VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 5, 2005

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

Serial No. 05-419
NL&OS/ETS R0
Docket Nos. 50-338/339
License Nos. NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGES
ADDITION OF ANALYTICAL METHODOLOGY TO COLR

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests amendments, in the form of changes to the Technical Specifications to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes will add a reference in 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 Critical Heat Flux (CHF) and Departure from Nucleate Boiling Ratio (DNBR) for the Advanced Mark-BW (AMBW) Fuel. In addition, plant specific application of the methodology requires NRC approval of site/fuel type/code specific Statistical Design Limits (SDLs). Specifically, the following is being requested:

- Inclusion of Topical Report DOM-NAF-2, including Appendix A, to the Technical Specification 5.6.5.b list of NRC approved methodologies used to determine core operating limits [i.e., the reference list of the North Anna Core Operating Limits Report (COLR)].

- Implementation of the NRC-approved VEP-NE-2-A Dominion "Statistical DNBR Evaluation Methodology" for AREVA AMBW fuel in North Anna cores with the VIPRE-D/BWU code/correlation set. In particular, Dominion seeks the review and approval of the Statistical Design Limits (SDLs) documented herein as per 10 CFR 50.59 since they constitute a Design Basis Limit for Fission Products Barrier (DBLFPB).

In September 2004, Dominion submitted Topical Report DOM-NAF-2 (including Appendix A, which describes the verification and qualification of the BWU CHF correlations) to the NRC for review and approval. DOM-NAF-2 provided documentation to describe the intended uses of VIPRE-D for Dominion applications. With the approval of DOM-NAF-2, the Technical Specification change request, and Design Basis Limit for Fission Products Barrier (DBLFPB), Dominion will be licensed to perform in-house the DNB analyses using the VIPRE-D/BWU code/correlation set for the intended uses described in DOM-NAF-2. This capability supports the use of AREVA AMBW fuel at North Anna Power Station, Units 1 and 2.

We have evaluated the proposed Technical Specifications changes and have determined that they do not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the changes do not involve a significant hazards consideration is included in Attachment 1. We have also determined that operation with the proposed changes will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes. The basis for our determination that the changes do not involve any significant increase in effluents or radiation exposure is also included in Attachment 1.

The proposed changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee, as well as, the Management Safety Review Committee.

Dominion requests approval of this license amendment request by September 1, 2006. This requested schedule permits in-house performance of DNB analyses with DOM-NAF-2 and the VIPRE-D/BWU code/correlation set in support of use of AREVA AMBW fuel at North Anna Power Station Units 1 and 2 for operating cycles 20 and 19, respectively. This change will be implemented within 60 days of NRC approval.

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services

Attachments

Commitments made in this letter:

1. No regulatory commitments are made in this letter.

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COMMONWEALTH OF VIRGINIA)
)
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The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President – Nuclear Support Services, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 5th day of July, 2005.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

Serial No. 05-419

DISCUSSION OF CHANGE

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

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DISCUSSION OF CHANGE

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests an amendment to Facility Operating License Numbers NPF-4 and NPF-7 in the form of changes to the Technical Specifications (TS) for North Anna Power Station Units 1 and 2. The proposed changes will add a reference in 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 Critical Heat Flux (CHF) and Departure from Nucleate Boiling Ratio (DNBR) for the Advanced Mark-BW (AMBW) Fuel. In addition, plant specific application of the methodology requires NRC approval of site/fuel type/code specific Statistical Design Limits (SDLs). Specifically, the following is being requested:

1. Implementation of the NRC-approved VEP-NE-2-A Dominion "Statistical DNBR Evaluation Methodology" (Reference 1) for AREVA AMBW fuel in North Anna cores with the VIPRE-D/BWU code/correlation set. In particular, Dominion seeks the review and approval of the Statistical Design Limits (SDLs) documented herein as per 10 CFR 50.59 since they constitute a Design Basis Limit for Fission Products Barrier (DBLFPB).

2. Inclusion of Topical Report DOM-NAF-2, including Appendix A (Reference 2), to the Technical Specification 5.6.5.b list of NRC approved methodologies used to determine core operating limits [i.e. the reference list of the North Anna Core Operating Limits Report (COLR)]. This would allow Dominion the use of the VIPRE-D/BWU code/correlation set to perform licensing calculations for AREVA AMBW fuel in North Anna cores, using the deterministic design limits (DDLs) qualified in Appendix A of the DOM-NAF-2 Topical Report, and the statistical design limits (SDLs) reviewed and approved in 1) above.

With these approvals, Dominion will be licensed to perform in-house the DNB analyses for the intended uses described in DOM-NAF-2 to support AREVA AMBW fuel at North Anna Power Station, Units 1 and 2 using the VIPRE-D/BWU code/correlation set.

The proposed changes qualify for categorical exclusion for an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change.

2.0 BACKGROUND

Dominion is purchasing fuel assemblies from Framatome ANP, an AREVA company (AREVA), for use at North Anna Power Station, Units 1 and 2. These assemblies have been inserted in Units 1 and 2, commencing with Cycles 18 and 17 respectively. The fuel assemblies are designated as the Advanced Mark-BW (AMBW). These assemblies are a one-for-one replacement for the resident fuel product, which is the North Anna Improved Fuel (NAIF) with ZIRLO components and PERFORMANCE+ debris resistant features (a Westinghouse fuel product) (Reference 3).

Currently, all Departure from Nucleate Boiling (DNB) analyses required to support the AMBW fuel at North Anna Power Station, Units 1 and 2 are performed by AREVA using the LYNXT thermal-hydraulic computer code with the AREVA BWU CHF correlations (LYNXT/BWU) (References 4 and 5). Prior to the use of AMBW, and currently for the NAIF fuel product, Dominion performed all DNB analysis in-house with the NRC-approved COBRA computer code and the WRB-1 CHF correlation (COBRA/WRB-1) (Reference 16). By adding the VIPRE-D code with the BWU CHF correlations (VIPRE-D/BWU) to the list of methodologies approved for the determination of core operating limits, Dominion intends to attain the capability to perform in-house all DNB analyses for AMBW fuel at North Anna, Units 1 and 2, thus retaining vendor independence.

VIPRE-D is the Dominion version of the computer code VIPRE (Versatile Internals and Components Program for Reactors - EPRI), developed for EPRI (Electric Power Research Institute) by Battelle Pacific Northwest Laboratories in order to perform detailed thermal-hydraulic analyses to predict CHF and DNBR of reactor cores (References 6 through 10). VIPRE-01 has been approved by the NRC (References 11 and 12). VIPRE-D, which is based upon VIPRE-01, MOD-02.1, was customized by Dominion to fit the specific needs of Dominion's nuclear plants and fuel products.

In September 2004, Dominion submitted Topical Report DOM-NAF-2 (including Appendix A, which describes the verification and qualification of the BWU CHF correlations) (Reference 2) to the NRC for review and approval. DOM-NAF-2 provided the necessary documentation to describe the intended uses of VIPRE-D for PWR licensing applications. Appendix A qualified the BWU CHF correlations with the VIPRE-D code and listed the deterministic code/correlation DNBR limits. In addition, Section 2.1 of DOM-NAF-2 listed the information to be provided to the NRC by Dominion for the review and approval of any plant specific application of the VIPRE-D code:

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

2) Statistical Design Limit(s) for the relevant code/correlation(s) (Section 3.2.2)

3) Any technical specification changes related to OTΔT, OPΔT, FΔI or other reactor protection function, as well as revised Reactor Core Safety Limits (Section 3.2.6).

4) List of UFSAR transients for which the code/correlations will be applied (Section 3.1.9).

This report provides the NRC the necessary documentation (items 1 through 4 above) to review and approve the application of the VIPRE-D methodology with the BWU CHF correlations for the thermal-hydraulic evaluation of AREVA AMBW fuel at North Anna Power Station, Units 1 and 2.

In late 1985, Virginia Power (Dominion) submitted to the NRC Topical Report VEP-NE-2-A (Reference 1) describing a proposed methodology for the statistical treatment of key uncertainties in core thermal-hydraulics DNBR analysis. The methodology provided DNBR margin through the use of statistical rather than deterministic uncertainty treatment. The margin could then be used to provide relief in areas where plant safety analysis is DNBR-limited. The methodology was reviewed and approved by the NRC in May 1987, and the SER provided by the NRC listed the following conditions (Reference 13):

1) The selection and justification of the Nominal Statepoints used to perform the plant specific implementation must be included in the submittal (Sections 3.1.6 and 3.1.8).

2) Justification of the distribution, mean and standard deviation for all the statistically treated parameters must be included in the submittal (Section 3.1.2).

3) Justification of the value of model uncertainty must be included in the plant specific submittal (Section 3.1.4).

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

The Statistical DNBR Evaluation Methodology was first implemented for North Anna in a package submitted to the NRC in June, 1987 (Reference 14) and approved in June 1989 (Reference 15) for both North Anna units following approval of the COBRA/WRB-1 Topical Report (Reference 16). This methodology was valid for North Anna cores containing Westinghouse NAIF fuel assemblies. Now that Dominion has purchased AREVA AMBW fuel for use at North Anna units 1 and 2, the existing NRC-approved implementation of the Statistical DNBR methodology is not applicable to AREVA AMBW fuel, because this fuel product uses a different CHF correlation (AREVA BWU Correlations), because a different code (VIPRE-D) will be used to perform the thermal-hydraulic evaluations, and because the key parameter uncertainties may be different for the AREVA AMBW fuel product.

As a consequence, this report provides the technical basis for the NRC review and approval of the implementation of the Dominion Statistical DNBR Evaluation Methodology for AREVA AMBW fuel at North Anna with VIPRE-D/BWU, as well as the SDLs obtained by this implementation (DOM-NAF-2 condition 2). This report also documents that the existing Reactor Core Safety Limits and protection functions (OTΔT, OPΔT, FΔI, etc) do not require revision as a consequence of this implementation (DOM-

NAF-2 Condition 3). The list of UFSAR transients for which the code/correlations will be applied is also included herein (DOM-NAF-2 Condition 4).

Section 3.1 of this report summarizes the implementation of the Dominion Statistical DNBR Evaluation Methodology to AREVA AMBW fuel at North Anna Power Station with the VIPRE-D/BWU code/correlation. Section 3.2 provides all the necessary information for the plant specific application of the VIPRE-D/BWU code/correlation to North Anna. This section lists the applicable Deterministic Design Limits (DDLs), Statistical Design Limits (SDLs) and Safety Analysis Limits (SALs), as well as the corresponding Retained Margin. The verification of the existing Reactor Core Safety Limits, Protection Setpoints and Chapter 15 events with the above DNBR limits is also documented in Section 3.2.

3.0 TECHNICAL EVALUATION

3.1 Implementation of the Statistical DNBR Evaluation Methodology

3.1.1 Methodology Review

In Appendix A to Topical Report DOM-NAF-2 (Reference 2), Dominion calculated a deterministic DNBR Limit for the VIPRE-D/BWU code/correlation pair. The Statistical DNBR Evaluation Methodology (Reference 1) is employed herein to determine a statistical DNBR limit. This new limit combines the correlation uncertainty with the DNBR sensitivities to uncertainties in key DNBR analysis input parameters. Even though the new DNBR limit (the Statistical Design Limit or SDL) is larger than the deterministic code/correlation limit (DDL), its use is advantageous as the Statistical DNBR Evaluation Methodology instructs analysts not to apply the evaluated uncertainties to the initial conditions for statepoint and transient analysis. Instead, nominal values are used.

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

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

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

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

3.1.2. Uncertainty Analysis

This section is included herein to satisfy Condition 2 in the SER of the Statistical DNBR Evaluation Methodology Topical Report (Reference 1).

Consistent with the Statistical DNBR Evaluation Methodology Topical Report (Reference 1), inlet temperature, pressurizer pressure, core thermal power, vessel flow rate, core bypass flow, the nuclear enthalpy rise factor and the engineering enthalpy rise factor were selected as the statistically treated parameters in the implementation analysis. In addition, the local and bundle spacing uncertainty, which is specific of AREVA AMBW fuel was considered. The magnitudes and functional forms of the uncertainties for the statistically treated parameters were derived in a rigorous analysis of plant hardware and measurement/calibration procedures, and have been summarized in Table 3.1.2-1.

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

Dominion has quantified the magnitude and distribution of uncertainty on the pressurizer pressure (system pressure) per the pressurizer pressure control system. The pressurizer pressure uncertainty was quantified as normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 3.30% of span or $\pm$ 26.4 psia. The impact of parameter surveillance was considered. The current parameter surveillance limit for pressurizer pressure of 2205 psig was determined to be acceptable. With this parameter surveillance limit, the pressurizer pressure uncertainty was conservatively defined as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 30 psia and a standard deviation (σ) of 15.306 psia.

Dominion has quantified the magnitude and distribution of uncertainty on the average temperature (Tavg) per the Tavg rod control system. The average temperature uncertainty was quantified as a normal, two-sided, 95% probability distribution with a magnitude of $\pm 3.26^{o}$F. The impact of parameter surveillance was considered. The current parameter surveillance limit for average temperature of 591.0^{o}F was determined to be acceptable. With this parameter surveillance limit, the average temperature uncertainty is conservatively defined as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 4.2°F ($\pm$ 1.96σ) and a standard deviation (σ) of 2.143°F.

Dominion has quantified the uncertainty on core power as measured by the secondary side heat balance as 1.390% at uprated power 2942.2 MWt. This parameter uncertainty is treated as a normal, two-sided, 95% probability distribution and its standard deviation was calculated by dividing this value by 1.96 to obtain 0.709%. The standard deviation used for the implementation of the Statistical DNBR Evaluation Methodology was 0.771%, which includes additional conservatism to allow for future changes in plant hardware or calibration procedures without invalidating the analysis.

Dominion has quantified the uncertainty on the reactor coolant system (RCS) flow as 2.6048%. This parameter uncertainty is treated as a normal, two-sided, 95% probability distribution and its standard deviation was calculated by dividing this value by 1.96 to obtain 1.329%. The standard deviation used for the implementation of the Statistical DNBR Evaluation Methodology was 1.46%, which includes additional conservatism to allow for future changes in plant hardware or calibration procedures without invalidating the analysis.

The two-sided, 95/95 tolerance interval (95% probability, 95% confidence) for the measurement uncertainty of the nuclear enthalpy rise factor, $F_{\Delta H}^{N}$, is 3.4%. Conservatively, the measured $F_{\Delta H}^{N}$ uncertainty was defined as a normal distribution with a 4% tolerance interval for consistency with previous applications (Reference 14).

AREVA has quantified the magnitude and distribution of uncertainty on the engineering hot channel factor, $F_{\Delta H}^{E}$, as it is a fuel specific parameter. The $F_{\Delta H}^{E}$ uncertainty was quantified as a normal probability distribution with a magnitude of $\pm$ 3.0%. The Statistical DNBR Evaluation Methodology (Reference 1) treats the $F_{\Delta H}^{E}$ uncertainty as a uniform probability distribution and a uniform probability distribution was incorporated for the implementation of VEP-NE-2-A for Westinghouse fuel at North Anna (Reference 14). For the implementation analysis documented herein, the engineering hot channel factor uncertainty was defined as a uniform probability distribution with a magnitude of $\pm$ 3.0%. It is noted that the uncertainty on the engineering hot channel factor is a multiplicative factor on the hot rod average power; therefore, it affects the surface heat flux of the hot rod as well as the enthalpy rise in the adjacent subchannels.

AREVA provided the magnitude and distribution of uncertainty on the bundle spacing factor. This uncertainty is related to the manufacturing tolerances of the AREVA AMBW fuel and to the evaluation methodology for fuel rod bowing. For the implementation analysis documented herein, the bundle spacing uncertainty was defined as a uniform probability distribution with a magnitude of $\pm$ 1.5%. It is noted that the uncertainty on the bundle spacing factor is a multiplicative factor on the hot rod average power; therefore, it affects the surface heat flux of the hot rod as well as the enthalpy rise in the adjacent subchannels.

The total core bypass flow consists of separate flow paths through the thimble tubes, direct leakage to the outlet nozzle, baffle joint leakage flow, upper head spray flow and core-baffle gap flow. These five components were each quantified based on the current North Anna core configuration, their uncertainties conservatively modeled and the flows and uncertainties totaled. The Monte Carlo analysis ultimately used a best estimate bypass flow of 5.5% with an uncertainty of 1.0%, both of which are larger than the calculated values. The implementation analysis assumed that the probability was uniformly distributed. In addition, no credit was taken for independence of any of the bypass flow uncertainties.

Table 3.1.2-1: North Anna Parameter Uncertainties

PARAMETER	NOMINAL VALUE	STANDARD DEVIATION	UNCERTAINTY	DISTRIBUTION
Pressure [psia]	2250	15.306 psia	±30.0 psia at 2σ	Normal
Temperature [°F]	553.7	2.143 °F	±4.2 °F at 2σ	Normal
Power [MW]	2942.2[1]	0.771%	±1.511% at 2σ	Normal
Flow [gpm]	295,000	1.46%	±2.862% at 2σ	Normal
$F_{\Delta H}^{N}$	1.587[2]	2.0%	±4.0% at 2σ	Normal
$F_{\Delta H}^{E}$	1.0	N/A	±3.0%	Uniform
Local & Bundle Spacing	1.0	N/A	±1.5%	Uniform
Bypass [%]	5.5	N/A	±1.0%	Uniform

1 The implementation of the Statistical DNBR Evaluation Methodology was performed assuming a 1.7% power uprate above the current North Anna nominal power of 2893 MWt. However, this value bounds current conditions.
2 The implementation of the Statistical DNBR Evaluation Methodology was performed assuming a maximum statistical $F_{\Delta H}^{N}$ equal to 1.587, which is not the current North Anna COLR value. However, this value bounds the current 1.49 value.

3.1.3. CHF Correlations

There are two BWU CHF correlations used for the calculation of DNBRs in AREVA Advanced Mark-BW fuel assemblies. BWU-N is only applicable in the presence of non-mixing grids, and BWU-Z is the enhanced mixing grid correlation approved for the Advanced Mark-BW fuel assembly design.

- BWU-N is used from the beginning of the heated length to the leading edge of the first structural mixing grid.
- BWU-Z is used from the leading edge of the first structural mixing grid to the leading edge of the second structural mixing grid.
- BWU-Z with a multiplicative performance factor (this enhanced form of BWU-Z is normally referred to as BWU-ZM) is used from the leading edge of the second structural mixing grid to the leading edge of the last structural mixing grid.
- For the uppermost span, in which the end of heated length occurs less than one grid span beyond the last mixing grid, the BWU-Z correlation should be used with a grid spacing equal to the effective grid spacing (the distance from the last grid to the end of heated length).

As a consequence, two SDLs were calculated, one for BWU-Z/ZM and another for BWU-N. BWU-Z and BWU-ZM are exactly the same correlation except for the multiplicative performance factor that is applied to BWU-ZM to correct for the thermal-hydraulic performance improvement due to the Mid-Span Mixing Grids. Because additional experimental tests had to be performed to qualify BWU-ZM, the correlation results in a slightly larger code/correlation uncertainty. However, since they follow the same equation, both correlations result in the same overall statistics, and it is appropriate to obtain an SDL applicable to both of them. In this implementation, BWU-ZM code/correlation uncertainties were used to obtain the BWU-Z/ZM SDL, because they are slightly more conservative.

3.1.4. Model Uncertainty Term

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

The VIPRE-D 14-channel production model for North Anna was used in the development of the VIPRE-D/BWU SDL for North Anna. Since this is the production model that Dominion intends to use for all North Anna evaluations once the Topical Report DOM-NAF-2 and Appendix A are approved, and the VIPRE-D code is added to the Technical Specification 5.6.5.b list of NRC approved methodologies used to determine core operating limits [i.e. the reference list of the North Anna Core Operating Limits Report (COLR)], there is no additional uncertainty associated with the use of this model. This is in contrast to the COBRA/WRB-1 implementation analysis for North Anna (Reference 14) which due to computer time restrictions employed a simplified model to develop the SDL instead of the more complex production model. In summary, it is concluded that no correction for model uncertainty is necessary, and the model uncertainty term is set to zero for the calculation of the total DNBR standard deviation.

3.1.5 Code Uncertainty

The code uncertainty accounts for any differences between Dominion's VIPRE-D and AREVA's LYNXT, with which the BWU CHF data were correlated, and any effect due to the modeling of a full core with a correlation based upon bundle test data. These uncertainties are clearly independent of the correlation, the model and parameter induced uncertainties. The code uncertainty was quantified at 5%, consistent with the factors specified for other thermal/hydraulic codes in Reference 1. The basis for this uncertainty is described in detail by NRC staff in Reference 13. Therein, the NRC Staff refers to the 5% uncertainty as being a 2σ value. The 5% code uncertainty is certainly conservative in light of the excellent VIPRE-D/LYNXT and VIPRE-D/CHF data comparisons. However, the 5% uncertainty serves as a conservative factor that may be shown to be wholly or partially unnecessary at a later time. A one-sided 95% confidence level on the code uncertainty is then 3.04% (=5.0%/1.645). The use of the 1.645 divisor (the one-sided 95% tolerance interval multiplier) is conservative since the NRC Staff considers the 5% uncertainty as being a 2σ value.

3.1.6. Monte Carlo Calculations

In order to perform the Monte Carlo analysis nine (BWU-Z/ZM) and sixteen (BWU-N) Nominal Statepoints covering the full range of normal operation and anticipated transient conditions were selected. These conditions spanned the pressure range between the high and low trip setpoints, inlet temperatures between normal operation and maximum heat-up, powers up to the 118% overpower limit and a bounding low flow event. The selected Nominal Statepoints are listed in Tables 3.1.6-1 and 3.1.6-2.

The Monte Carlo analysis itself consisted of 2000 calculations performed around each of the nine (BWU-Z/ZM) and sixteen (BWU-N) Nominal Statepoints. As described in Section 3.1, the DNBR standard deviation at each Nominal Statepoint was augmented by the code/correlation uncertainty, the small sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation. The limiting pin peak SDL was calculated to be 1.31 (1.3073) for VIPRE-D/BWU-Z/ZM and 1.32 (1.3153) for VIPRE-D/BWU-N. The Monte Carlo Statepoint analysis is summarized in Tables 3.1.6-3 and 3.1.6-4 for BWU-Z/ZM and BWU-N respectively.

TABLE 3.1.6-1: Nominal Statepoints for AREVA Advanced Mark-BW Fuel with BWU-Z/ZM

STATE POINT	PRESSURE [psia]	INLET TEMPERATURE [°F]	POWER [%]	FLOW [%]	$F_{\Delta H}^{N}$ [3]	AXIAL OFFSET [%]	MDNBR
A	2400.0	624.5	98	100	1.597	0.0	1.313
B	2400.0	592.7	118	100	1.587	0.0	1.310
C	2250.0	609.7	104	100	1.587	0.0	1.312
D	2250.0	585.3	118	100	1.587	0.0	1.316
E	2000.0	592.0	110	100	1.587	0.0	1.310
F	2000.0	576.3	118	100	1.587	0.0	1.314
G	1860.0	581.2	114	100	1.587	0.0	1.312
H	1860.0	573.0	118	100	1.587	0.0	1.311
I	2250.0	553.7	100	62	1.587	0.0	1.315

TABLE 3.1.6-2: Nominal Statepoints for AREVA Advanced Mark-BW Fuel with BWU-N

STATE POINT	PRESSURE [psia]	TEMPERATURE [°F]	POWER [%]	FLOW [%]	$F_{\Delta H}^{N}$ [3]	AXIAL OFFSET [%]	MDNBR
A	2400.0	624.5	80	100	1.682	-48.0	1.402
B	2400.0	584.5	118	100	1.587	-48.0	1.403
C	2250.0	616.5	80	100	1.682	-48.0	1.409
D	2250.0	575.5	118	100	1.587	-48.0	1.404
E	2000.0	603.0	80	100	1.682	-48.0	1.400
F	2000.0	559.0	118	100	1.587	-48.0	1.404
G	1860.0	595.0	80	100	1.682	-48.0	1.407
H	1860.0	549.5	118	100	1.587	-48.0	1.407
I	2250.0	551.5	100	64	1.587	-48.0	1.404
A1	2400.0	639.0	80	100	1.682	-32.5	1.403
C1	2250.0	632.5	80	100	1.682	-32.5	1.407
E1	2000.0	620.5	80	100	1.682	-32.5	1.408
G1	1860.0	614.5	80	100	1.682	-32.5	1.407
A2	2400.0	645.5	80	100	1.682	-24.8	1.401
C2	2250.0	639.5	80	100	1.682	-24.8	1.404
H2	1860.0	581.0	118	100	1.587	-24.8	1.401

3 The part-power multiplier described in the North Anna Technical Specifications is used for less than 100% power statepoints.

Table 3.1.6-3: Peak Pin SDL Results for BWU-Z/ZM

STATEPOINT	Randomized DNB s_{DNBR}	Total DNB s_{TOTAL}	Pin Peak $SDL_{95/95}$
A	0.1530	0.1690	1.2780
B	0.1697	0.1868	**1.3073**
C	0.1540	0.1700	1.2797
D	0.1653	0.1821	1.2996
E	0.1548	0.1709	1.2811
F	0.1574	0.1737	1.2858
G	0.1497	0.1655	1.2722
H	0.1554	0.1716	1.2822
I	0.1601	0.1766	1.2904

Table 3.1.6-4: Peak Pin SDL Results for BWU-N

STATEPOINT	Randomized DNBR s_{DNBR}	Total DNBR s_{TOTAL}	Pin Peak $SDL_{95/95}$
A	0.1819	0.1917	**1.3153**
B	0.1734	0.1830	1.3010
C	0.1788	0.1885	1.3101
D	0.1734	0.1830	1.3010
E	0.1755	0.1851	1.3045
F	0.1766	0.1863	1.3064
G	0.1752	0.1849	1.3041
H	0.1758	0.1854	1.3050
I	0.1713	0.1808	1.2974
A1	0.1766	0.1863	1.3064
C1	0.1733	0.1828	1.3008
E1	0.1770	0.1866	1.3070
G1	0.1729	0.1824	1.3001
A2	0.1779	0.1876	1.3086
C2	0.1757	0.1853	1.3048
H2	0.1763	0.1860	1.3059

3.1.7. Full Core DNB Probability Summation

After the development of the pin peak 95/95 DNBR limits for both BWU-Z/ZM and BWU-N, the data statistics were used to determine the number of rods expected in DNB. The DNB probability summation for VIPRE-D/BWU-Z/ZM is summarized in Table 3.1.7-1. As it may be seen, in order to meet the 99.9% criterion it was necessary to increase the 95/95 pin peak SDL limit to 1.34 for VIPRE-D/BWU-Z/ZM. The DNB probability summation for VIPRE-D/BWU-N is summarized in Table 3.1.7-2. In order to meet the 99.9% criterion it was necessary to increase the 95/95 pin peak SDL limit to 1.38 for VIPRE-D/BWU-N. The full core DNB probability summation will be reevaluated on a reload basis to verify the applicability of the conservative reference fuel rod census ($F_{\Delta H}^N$ versus % of core with the same of greater $F_{\Delta H}^N$ or rod power) used in the implementation analysis.

Table 3.1.7-1: Full Core DNB Probability Summation for BWU-Z/ZM

STATEPOINT	S_{TOTAL}	% of Rods in DNB	Full Core $SDL_{99.9}$
A	0.1690	0.089	1.32
B	0.1868	0.092	**1.34**
C	0.1700	0.089	1.32
D	0.1821	0.096	1.33
E	0.1709	0.092	1.32
F	0.1737	0.093	1.32
G	0.1655	0.091	1.31
H	0.1716	0.093	1.32
I	0.1766	0.098	1.32

Table 3.1.7-2: Full Core DNB Probability Summation for BWU-N

STATEPOINT	S_{TOTAL}	% of Rods in DNB	Full Core $SDL_{99.9}$
A	0.1917	0.091	**1.38**
B	0.1830	0.090	1.37
C	0.1885	0.095	1.37
D	0.1830	0.100	1.36
E	0.1851	0.087	1.37
F	0.1863	0.097	1.37
G	0.1849	0.088	1.37
H	0.1854	0.094	1.37
I	0.1808	0.089	1.36
A1	0.1863	0.092	1.37
C1	0.1828	0.097	1.36
E1	0.1866	0.086	1.38
G1	0.1824	0.087	1.37
A2	0.1876	0.095	1.37
C2	0.1853	0.090	1.37
H2	0.1860	0.092	1.37

3.1.8. Verification of Nominal Statepoints

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

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

The method outlined above was followed to determine that the set of Nominal Statepoints selected for both the BWU-Z/ZM and BWU-N CHF correlations were in fact bounding for any set of conditions to which the methodology may be applied. All the regression analyses performed for each independent variable showed extremely low R^2 correlation factors, which indicates that the unrandomized DNBR standard deviations are not related to the independent variables evaluated. This substantiates the fact that the DNBR standard deviation has been conservatively maximized for any conceivable Condition I, Condition II or low flow DNB event. Figures 3.1.8-1 and 3.1.8-2 display two sample regression plots for both BWU-Z/ZM and BWU-N and clearly show the trends discussed above.

Figure 3.1.8-1: Variation of the Standard Deviation of the Unrandomized DNBRs with Inlet Temperature for the BWU-Z/ZM CHF Correlation

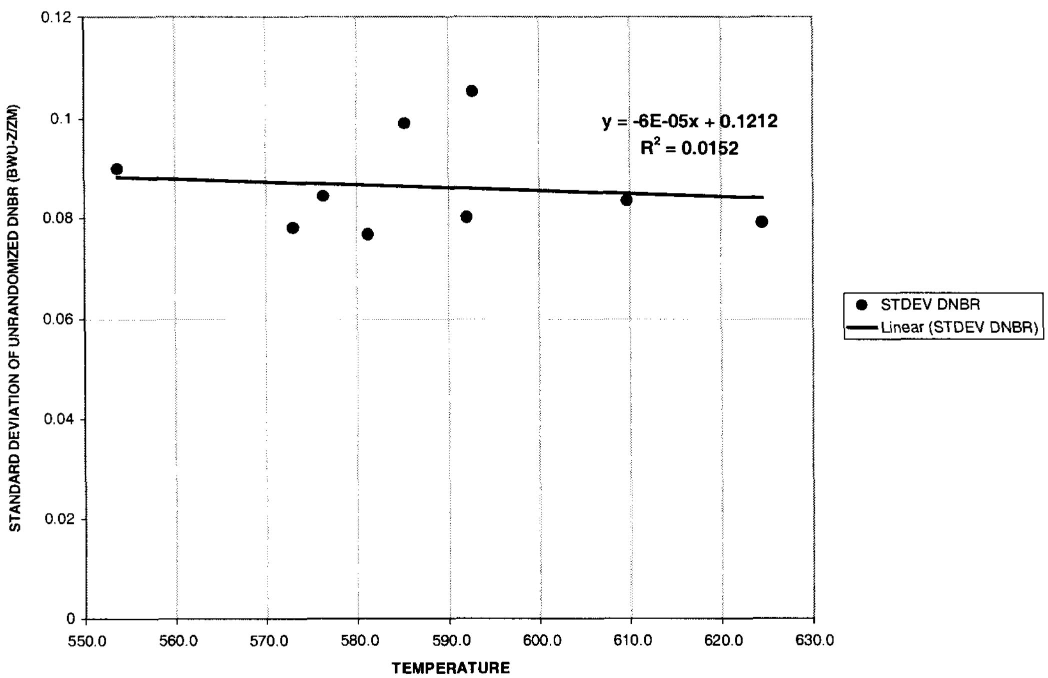
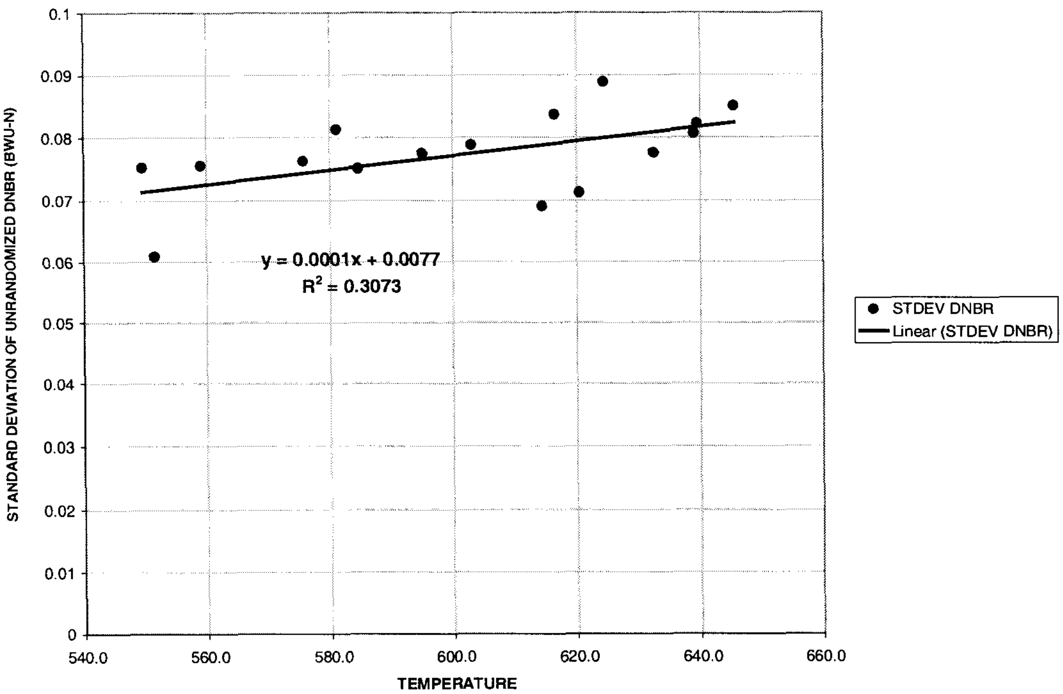


Figure 3.1.8-2: Variation of the Standard Deviation of the Unrandomized DNBRs with Inlet Temperature for the BWU-N CHF Correlation



3.1.9. Scope of Applicability

The Statistical DNBR Evaluation Methodology may be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWSC), and to the Loss of Flow analysis, the Locked Rotor Accident and the Single Rod Cluster Control Assembly Withdrawal at Power, SRWAP. The accidents to which the methodology is applicable are listed in Table 3.1.9-1 (This table corresponds to Table 2.1-1 in Reference 2). The range of application is consistent with previous applications of the Statistical DNBR Evaluation Methodology to North Anna (Reference 14). This methodology will not be applied to accidents that begin from zero power where the parameter uncertainties are higher.

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

Table 3.1.9-1: UFSAR Transients Analyzed with VIPRE-D/BWU for North Anna

ACCIDENT	UFSAR SECTION	APPLICATION
Rod cluster control assembly bank withdrawal from subcritical	15.2.1	DET-DNB
Rod cluster control assembly bank withdrawal at power	15.2.2	STAT-DNB
Rod cluster control assembly misalignment / Dropped rod/bank	15.2.3	STAT-DNB
Uncontrolled boron dilution	15.2.4	non-DNB
Partial loss of forced reactor coolant flow	15.2.5	STAT-DNB
Startup of an inactive reactor coolant loop	15.2.6	STAT-DNB
Loss of external electrical load and/or turbine trip	15.2.7	STAT-DNB
Loss of normal feedwater	15.2.8	STAT-DNB
Loss of offsite power	15.2.9	STAT-DNB
Excessive heat removal due to feedwater system malfunction	15.2.10	STAT-DNB
Excessive load increase	15.2.11	STAT-DNB
Accidental depressurization of the reactor cooling system	15.2.12	STAT-DNB
Accidental depressurization of the main steam system	15.2.13	DET-DNB
Inadvertent operation of emergency core cooling system during power operation	15.2.14	STAT-DNB
Complete loss of flow	15.3.4	STAT-DNB
Single rod cluster control assembly withdrawal at full power	15.3.7	STAT-DNB
Rupture of a main steam pipe	15.4.2.1	DET-DNB
Major rupture of a main feedwater pipe	15.4.2.2	non-DNB
Locked reactor coolant pump rotor or shaft break	15.4.4	STAT-DNB

3.1.10. Summary of Analysis

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

For BWU-Z/ZM, at the limiting Nominal Statepoint (B), the standard deviation of the randomized DNBR distributions was found to be 0.1697. This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation of 0.1868, as listed in Table 3.1.6-3. The use of this number in Equation 3.1 yields a pin peak DNBR limit of 1.3073 with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core, which increased the SDL to 1.34.

For BWU-N, at the limiting Nominal Statepoint (A), the standard deviation of the randomized DNBR distribution was found to be 0.1819. This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation of 0.1917, as listed in Table 3.1.6-4. The use of this number in Equation 3.1 yields a pin peak DNBR limit of 1.3153 with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core, which increased the SDL to 1.38.

3.2. Application of VIPRE-D/BWU to North Anna Power Station

Table 3.1.9-1 satisfies Condition 4 in DOM-NAF-2, as it lists the North Anna UFSAR events for which the VIPRE-D code with the BWU CHF correlations will be applied.

VIPRE-D/BWU together with the Statistical DNBR Evaluation Methodology will be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWSC), and to the Loss of Flow analysis, the Locked Rotor Accident and the Single Rod Cluster Control Assembly Withdrawal at Power, SRWAP. The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}{}^{N}$ (measurement component) and $F_{\Delta H}{}^{E}$ and bundle spacing uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

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

3.2.1 VIPRE-D/BWU Deterministic Design Limits (DDL)

Appendix A of Topical Report DOM-NAF-2 (Reference 2) documents the qualification of the AREVA BWU CHF correlations with the VIPRE-D computer code. This document lists the DNBR deterministic design limits for VIPRE-D/BWU-Z, VIPRE-D/BWU-ZM and VIPRE-D/BWU-N that yield a 95% non-DNB probability at a 95% confidence level. These DDLs were obtained for the VIPRE-D/BWU code/correlation pair, and are independent of the specific plant. These limits are applicable to the analysis of deterministic events of AREVA AMBW fuel in North Anna cores with the VIPRE-D/BWU code.

Table 3.2.1-1: VIPRE-D DNBR Deterministic Design Limits
for BWU-Z, BWU-ZM and BWU-N (Reference 2)

VIPRE-D/BWU-Z	
DNBR limit below 700 psia	1.59
DNBR limit 700 – 2,400 psia	1.20
VIPRE-D/BWU-ZM	
DNBR limit below 594 psia	1.59
DNBR limit at or above 594 psia	1.18
VIPRE-D/BWU-N	
DNBR limit below 1200 psia	1.39
DNBR limit at or above 1200 psia	1.22

3.2.2 VIPRE-D/BWU Statistical Design Limits (SDL) for North Anna

The Statistical Design Limits for North Anna cores containing AREVA AMBW fuel assemblies with the VIPRE-D/BWU code were derived in Section 3.1 of this report. The SDL for VIPRE-D/BWU-Z/ZM is 1.34. The SDL for VIPRE-D/BWU-N is 1.38. These limits provide a peak rod DNB protection with at least 95% probability at a 95% confidence level and a 99.9% DNB protection for the full core. These SDLs are plant specific as they already include the North Anna specific uncertainties for the key parameters accounted for in the application of the Statistical DNBR Evaluation Methodology. Therefore, these limits are applicable to the analysis of statistical DNB events of AREVA AMBW fuel in North Anna cores with the VIPRE-D/BWU code.

3.2.3. Safety Analysis Limits (SAL)

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

For deterministic DNB analyses, the design DNBR limit is set equal to the applicable code/correlation limit and it is termed the deterministic design limit (DDL). For statistical DNB analyses, the design DNBR limit is set equal to the applicable statistical design limit (SDL). These design limits are two of the Design Basis Limits for Fission Product Barriers (DBLFPB) described in Reference 17. The DDLs and SDLs are fixed and any changes to their value require NRC review and approval. However, the safety analysis limits for deterministic and statistical DNB analyses (SAL_{DET} and SAL_{STAT}, respectively) are self-imposed and may be changed without prior NRC review and approval, provided the changes meet the criteria established in Reference 17.

A deterministic and statistical SAL equal to 1.60 has been selected for AREVA AMBW fuel at North Anna cores with the VIPRE-D code and all the AREVA BWU CHF correlations. This SAL is applicable for all deterministic analyses for a maximum peaking factor $F_{\Delta H}^N$ equal to 1.65 and for all statistical analyses for a maximum peaking factor $F_{\Delta H}^N$ equal to 1.587. The only exception is the deterministic SAL for BWU-Z/ZM at pressures lower than 700 psia that was selected to be 1.85 in order to provide sufficient retained margin to accommodate the transition core penalty for a second mixed core. It must be noted, however, that currently there are no evaluated events that are limiting in this region.

3.2.4. Retained Margin

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

$$\text{Retained Margin} = \left[\frac{\text{Safety Analysis Limit - Design Limit}}{\text{Safety Analysis Limit}}\right]\cdot 100$$

The resulting available retained margins are listed in Tables 3.2.4-1 and 3.2.4-2.

Table 3.2.4-1: DNBR Limits and Retained Margin for Deterministic DNB Applications at North Anna

DETERMINISTIC DNB APPLICATIONS				
DNB CORRELATION	**PRESSURE**	**DDL**	**SAL_{DET}**	**RETAINED MARGIN [%]**
BWU-Z/ZM	< 700 psia	1.59	1.85[4]	14.0
	700 – 2,400 psia	1.20	1.60	25.0
BWU-N	< 1200 psia	1.39	1.60	13.1
	≥ 1200 psia	1.22	1.60	23.7

4 There are no known deterministic events that would apply the BWU-Z/ZM correlation below 700 psia, but to accommodate the transition core penalties for the 2nd mixed cores an increased SAL has been selected.

Table 3.2.4-2: DNBR Limits and Retained Margin for Statistical DNB Applications at North Anna

STATISTICAL DNB APPLICATIONS			
DNB CORRELATION	**SDL**	**SAL_{STAT}**	**RETAINED MARGIN [%]**
BWU-Z/ZM	1.34	1.60	16.2
BWU-N	1.38	1.60	13.7

This method of defining retained DNBR margin allows all the margin to be found in a single, clearly defined location. The retained DNBR margin can be used to offset generic DNBR penalties, which may be difficult to model mechanistically in the DNBR analysis calculations, such as the NAIF/Advanced Mark-BW transition core penalty.

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

3.2.5 Transition Core Penalties

The AREVA Design Report (Reference 3) documents the transition core DNBR penalties for the BWU-Z/ZM and BWU-N CHF correlations for application to the AREVA AMBW fuel product in mixed-core configurations at North Anna. These transition core penalties are listed in Table 3.2.5-1.

Table 3.2.5-1: North Anna AREVA AMBW Transition Core Penalties

	FIRST CORE (N2C17 & N1C18)	**SECOND CORE (N2C18 & N1C19)**
BWU-Z/ZM	19.8%	12.8%
BWU-N	0.7%	0.7%

N1C18 and N2C17 are the cores currently loaded in North Anna Power Station, Units 1 and 2 respectively. N1C19 and N2C18, which are second core mixed configurations, will be loaded into North Anna Power Station, Units 1 and 2 during the Spring 2006 and the Fall 2005 refueling outages respectively. Given the submittal date of this report, it is unlikely that the License Amendment Request included herein will be reviewed and approved by the NRC before the reload safety analyses are initiated for the second core mixed configuration. However, it can be shown that the SALs and SDLs documented herein provide enough retained margin to support a second core configuration. Finally, the SDLs and SALs documented herein certainly provide sufficient retained margin for a third core configuration, which will most probably be a full core of AREVA AMBW assemblies that will not carry any transition core penalties.

3.2.6 Verification of Existing Reactor Core Safety Limits, Protection Setpoints and Chapter 15 Events

According to condition 3 in DOM-NAF-2, Dominion would submit to the NRC any Technical Specification changes related to Reactor Core Safety Limits or the OTΔT, OPΔT and FΔI trip setpoints as a consequence of revised DDLs or SDLs. This section shows that no changes to the Reactor Core Safety Limits or the OTΔT, OPΔT and FΔI trip setpoints are necessary as a consequence of the implementation of the Statistical DNBR Evaluation Methodology to AREVA AMBW fuel at North Anna with VIPRE-D/BWU, as the existing Reactor Core Safety Limits and the OTΔT, OPΔT and FΔI trip setpoints are fully supported by the DDLs, SDLs and SALs listed in sections 3.2.1, 3.2.2 and 3.2.3.

To demonstrate that the DNB performance of the Advanced Mark-BW fuel is acceptable, Dominion performed calculations for full-core configurations of AREVA AMBW fuel. The calculations were performed using the VIPRE-D/BWU code/correlation pair and the statepoint conditions defined for Reference 3 to demonstrate acceptable DNB performance for the AREVA AMBW fuel. The selected statepoints include the core thermal limits (CTLs), axial offset envelopes (AOs), rod withdrawal at power (RWAP), rod withdrawal from subcritical (RWSC), control rod misalignment, rod urgent failure, main steam line break (MSLB), loss of flow (LOFA), and locked rotor events (LOCROT). These various statepoints provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power FΔh), pressure and temperature (CTLs); (b) limiting axial flux shapes at several axial offsets (AOs); (c) elevated hot rod power (misaligned rod); and (d) low flow (LOFA and LOCROT). The statepoints for the RWSC and MSLB were evaluated with deterministic DNB methods. The remaining statepoints were evaluated using statistical DNB methods. The evaluation criterion for these analyses is that the minimum DNBR must be equal to or greater than the applicable safety analysis limit (Section 3.2.3).

As noted in Reference 3, the ultimate goal of the thermal-hydraulic analysis was to support a full-power radial power distribution factor ($F_{\Delta H}^N$) limit of 1.587 for reload cores that include the Advanced Mark-BW fuel. Thus, the statepoint conditions for the Advanced Mark-BW included $F_{\Delta H}^N$ values for each condition that were scaled by the ratio of the ultimate full-power $F_{\Delta H}^N$ limit to the current full-power $F_{\Delta H}^N$ limit (i.e., 1.587 / 1.490). The exceptions to this rule were the nominal, rod urgent failure, and main steamline break statepoints which were evaluated with $F_{\Delta H}^N$ values that were equal to a bounding reload set of values. These are non-limiting conditions and it is sufficient to demonstrate acceptable performance with this bounding set.

The results of the calculations demonstrate that the minimum DNBR values are equal to or greater than the applicable safety analysis limit of 1.60 for all the Reactor Core Safety Limits, the OTΔT, OPΔT and FΔI trip setpoints, as well as all the evaluated Chapter 15 events (including the LOFA and LOCROT) with an $F_{\Delta H}^N$ of 1.587.

4.0 PROPOSED TECHNICAL SPECIFICATIONS CHANGES

Analysis of the Advanced Mark-BW fuel design at North Anna with Dominion specific approved tools and methods will require a revision to the existing plant Technical Specifications. This change is administrative in nature involving the addition of a reference that supports the Core Operating Limits Report (COLR). The specific proposed changes are provided below[5].

TS 5.6.5.b, CORE OPERATING LIMITS REPORT (COLR)

This section is revised to include an additional reference that reflects the proposed change above. The additional reference describes Dominion specific analytical methods used in determining core limits that are applicable to the Advanced Mark-BW fuel product. The following addition is proposed:

19. 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."

5.0 SAFETY SIGNIFICANCE SUMMARY

The VIPRE-01 code has been approved by the NRC and is widely used throughout the nuclear industry for PWR safety analyses. VIPRE-D is the Dominion version of VIPRE-01. Topical Report DOM-NAF-2, which documents the use of VIPRE-D for the thermal-hydraulic evaluation of nuclear reactor cores, is a document that is currently under review by the NRC. Dominion has validated VIPRE-D with extensive code benchmark calculations using the modeling methods outlined in DOM-NAF-2, and the accuracy of the VIPRE-D models has been demonstrated through comparisons with other NRC-approved methodologies, including AREVA's LYNXT/BWU. VIPRE-D includes the critical heat flux (CHF) correlations to be used for the evaluation of AREVA Advanced Mark-BW fuel type (BWU CHF Correlations). These correlations were documented and qualified in Appendix A of DOM-NAF-2 (Reference 2) and are currently under review by the NRC for use with such fuel product. The VIPRE-D BWU CHF correlations will be used within the NRC approved parameter ranges of the BWU CHF correlations, including fuel assembly geometry and grid spacers. The DNBR design limits applied to the BWU CHF correlations were derived using fluid conditions predicted by the VIPRE-D code. In summary, the proposed DOM-NAF-2 (Reference 2) describes a methodology that is fully applicable to reload design.

The application of DOM-NAF-2 is in conjunction with the implementation of VEP-NE-2-A (Reference 1), which has been used to calculate the Statistical Design Limits (SDLs) applicable to VIPRE-D/BWU for AREVA AMBW fuel at North Anna. Setpoint safety analysis evaluations have been performed to verify that the existing Reactor Core

[5] This Technical Specifications Change Request assumes the approval of the Topical Report DOM-NAF-2, which is currently under NRC review. Approval is anticipated to occur in October 2005.

Safety Limits and protection functions (OTΔT, OPΔT, FΔI, etc) continue to be applicable for the VIPRE-D/BWU code/correlation set and the newly calculated SDLs. All applicable Chapter 15 analyses were evaluated with the VIPRE-D/BWU code/correlation and the Statistical DNBR Evaluation Methodology, and they all were demonstrated to have acceptable results. In conclusion, the statepoint analysis is the basis for demonstrating the acceptability of the change.

6.0 REGULATORY ANALYSIS

6.1 Significant Hazards Consideration

Virginia Electric and Power Company (Dominion) plans to use the VIPRE-D/BWU computer code to perform thermal-hydraulic evaluations of AREVA Advanced Mark-BW fuel at North Anna Units 1 and 2 cores. VIPRE-D is the Dominion version of VIPRE-01, which has been customized to fit Dominion's plants and fuel products. The VIPRE-01 code has been approved by the NRC and is widely used throughout the nuclear industry for PWR safety analyses. Dominion has shown VIPRE-D compliance with the requirements of the NRC SERs regarding VIPRE-01 code applications. Dominion has validated VIPRE-D with extensive code benchmark calculations using approved modeling methods, and the accuracy of the VIPRE-D models has been demonstrated through comparisons with other NRC-approved methodologies, including AREVA's LYNXT/BWU. In particular, Dominion has shown that VIPRE-D results are essentially the same as LYNXT results, which is the code currently licensed to perform thermal-hydraulic analyses of AREVA Advanced Mark-BW fuel at North Anna cores. Finally, a statepoint evaluation for all applicable Chapter 15 analyses was performed with the VIPRE-D/BWU code/correlation and the Dominion Statistical DNBR Evaluation Methodology, and it resulted in acceptable results for all evaluated events.

It is concluded that the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is delineated below:

1. The probability of occurrence or the consequences of an accident previously evaluated are not significantly increased.

   Neither the code/CHF correlation pair nor the Statistical DNBR Evaluation Methodology make any contribution to the potential accident initiators and thus cannot increase the probability of any accident. Further, since both the deterministic and statistical DNBR limits meet the required design basis of avoiding DNB with 95% probability at a 95% confidence level, the use of the new code/correlation and Statistical DNBR Evaluation Methodology do not increase the potential consequences of any accident. Finally, the addition of a full core DNB design limit provides increased assurance that the consequences of a postulated accident which included radioactive release would be minimized because the overall number of rods in DNB would not exceed the 0.1% level. All the pertinent evaluations to be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable have been performed with VIPRE-D/BWU and found to be acceptable. The use of a different code/correlation pair will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. The use of the VIPRE-D/BWU code/correlation pair will not result in a measurable impact on normal operating plant releases, and will not increase the predicted radiological consequences of accidents postulated in the UFSAR. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created.

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

3. The margin of safety is not significantly reduced.

   North Anna Technical Specification 2.1 specifies that any DNBR limit established by any used code/correlation must provide at least 95% non-DNB probability at a 95% confidence level. The use of VIPRE-D/BWU with the SDLs listed in this package provides that protection, just as LYNXT/BWU and applicable SDLs did. The required DNBR margin of safety for the North Anna Nuclear units, which in this case is the margin between the 95/95 DNBR limit and clad failure, is therefore not reduced. Therefore, the margin of safety as defined in the Bases to the North Anna Units 1 and 2 Technical Specifications is not significantly reduced.

Based on the above information, the use of VIPRE-D/BWU to perform thermal-hydraulic analyses of AREVA Advanced Mark-BW fuel in North Anna cores will not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. It is concluded that the proposed use of the VIPRE-D/BWU code meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

6.2 Environmental Assessment

These Technical Specification changes to allow the use of the VIPRE-D/BWU computer code to perform thermal-hydraulics evaluations of AREVA Advanced Mark-BW fuel at North Anna meet the eligibility criteria for categorical exclusion from an environmental assessment set forth in 10 CFR 51.22(c)(9), as discussed below:

(i) The license condition and associated exemptions from the Code of Federal Regulations involve no Significant Hazards Consideration.

As discussed in the evaluation of the Significant Hazards Consideration (Section 6.1), the use of the VIPRE-D/BWU computer code and its applicable fuel design limits for DNBR to perform thermal-hydraulics evaluations of AREVA Advanced Mark-BW fuel at North Anna, will not involve a significant increase in the probability or consequences of an accident previously evaluated. The possibility of a new or different kind of accident from any accident previously evaluated is also not created, and the proposed use of VIPRE-D/BWU does not involve a significant reduction in a margin of safety. Therefore, the proposed use of the VIPRE-D/BWU computer code and its applicable fuel design limits for DNBR meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

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

The use of the VIPRE-D/BWU computer code and its applicable fuel design limits for DNBR to perform thermal-hydraulics evaluations of AREVA Advanced Mark-BW fuel at North Anna does not affect the way in which the fuel is handled, operated, and stored. The performance of VIPRE-D/BWU has been benchmarked against LYNXT/BWU, and it provides results than are essentially the same. All applicable Chapter 15 analyses have been evaluated with the VIPRE-D/BWU code/correlation and the Statistical DNBR Evaluation Methodology, and they all resulted in acceptable results. Therefore analyses performed with VIPRE-D/BWU as part of the cycle specific reload evaluation, will continue to ensure the integrity of the cladding as a fission product barrier for the planned operating conditions. There will be no measurable increase in the isotopic levels in the coolant associated with use of VIPRE-D/BWU and its applicable fuel design limits for DNBR for the analysis of AREVA AMBW fuel, and so there is no effect on normal operating plant releases. It is concluded that the existing radiological consequences analyses for LYNXT/BWU remain applicable for VIPRE-D/BWU. Therefore, the use of the VIPRE-D/BWU computer code and its applicable fuel design limits for DNBR to perform thermal-hydraulics evaluations of AREVA Advanced Mark-BW fuel at North Anna will not significantly change the types, or significantly increase the amounts, of effluents that may be released offsite.

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

The use of the VIPRE-D/BWU computer code and its applicable fuel design limits for DNBR to perform thermal-hydraulics evaluations of AREVA Advanced Mark-BW fuel at North Anna does not affect the way in which the fuel is handled, operated, and stored. The use of the VIPRE-D/BWU computer code will not significantly affect the plant operating conditions. Cycle specific reload evaluations will verify that fuel rod design criteria are satisfied, ensuring that cladding integrity is maintained. The use of the VIPRE-D/BWU computer code will not significantly increase radiation levels compared to the thermal-hydraulics codes used currently, so individual and cumulative occupational exposures are unchanged.

Based on the above, the proposed use of the VIPRE-D/BWU computer code to perform thermal-hydraulics evaluations of AREVA Advanced Mark-BW fuel at North Anna does not have a significant effect on the environment, and meets the criteria of 10 CFR 51.22(c)(9). Therefore, the proposed Technical Specification changes qualify for a categorical exclusion from a specific environmental review by the Commission, as described in 10 CFR 51.22.

7.0 CONCLUSIONS

This amendment request includes all the information to review and approve the plant specific application of the VIPRE-D code with the AREVA BWU CHF correlations (VIPRE-D/BWU) for the thermal-hydraulic analysis of AREVA Advanced Mark-BW Fuel in North Anna Power Station cores.

Dominion's Statistical DNBR Evaluation Methodology has been used to derive a Statistical Design Limit (SDL) of 1.34 for the BWU-Z/ZM CHF correlation and the VIPRE-D code for AREVA AMBW fuel at North Anna. Similarly, a SDL of 1.38 has been derived for the BWU-N CHF correlation and the VIPRE-D code for AREVA AMBW fuel at North Anna. These limits provide peak rod DNB protection with at least 95% probability at a 95% confidence level and 99.9% DNB protection for the full core. A Safety Analysis Limit (SAL) has been selected for DNB analyses of North Anna cores containing AREVA AMBW fuel assemblies. The existing Reactor Core Safety Limits, OTΔT, OPΔT and FΔI trip setpoints as well as the current analyses of applicable UFSAR Chapter 15 events were shown to be bounding, and will not be changed. Dominion seeks the approval of the Statistical Design Limits (SDLs) documented herein as it is a Design Basis Limit for Fission Products Barrier (DBLFPB).

Finally, Dominion is seeking the approval for the inclusion of Topical Report DOM-NAF-2, including Appendix A, to the Technical Specification 5.6.5.b list of USNRC approved methodologies used to determine core operating limits [i.e. the reference list of the North Anna Core Operating Limits Report (COLR)]. This would allow Dominion the use of the VIPRE-D/BWU code to perform licensing calculations for AREVA AMBW fuel in North Anna cores, using the deterministic design limits (DDLs) qualified in Appendix A of Topical Report DOM-NAF-2, and the statistical design limits (SDLs) documented herein.

8.0 REFERENCES

1. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," R. C. Anderson, June 1987.
2. Letter from L. N. Hartz (Dominion) to Document Control Desk (NRC), "Virginia Electric and Power Company (Dominion) Dominion Nuclear Connecticut, Inc. (DNC) North Anna Power Station Units 1 and 2, Millstone Power Station Units 2 and 3, Surry Power Station Units 1 and 2, Request for Approval of Topical Report DOM-NAF-2 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," Serial No. 04-606, dated September 30, 2004.
3. Letter from L. N. Hartz (Dominion) to US NRC Document Control Desk "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," Serial No. 02-167, dated March 28, 2002. (Proprietary version).
4. Topical Report, BAW-10199P-A, "The BWU Critical Heat Flux Correlations," B&W Fuel Company, August 1996, including Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids", Framatome Cogema Fuels, November 2000.
5. Topical Report, BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores," B&W Fuel Company, December 1988.
6. Technical Report, EPRI NP-2511-CCM-A Volume 1, Revision 4, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 1: Mathematical Modeling," February 2001.
7. Technical Report, EPRI NP-2511-CCM Volume 2, Revision 4, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 2: User's Manual," February 2001.
8. Technical Report, EPRI NP-2511-CCM-A Volume 3, Revision 4, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 3: Programmer's Manual," February 2001.
9. Technical Report, EPRI NP-2511-CCM-A Volume 4, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 4: Applications," April 1987.
10. Technical Report, EPRI NP-2511-CCM Volume 5, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 5: Guidelines," March 1988.
11. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4," May 1, 1986.
12. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), "Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, 'VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores,' (TAC No. M79498)," October 30, 1993.

References (continued)

13. Letter from L. B. Engle (NRC) to W. L. Stewart (Virginia Power), "Statistical DNBR Evaluation Methodology, VEP-NE-2, Surry Power Station, Units No. 1 & No. 2 (Surry-1&2) and North Anna Power Station, Units No. 1 & No. 2 (NA-1&2)," Serial No. 87-335 dated May 28, 1987.
14. Letter from W. L. Stewart (VP) to Document Control Desk (NRC), "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 Proposed Technical Specifications Change," Serial No. 87-231, dated June 17, 1987.
15. Letter from L. B. Engle (NRC) to W. R. Cartwright (Virginia Power), "North Anna Units 1 and 2 - Approval of Continued Use of Negative Moderator Coefficient for NA-1 and Issuance of Amendment for NA-2 (TAC Nos. 71071 and 71072)," June 30, 1989.
16. Topical Report, VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," R. C. Anderson and N. P. Wolfhope, November 1986.
17. Technical Report, NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," Nuclear Energy Institute, November 2000.

ATTACHMENT 2

Serial No. 05-419

MARKED-UP AND PROPOSED TECHNICAL SPECIFICATIONS CHANGES

COLR – ANALYTICAL METHODOLOGY

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

14. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."

15. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."

16. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

INSERT 1

17. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."

18. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).

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.

5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

Insert 1

19. 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.”

Attachment 3

Serial No. 05-419

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

COLR – ANALYTICAL METHODOLOGY

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

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15. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."

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