evaluated. Therefore, the NRC staff concludes that the DNBR standard deviations have been maximized, and the SDLs thus derived are also maximized and acceptable.

3.2.6 Core-Wide DNB Probability Analysis

As an additional criterion, the core-wide DNB probability analysis was performed to determine an SDL that would result in an expected number of rods in DNB to be less than 0.1 percent of the total number of rods in the core. This is done using a bounding census of fuel rod radial peaking factors (provided in the response to the NRC staff's RAI, Reference 2). The DNBR of each fuel rod can be calculated by the use of a conservative DNB sensitivity to rod power. Given the DNBR standard deviation, the probability of DNB is calculated for each rod and summed over the entire core. The SDL is determined iteratively until an expected number of rods in DNB is less than 0.1 percent of the total number of rods in the core, that is, 99.9 percent of fuel rods in the core do not experience DNB. Tables 3.1.7-1 and 3.1.7-2 of the licensee's submittal dated July 5, 2005, provide the core-wide DNB probability analysis results for the BWU-Z/ZM and BWU-N correlations, respectively. The core-wide SDLs are determined to be 1.34 and 1.38 for the BWU-Z/ZM and BWU-N correlation, respectively. Since these core-wide DNB probability SDLs are higher than the SDLs determined from the peak rod 95/95 SDLs, they are chosen as the SDLs.

3.2.7 Safety Analysis Limits (SAL)

The DDLs and SDLs discussed above are the design-basis limits for the deterministic and statistical DNBR applications of the analyses of the design-basis events that must not be exceeded for anticipated operational occurrences as required by GDC 10. However, in actual safety analyses, the licensee may choose higher DNBR limits, referred to as the safety analysis limits (SAL) to maintain margins from the design limits. The licensee selected a deterministic and statistical SAL of 1.60 for the AMBW fuel at North Anna 1 and 2 cores that were analyzed with the VIPRE-D code and all BWU correlations. The only exception is the deterministic SAL for BWU-Z/ZM correlations at pressures below 700 psia that was selected to be 1.85. Since these SALs are higher than the DDLs and SDLs discussed earlier in this safety evaluation, the NRC staff finds the SALs to be acceptable.

3.2.8 Transition Core Penalties

In Section 3.2.5 of the licensee's submittal dated July 5, 2005, the licensee indicated that the first application of the VIPRE-D/BWU code/correlation set will be the third transition core with the AMBW fuel, and will most probably be a full core of the AMBW assemblies that will not carry any transition core penalty. The first two transition cores with the AMBW fuel and existing fuel assemblies were analyzed with the LYNXT code and the BWU correlations. Should a transition mixed-core configuration exist, the licensee will need to ensure that the retained margin between the SALs and the SDLs or DDLs is sufficient to cover the necessary mixed-core penalty shown on Table 3.2.5-1 of the July 5, 2005, submittal.

3.2.9 Verification of Existing Reactor Core Safety Limits and Protection Setpoints

The licensee stated that it performed calculations for the full-core AMBW fuel using the VIPRE-D/BWU code/correlation set and the statepoint conditions that were defined in its March 28, 2002, license amendment requests to use AMBW fuel (Reference 7). The NRC staff approved of these license amendments for North Anna 1 and 2 on August 20 and April 1, 2004, respectively. These statepoints represent points on the reactor core safety limit (RCSL) lines, limited axial flux shapes at several axial offsets, and statepoints for several transient events.

Most of the statepoints were evaluated with the statistical methodology, with the exception of the statepoints for the rod withdrawal from subcritical condition and the main steam line break events that were evaluated with the deterministic methodology, consistent with the application of the statistical methodology listed in Table 3.1.9-1 of the licensee's submittal dated July 5, 2005 (Reference 1). The results of the calculations demonstrate that the minimum DNBR values are equal to or greater than the SAL of 1.60. Therefore, the existing RCSLs remain unchanged as a result of implementation of the Topical Report DOM-NAF-2 VIPRE-D/BWU code/correlation set and the VEP-NE-2-A statistical DNBR evaluation methodology. Consequently, the setpoints of the Overtemperature ΔT, Overpower ΔT, and f(ΔI) trip and reset functions that are based on the RCSLs also remain unchanged, and the NRC staff finds this acceptable.

3.3 Conclusion

The NRC staff has reviewed the licensee's request for (1) inclusion of the NRC-approved Topical Report DOM-NAF-2-A and associated Appendix A in the North Anna 1 and 2 TSs, and (2) implementation of the NRC-approved VEP-NE-2-A for the statistical evaluation methodology, including the DDLs, SDLs, and SALs for the VIPRE-D/BWU set. The NRC staff concludes that the licensee's proposed license amendments are acceptable since the use of the stated methodology will ensure that core operating limits will be determined such that all applicable limits of the analysis will be met.

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 a requirement 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 (70 FR 48208). 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 Commission 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) 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.

7.0 REFERENCES

1. Letter from E. S. Grecheck, Virginia Electric and Power Company, to NRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2[,] Proposed Technical Specification Changes[,] Addition of Analytical Methodology to COLR," July 5, 2005, Serial No. 05-419.

2. Letter from W. R. Matthews, Virginia Electric and Power Company, to NRC, "Virginia Electric and Power Company (Dominion)[,] North Anna Power Station Unit Nos. 1 and 2[,] Response to Request for Additional Information on Proposed Technical Specification Changes on Addition of Analytical Methodology to the Core Operating Limits Report (TAC Nos. MC7526 and MC7527)," March 30, 2006, Serial No. 06-142.

3. Letter from E. S. Grecheck, Virginia Electric and Power Company, to NRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Unit Nos. 1 and 2[,] Proposed Technical Specification Changes on Addition of Analytical Methodology to the Core Operating Limits Report, Administrative Correction[,]" April 13, 2006, Serial No. 06-142A.

4. Letter from E. S. Grecheck, Virginia Electric and Power Company, to NRC, "Virginia Electric and Power Company (Dominion)[,] North Anna Power Station Unit Nos. 1 and 2[,] Response to Request for Additional Information on Proposed Technical Specification Changes on Addition of Analytical Methodology to the Core Operating Limits Report (TAC Nos. MC7526 and MC7527)" May 11, 2006, Serial No. 06-142B.

5. Letter from L. N. Hartz, Virginia Electric and Power Company, to NRC, "Virginia Electric and Power Company (Dominion), Dominion Nuclear Connecticut, Inc. (DNC), North Anna and Surry Power Stations Units 1 and 2, Millstone Power Station Units 2 and 3, Request for Approval of Topical Report DOM-NAF-2, Reactor Core Thermal-Hydraulic Using the VIPRE-D Computer Code, Including Appendix A - Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," September 30, 2004, Serial No. 04-606.

6. Letter from C. I. Grimes, NRC, to D. A. Christian, Virginia Electric and Power Company, "Millstone Power Station, Unit Nos. 2 and 3 (Millstone 2 and 3), North Anna Power Station, Unit Nos. 1 and 2 (North Anna 1 and 2), and Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2) - Approval of Dominion's Fleet Report DOM-NAF-2, 'Reactor Core Thermal-Hydraulic Using the VIPRE-D Computer Code,' (TAC Nos. MC4571, MC4572, MC4573, MC4574, MC4575, and MC4576)," April 4, 2006.

7. Letter from L. N. Hartz, Virginia Electric and Power Company, to NRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes and Exemption Request, Use of Framatome ANP Advanced Mark-BW Fuel," March 28, 2002, Serial No. 02-167.

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