VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 19, 2010

10CFR50.90

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

Serial No.	10-404
NL&OS/ETS	R0
Docket Nos.	50-338/339
License Nos.	NPR-4/7

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED LICENSE AMENDMENT REQUEST(LAR)
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 (TS) to Facility Operating License Numbers NPR-4 and NPR-7 for North Anna Power Station Units 1 and 2, respectively. The proposed LAR requests the inclusion of NRC approved Appendix C of Dominion Fleet Report DOM-NAF-2-A, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," to the Technical Specification 5.6.5.b, as a referenced analytical methodology.

Furthermore, plant specific application of the methodology requires approval of the Statistical Design Limit (SDL) for the relevant code/correlation pair. Consequently, in addition to the inclusion of Fleet Report DOM-NAF-2-A, including Appendix C, in TS 5.6.5.b, Dominion also requests NRC review and approval of the implementation of the Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for Westinghouse RFA-2 fuel at North Anna using the VIPRE-D/WRB-2M code/correlation pair, as well as the SDL obtained by this implementation. Pursuant to 10 CFR 50.59, the SDL discussed in Attachment 4 requires NRC review and approval since the SDL establishes a Design Basis Limit for Fission Product Barrier (DBLFPB).

Upon approval of this amendment request and the SDL documented in Attachment 4, Dominion will be capable of performing in-house DNB analyses for the intended uses described in DOM-NAF-2-A using VIPRE-D to support the use of Westinghouse RFA-2 fuel at North Anna Units 1 and 2. Dominion is currently planning to use Westinghouse RFA-2 fuel in North Anna Units 1 and 2 commencing with North Anna Unit 1, Cycle 23 (Spring 2012) and North Anna Unit 2, Cycle 23 (Spring 2013).

A discussion of the proposed changes is provided in Attachment 1. The marked-up and typed proposed TS pages are provided in Attachments 2 and 3, respectively. Attachment 4 provides the technical basis for: 1) adding Appendix C of Dominion Fleet Report DOM-NAF-2-A to the list of USNRC approved methodologies for determining core operating limits, 2) the implementation of the Dominion Statistical DNBR Evaluation Methodology for Westinghouse RFA-2 fuel at North Anna with the VIPRE-D/WRB-2M code/correlation pair, and 3) the SDL obtained by this implementation.

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

Approval of the proposed amendments is requested by July 21, 2011. Dominion also requests a 60-day implementation period following NRC approval of the requested license amendments.

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

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA

COUNTY OF HENRICO

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

Acknowledged before me this 19th day of July, 2010.

My Commission Expires: 4/30/13

Notary Public

I was commissioned a notary public as Ginger L. Alligood.

GINGER LYNN MELTON
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

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

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Avenue, NE
Suite 1200
Atlanta, Georgia 30303-1257

NRC Senior Resident Inspector
North Anna Power Station

Ms. K. R. Cotton
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

Dr. V. Sreenivas
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center, Suite 300
4201 Dominion Blvd.
Glen Allen, Virginia 23060

State Health Commissioner
Virginia Department of Health
James Madison Building – 7th Floor
109 Governor Street
Room 730
Richmond, Virginia 23219

ATTACHMENT 1

DISCUSSION OF CHANGE

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

DISCUSSION OF CHANGE

1.0 INTRODUCTION

Virginia Electric and Power Company (Dominion) proposes to modify the North Anna Power Station Technical Specifications to include Fleet Report DOM NAF-2-A, Appendix C, "Qualification of the Westinghouse WRB 2M Critical Heat Flux (CHF) Correlation in the Dominion VIPRE-D Computer Code," 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)}.

Approval of this change will allow Dominion to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse Robust Fuel Assembly – 2 (RFA-2) fuel in North Anna cores, using the deterministic design limits (DDLs) documented in Appendix C of the DOM-NAF-2-A Fleet Report and the statistical design limit (SDL) documented in Attachment 4. The DDLs were approved as part of the review and approval of DOM-NAF-2-A Appendix C (Reference 1). Dominion requests the NRC review and approval of the SDL documented herein, consistent with 10 CFR 50.59(c)(2)(vii) the change establishes a Design Basis Limit for Fission Product Barrier (DBLFPB).

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

2.0 BACKGROUND

Dominion plans to purchase fuel assemblies from Westinghouse for use at North Anna Power Station, Units 1 and 2. These assemblies are planned to be inserted in Units 1 and 2, commencing with Cycle 23 for both units. The Westinghouse 17x17 RFA-2 fuel product is a replacement for the resident fuel product, which is the AREVA Advanced Mark-BW (AMBW). The Westinghouse RFA-2 fuel product contains modified mid-grids and modified intermediate flow mixer grids (IFMs).

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 critical heat flux (CHF) and Departure from Nucleate Boiling Ratio (DNBR) of reactor cores. VIPRE-01 has been approved by the U. S. Nuclear Regulatory Commission (USNRC) (References 3 and 4). VIPRE-D is based upon VIPRE-01, MOD-02.1, and is customized by Dominion to fit the specific needs of Dominion's nuclear plants and fuel products.

In April 2006, Dominion obtained generic approval of Fleet Report DOM-NAF-2-A (Reference 1) from the USNRC. DOM-NAF-2-A provides the necessary documentation to describe the intended uses of VIPRE-D for PWR licensing applications. Appendix C documents the qualification of the WRB-2M CHF correlation with the VIPRE-D code and listed the DDLs for the WRB-2M and W-3 CHF correlations, which was approved generically by the USNRC in April 2009 (Reference 1). The implementation of this methodology on a site specific basis requires USNRC approval of site, fuel type and code specific DNBR design limits (i.e., SDL and/or DDLs) and a Technical Specification change to include DOM-NAF-2-A, with applicable Appendix, in the COLR list of references.

3.0 PROPOSED TECHNICAL SPECIFICATIONS CHANGE

Analysis of the RFA-2 fuel design at North Anna with Dominion-specific, USNRC-approved methods will require a revision to the existing plant Technical Specifications. This change is administrative in nature involving the modification of a reference that supports the COLR. The proposed change is provided below.

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

This section is appended to modify a reference that reflects the proposed change below. The modification describes a Dominion-specific analytical method used in determining core limits that are applicable to the Westinghouse RFA-2 fuel product. The following bolded text is proposed to be added to COLR Item 19:

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,” **and Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code."**

4.0 TECHNICAL EVALUATION

This package includes the technical basis and the required documentation to support the site specific application of the VIPRE-D thermal hydraulics code with the WRB-2M and W-3 CHF correlations to Westinghouse RFA-2 fuel in North Anna Power Station Units 1 and 2 cores.

Departure from nucleate boiling (DNB) analyses for the Westinghouse RFA-2 fuel product will use the USNRC-approved VIPRE-D code and the W-3 or WRB-2M CHF correlations described in DOM-NAF-2-A Appendix C (Reference 1) depending upon the transient. The W-3 correlation is only used below the first mixing grid or when the local thermodynamic conditions are outside of the range of validity of the WRB-2M CHF correlation, such as the main steam-line break evaluation, where there is reduced temperature and pressure. The W-3 CHF correlation is always used deterministically.

The Dominion Statistical DNBR Evaluation Methodology in Topical Report VEP-NE-2-A (Reference 2) is applied to all statistically-treated events. The analysis to support the implementation of DOM-NAF-2-A and VEP-NE-2-A for the Westinghouse RFA-2 fuel product is summarized below. The resulting SDL is 1.25 for transients analyzed using the Statistical DNBR Evaluation Methodology and the WRB-2M CHF correlation. In addition, the DDL is 1.14 for the WRB-2M CHF correlation (Reference 1). As described in Appendix C of DOM-NAF-2-A, the DDL for the W-3 CHF correlation is 1.30 above 1000 psia and 1.45 below 1000 psia for steam line break events. Upon USNRC review and approval of this change request, these DNBR design limits will be approved for application to the Westinghouse RFA-2 fuel product at North Anna.

Specifically, Attachment 4 provides the technical basis to support implementation of the Dominion Statistical DNBR Evaluation Methodology for Westinghouse RFA-2 fuel at North Anna with the VIPRE-D/WRB-2M code/correlation pair, as well as the SDL obtained by this implementation. Attachment 4 also confirms that the existing Reactor Core Safety Limits and protection functions (over-temperature ΔT (OTΔT), over-power ΔT (OPΔT), axial power distribution (FΔI), etc) do not require revision as a consequence of this implementation. The Statistical DNBR Evaluation Methodology is applied to the list of UFSAR transients listed in Attachment 4. Finally, all applicable Chapter 15 analyses were evaluated with the VIPRE-D/WRB-2M code/correlation pair and the Statistical DNBR Evaluation Methodology, and they all were demonstrated to have acceptable results. These evaluations support plant operation at the current licensed power level of 2940 MWt and will become the Analysis of Record (AOR) co-current with the transition to the 17x17 RFA-2 fuel product at North Anna.

5.0 SAFETY SIGNIFICANCE SUMMARY

The VIPRE-01 code has been approved by the USNRC and is widely used throughout the nuclear industry for PWR safety analyses. VIPRE-D is the Dominion version of VIPRE-01. Fleet Report DOM-NAF-2-A (Reference 1) documents the use of VIPRE-D for the thermal-hydraulic evaluation of nuclear reactor cores. VIPRE-D includes the CHF correlations to be used for the evaluation of Westinghouse RFA-2 fuel (WRB-2M and W-3 CHF correlations). These correlations are documented in Appendix C of DOM-NAF-2-A. In summary, DOM-NAF-2-A describes a methodology that is fully applicable for reload design.

The application of DOM-NAF-2-A (Reference 1) in conjunction with VEP-NE-2-A (Reference 2) is used to calculate the SDL applicable to the VIPRE-D/WRB-2M code/correlation pair for Westinghouse RFA-2 fuel at North Anna. Setpoint safety analysis evaluations have been performed to verify that the existing Reactor Core Safety Limits and protection functions (OTΔT, OPΔT, FΔI, etc) continue to be applicable for the VIPRE-D/WRB-2M code/correlation pair and the newly calculated SDL. The applicable Chapter 15 analyses were evaluated and were demonstrated to have acceptable results. In conclusion, the statepoint analysis is the basis for demonstrating the acceptability of the change.

6.0 REGULATORY EVALUATION

6.1 Applicable Regulatory Requirements/Criteria

Section 2.1 of Fleet Report DOM-NAF-2-A lists the information to be provided to the USNRC by Dominion for review and approval of any plant specific application of the VIPRE-D code:

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

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

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

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

The Safety Evaluation Report (SER) provided by the USNRC for Topical Report VEP-NE-2-A lists the following conditions for its use:

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

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

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

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 DOM-NAF-2-A).

Technical Specification 2.1, "Safety Limits," states that "The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5 [COLR]."

Attachment 4 (Technical Basis for the Proposed Technical Specification Change) provides the justification for the 95/95 DNBR limits for application to the RFA-2 fuel product.

Based on the information contained in Attachment 4, Dominion concludes that the proposed change meets regulatory requirements and criteria.

6.2 Determination of No Significant Hazards Consideration

Dominion proposes a change to the North Anna Power Station Units 1 and 2 Technical Specifications pursuant to 10CFR50.90. The proposed change adds an additional appendix (Appendix C) to the VIPRE-D code currently listed in Technical Specification (TS) 5.6.5.b. Item 19. VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs will be used to perform licensing calculations of Westinghouse RFA-2 fuel in North Anna cores, using the deterministic design limits (DDLs) documented in Appendix C of the DOM-NAF-2-A Fleet Report and the statistical design limit (SDL) using the VIPRE-D/WRB-2M code/correlation pair for RFA-2 fuel at North Anna.

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

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

   Response: No.

   Approval of the proposed changes will allow Dominion to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse RFA-2 fuel in North Anna Cores, using the DDLs documented in Appendix C of the DOM-NAF-2-A Fleet Report and the SDL documented herein. Neither the code/correlation pair nor the Statistical Departure from Nucleate Boiling Ratio (DNBR) Evaluation Methodology affect accident initiators and thus cannot increase the probability of any accident. Further, since both the deterministic and statistical DNBR limits meet the required design basis of avoiding Departure from Nucleate Boiling (DNB) with 95% probability at a 95% confidence level, the use of the new code/correlation and Statistical DNBR Evaluation Methodology do not increase the potential consequences of any accident. Finally, the full core DNB design limit provides increased assurance that the consequences of a postulated accident which includes radioactive release would be minimized because the overall number of rods in DNB would not exceed the 0.1% level. The pertinent evaluations to be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable have been performed and determined to be acceptable. The use of a different code/correlation pair will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. The use of the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse RFA-2 fuel in North Anna cores will not result in a measurable impact on normal operating plant releases and will not increase the predicted radiological consequences of accidents postulated in the UFSAR. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

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

   Response: No.

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

   The use of VIPRE-D/WRB-2M and the VIPRE-D/W-3 code/correlation pairs and the applicable fuel design limits for DNBR does not impact any of the applicable design criteria and the 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/WRB-2M and VIPRE-D/W-3 is acceptable. Design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs and 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. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   Response: No.

   Approval of the proposed changes will allow Dominion to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations of Westinghouse RFA-2 fuel in North Anna cores, using the DDLs documented in Appendix C of the DOM-NAF-2-A Fleet Report and the SDL documented herein. The SDL has been developed in accordance with the Statistical DNBR Evaluation Methodology. North Anna TS 2.1, "Safety Limits," specifies that any DNBR limit established by any code/correlation must provide at least 95% non-DNB probability at a 95% confidence level. The DNBR limits meet the design basis of avoiding DNB with 95% probability at a 95% confidence level. The required DNBR margin of safety for North Anna Power Station, which in this case is the margin between the 95/95 DNBR limit and clad failure, is therefore not reduced. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

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

6.3 Environmental Assessment

The proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(1) as follows:

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

As described in Section 6.2 above, the proposed change involves no significant hazards consideration.

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

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

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

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

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

6.4 Regulatory Conclusion

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

7.0 CONCLUSIONS

DOM-NAF-2-A, including Appendix C, has been demonstrated to be applicable to establish and support the reload design parameters for North Anna Power Station Units 1 and 2 for Westinghouse RFA-2 fuel. Upon USNRC approval, Fleet Report DOM-NAF-2-A, Appendix C will be added to the list of approved COLR references.

8.0 REFERENCES

1. Fleet Report, DOM-NAF-2, Rev. 0.1-A "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," R. S. Brackmann, July 2009. [ADAMS Accession No. ML092190894]

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

3. Letter from C. E. Rossi (USNRC) 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.

4. Letter from A. C. Thadani (USNRC) 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.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATIONS PAGES (MARK-UP)

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."

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).

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~~."~~

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.

," and Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code."

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS PAGES (TYPED)

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."

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).

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," and Appendix C,"Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code."

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.

ATTACHMENT 4

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

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

TABLE OF CONTENTS

1. Introduction

This report provides the plant specific application of the Statistical DNBR Methodology for North Anna Power Station (NAPS) cores containing Westinghouse 17x17 Robust Fuel Assembly – 2 (RFA-2) fuel product. The Westinghouse RFA-2 fuel product contains modified mid-grids and modified intermediate flow mixer grids (IFMs). Specifically, this report supports the application of U.S. Nuclear Regulatory Commission (USNRC) approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 1) to NAPS, where DNBR stands for Departure from Nucleate Boiling Ratio. It provides the technical basis and documentation required by the USNRC to evaluate the plant specific application of the VEP-NE-2-A methodology to NAPS. This application employs the VIPRE-D thermal-hydraulic computer code (DOM-NAF-2-A, Reference 2) with the Westinghouse WRB-2M Critical Heat Flux (CHF) correlation (VIPRE-D/WRB-2M code/correlation pair) for the thermal-hydraulic analysis of the Westinghouse 17x17 RFA-2 fuel product at NAPS. In particular, Dominion requests the NRC review and approval of the Statistical Design Limit (SDL) documented herein consistent with 10 CFR 50.59(c)(2)(vii) since the proposed change establishes a Design Basis Limit for a Fission Product Barrier (DBLFPB).

Dominion is also seeking approval for the inclusion of Fleet Report DOM-NAF-2-A, Appendix C, (Reference 2) to the Technical Specification (T.S.) 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/WRB-2M code/correlation pair to perform licensing calculations for the Westinghouse 17x17 RFA-2 fuel in North Anna's cores, using the deterministic design limit (DDL) qualified in Appendix C of Fleet Report DOM-NAF-2-A, and the SDL identified herein.

With these approvals, Dominion will be licensed to perform in-house Departure from Nucleate Boiling (DNB) analyses for the intended uses described in Fleet Report DOM-NAF-2-A to support North Anna Power Station, Units 1 and 2 operation with the Westinghouse 17x17 RFA-2 fuel product.

2. Background

Dominion is purchasing fuel assemblies from Westinghouse for use at North Anna Power Station, Units 1 and 2. These assemblies are planned to be inserted in Units 1 and 2, commencing with Cycle 23 for both units. The fuel assemblies are designated as the Westinghouse 17x17 RFA-2 fuel product (Reference 3). These assemblies are a replacement for the resident fuel product, which is the AREVA Advanced Mark-BW (AMBW) fuel product.

2.1 Fleet Report DOM-NAF-2-A

The computer code VIPRE (Versatile Internals and Components Program for Reactors - EPRI) was developed for the Electric Power Research Institute (EPRI) by Battelle Pacific Northwest Laboratories to perform detailed thermal-hydraulic analyses to predict CHF and DNBR of reactor cores. VIPRE-01 was approved by the U.S. Nuclear Regulatory Commission (USNRC) in References 4 and 5 for referencing in licensing applications. VIPRE-D is the Dominion version of the VIPRE computer code based upon VIPRE-01, MOD-02.1. VIPRE-D was developed to fit the specific needs of Dominion's nuclear plants and fuel products by adding vendor specific CHF correlations and customizing its input and output. Dominion has not made any modifications to the NRC-approved constitutive models and algorithms contained in VIPRE-01.

Dominion's approved Fleet Report DOM-NAF-2-A (including Appendix C, which describes the verification and qualification of the WRB-2M CHF correlation) (Reference 2) has been reviewed and approved by the USNRC in Reference 6. Fleet Report DOM-NAF-2-A provided the necessary documentation to describe Dominion's use of the VIPRE-D code, including modeling and qualification for Pressurized Water Reactors (PWR) thermal-hydraulic design and demonstrated that the VIPRE-D methodology is appropriate for PWR licensing applications. Appendix C qualified the WRB-2M CHF correlation with the VIPRE-D code and listed the deterministic code/correlation DNBR limits. The WRB-2M CHF correlation is applicable for the DNBR evaluation of the Westinghouse 17x17 RFA-2 fuel product [refer to Section 2.3 for further discussion].

In addition, Section 2.1 of Fleet Report DOM-NAF-2-A listed the information to be provided to the USNRC by Dominion for the review and approval of any plant specific application of the VIPRE-D code:

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

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

3) Any technical specification changes related to thermal over-temperature ΔT (OTΔT), over-power ΔT (OPΔT), axial power distribution (FΔI) or other reactor protection function, as well as revised Reactor Core Safety Limits (Section 4.5).

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

This report provides the technical basis (items 1 through 4 above) to support implementation of the Dominion Statistical DNBR Evaluation Methodology for 17x17 RFA-2 fuel at NAPS with the VIPRE-

D/WRB-2M code/correlation pair, as well as the SDL obtained by this implementation (DOM-NAF-2-A Condition 2). This report also documents that the existing Reactor Core Safety Limits and protection functions (OTΔT, OPΔT, FΔI, etc.) do not require revision as a consequence of this implementation (DOM-NAF-2-A Condition 3). The list of UFSAR transients for which the code/correlation pair will be applied is also included herein (DOM-NAF-2-A Condition 4).

2.2 Topical Report VEP-NE-2-A

In 1985, Virginia Power (Dominion) submitted Topical Report VEP-NE-2-A (Reference 1) to the USNRC describing a proposed methodology for the statistical treatment of key uncertainties in core thermal-hydraulic DNBR analysis. The methodology provided DNBR margin through the use of statistical rather than deterministic uncertainty treatment. The methodology was reviewed and approved by the USNRC in May 1987, and the Safety Evaluation Report (SER) provided by the USNRC listed the following conditions for its use (Reference 7):

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

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

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

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

The VEP-NE-2-A methodology is currently implemented at North Anna consistent with our Licensed Amendment Request (LAR) submitted to the USNRC on July 5, 2005 (Reference 8), as supplemented by letters dated March 30, April 13, and May 11, 2006 (References 9, 10 and 11) and approved by the USNRC on July 21, 2006 (Reference 12). The USNRC approved Dominion to perform DNB analyses for AREVA AMBW fuel at NAPS using the VIPRE-D/BWU code/correlation pair. In addition, DOM-NAF-2-A, including Appendix A (Reference 2) was added to the list of methodologies approved for the determination of core operating limits in TS 5.6.5.b. In order for Dominion to maintain vendor independence, Dominion intends to perform in-house DNB analyses for 17x17 RFA-2 fuel at North Anna Power Station, Units 1 and 2 by adding Appendix C of DOM-NAF-2-A to the list of methodologies approved for determination of core operating limits.

2.3 Westinghouse's Fuel Criteria Evaluation Process for the Application of the WRB-2M CHF Correlation to the RFA-2 Fuel Product

In WCAP-12488-A (Reference 13), Westinghouse described a process and criteria that it intends to apply to changes or improvements in existing fuel designs that will not require prior NRC review and approval when these criteria are satisfied. Westinghouse also will apply these criteria to adjustments or improvements of fuel performance design evaluation models, based on new data, without NRC review and approval. The NRC staff reviewed the Westinghouse fuel design criteria evaluation process described in WCAP-12488 and found it acceptable for licensing applications.

Westinghouse submitted LTR-NRC-01-44 (Reference 14) to the NRC documenting the minor design changes between the RFA and RFA-2 fuel products via Westinghouse's NRC-approved Fuel Criteria Evaluation Process (FCEP) (Reference 13). In LTR-NRC-01-44 (Reference 14), Westinghouse notified the NRC that the WRB-1 and WRB-2 DNB correlations are applicable to the RFA-2 fuel product. Westinghouse revised this document in LTR-NRC-02-55 (Reference 3) to include the applicability of the WRB-2M correlation to the RFA-2 fuel mid-grid/IFM grid design modifications (i.e., the 17x17 RFA-2 fuel product).

3. Implementation of the Statistical DNBR Evaluation Methodology

3.1 Methodology Review

In Appendix C to Fleet Report DOM-NAF-2-A (Reference 2), Dominion calculated a DDL for the VIPRE-D/WRB-2M code/correlation pair. The Statistical DNBR Evaluation Methodology (Reference 1) is employed herein to develop an SDL for NAPS. 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 SDL) is larger than the deterministic code/correlation design limit, its use is advantageous as the Statistical DNBR Evaluation Methodology permits the use of nominal values for operating initial conditions instead of requiring the application of evaluated uncertainties to the initial conditions for statepoint and transient analysis.

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

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

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

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

3.2 Uncertainty Analysis

This section is included herein to satisfy Condition 2 in the SER (Reference 7) of VEP-NE-2-A (Reference 1).

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

The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and core inlet temperature were quantified using all sensor, rack, and other component uncertainties. Then, the uncertainties were combined in a manner consistent with their relative dependence or independence to quantify the total uncertainty for each parameter. 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 $\pm$ 3.67% of span or $\pm$ 29.3 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 $\pm$ 30.0 psia and a standard deviation (σ) of 15.31 psia. The applied uncertainty is unchanged from that employed in Reference 8 and subsequently approved in Reference 12.

Dominion has quantified the magnitude and distribution of uncertainty on the average temperature (T_{AVG}) per the T_{AVG} rod control system. The average temperature uncertainty was quantified as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 3.22% of span or $\pm$ 3.22°F. The impact of parameter surveillance was considered. The current parameter surveillance limit for average temperature of 591.0°F was determined to be acceptable. With this parameter surveillance limit, the average temperature uncertainty was conservatively defined as a normal, two-sided, 95% probability distribution with a magnitude of $\pm$ 4.2°F and a standard deviation (σ) of 2.143°F. The applied uncertainty is unchanged from that employed in Reference 8 and subsequently approved in Reference 12.

Dominion has quantified the uncertainty on core power as measured by the secondary side heat balance as 0.862% at the non-uprated power 2893 MWt . The core power uncertainty associated with the MUR uprated power (2940 MWt) is less than 0.4%, but will not be credited. The power parameter uncertainty was conservatively treated as a normal, two-sided, 95% probability distribution with a magnitude of ±1.511% and a standard deviation (σ) of 0.711%. The applied uncertainty is bounding and unchanged from that employed in Reference 8 and subsequently approved in Reference 12.

Dominion has quantified the uncertainty on the reactor coolant system (RCS) flow as 2.390%. This parameter uncertainty is treated as a normal, two-sided, 95% probability distribution with a magnitude of ±2.862% and a standard deviation (σ) of 1.46%. The applied uncertainty is bounding and unchanged from that employed in Reference 8 and subsequently approved in Reference 12.

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 3.39%. Conservatively, the measured $F_{\Delta H}^{N}$ uncertainty was defined as a normal distribution with a 4% tolerance interval for consistency with previous applications.

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

The total core bypass flow consists of separate flow paths through the thimble tubes, direct leakage to the outlet nozzle, baffle joint leakage flow, upper head spray flow and core-baffle gap flow. These five components were each quantified based on the current 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%. The analysis assumed that the probability was uniformly distributed.

Table 3.2-1: North Anna Parameter Uncertainties

PARAMETER	NOMINAL VALUE	STANDARD DEVIATION	UNCERTAINTY	DISTRIBUTION
Pressure [psia]	2250	15.31 psi	±30.0 psi at 2σ	Normal
Temperature [°F]	553.7	2.143°F	±4.2°F at 2σ	Normal
Power [MWt]	2,940	0.771%	±1.511% at 2σ	Normal
Flow [gpm]	295,000	1.46%	±2.862% at 2σ	Normal
$F_{\Delta H}^{N}$	1.587	2.0%	±4.0% at 2σ	Normal
$F_{\Delta H}^{E}$	1.0	N/A	±3.0%	Uniform
Bypass [%]	5.5	N/A	±1.0%	Uniform

3.3 CHF Correlations

The WRB-2M/W-3 CHF correlations are used for the calculation of DNBRs in the Westinghouse 17x17 RFA-2 fuel product. Only the WRB-2M CHF correlation is applicable to the operating conditions for which the Statistical DNBR Evaluation Methodology applies. Table 3.3-1 presents the Design Limit correlation data for VIPRE-D/WRB-2M code/correlation pair. The W-3 correlation is only used below the first mixing grid or when the local thermodynamic conditions are outside of the range of validity of the WRB-2M CHF correlation, such as the main steam-line break evaluation, where there are reduced temperature and pressure. The W-3 CHF correlation is always used deterministically.

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

	WRB-2M
Average M/P	1.0006
S(M/P)	0.0640
n	241
K*	1.0824
K x S(M/P)	0.06927

3.4 Model Uncertainty Term

This section is included herein to satisfy Condition 3 in the SER (Reference 7) of the Statistical DNBR Evaluation Methodology Topical Report (Reference 1).

The VIPRE-D 20-channel production model for North Anna with the 17x17 RFA-2 fuel product was used in the development of the VIPRE-D/WRB-2M code/correlation pair SDL for North Anna. Since this is the production model that Dominion intends to use for all North Anna evaluations, there is no additional uncertainty associated with the use of this model. In summary, it is concluded that no correction for model uncertainty is necessary, and the model uncertainty term is set to zero for the calculation of the total DNBR standard deviation.

3.5 Code Uncertainty

The code uncertainty accounts for any differences between Dominion's VIPRE-D and Westinghouse's THINC and VIPRE (i.e. VIPRE-W) codes, with which the WRB-2M 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

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

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

induced uncertainties. The code uncertainty was quantified at 5%; consistent with the factors specified for other thermal/hydraulic codes in Reference 1. The basis for this uncertainty is described in detail by USNRC staff in Reference 7. In Reference 7, the USNRC Staff refers to the 5% uncertainty as being a 2σ value. The 5% code uncertainty is certainly conservative in light of the excellent VIPRE-D/VIPRE-W and VIPRE-D/CHF data comparisons. However, the 5% uncertainty serves as a conservative factor that may be shown to be wholly or partially unnecessary at a later time. A one-sided 95% confidence level on the code uncertainty is then 3.04% $(=(5.0\%)/1.645)$. The use of the 1.645 divisor (the one-sided 95% tolerance interval multiplier) is conservative since the USNRC Staff considers the 5% uncertainty to be a 2σ value.

3.6 Monte Carlo Calculations

In order to perform the Monte Carlo analysis, nine Nominal Statepoints covering the full range of normal operation and anticipated transient conditions were selected. These statepoints must span the range of conditions over which the statistical methodology will be applied. Two statepoints were selected at each of the four Reactor Core Safety Limit (RCSL) pressures (2400, 2250, 2000, and 1860 psia). For each of the RCSLs, a high power statepoint at 118% and a statepoint near the intercept of the DNBR limit line with the vessel exit boiling line were chosen. In order to apply the methodology to low flow events, a low flow statepoint is also included. The selected Nominal Statepoints are listed in Table 3.6-1.

Table 3.6-1: Nominal Statepoints for Westinghouse 17x17 RFA-2 Fuel at North Anna with VIPRE-D/WRB-2M

STATE POINT	PRESSURIZER PRESSURE [psia]	INLET TEMPERATURE [°F]	POWER [%]	FLOW [%]	$F_{\Delta H}^{N}$	MDNBR
A	2400.0	605.07	118	100	1.587	1.242
B	2400.0	613.30	113	100	1.587	1.241
C	2250.0	596.99	118	100	1.587	1.240
D	2250.0	608.45	111	100	1.587	1.241
E	2000.0	585.74	118	100	1.587	1.240
F	2000.0	598.32	111	100	1.587	1.241
G	1860.0	581.28	118	100	1.587	1.244
H	1860.0	588.98	114	100	1.587	1.242
I	2250.0	553.70	108	62.98	1.587	1.242

The Monte Carlo analysis itself consisted of 2000 calculations performed around each of the nine Nominal Statepoints. As described in Section 3.1, the DNBR standard deviation at each Nominal Statepoint was augmented by the code/correlation uncertainty, the small sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation.

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

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

[Equation 3.2]

where:

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

Note that this equation differs slightly from the equation listed in Reference 1. It has an additional factor applied to the Randomized DNBR s_{DNBR}, the $1/N$ factor to correct for the uncertainty in the mean of the correlation. This factor has been used in previous implementations of the Statistical DNBR Evaluation Methodology, such as Reference 8 as supplemented by Reference 11, which was subsequently approved in Reference 12.

The limiting peak fuel rod SDL was calculated to be 1.243 for VIPRE-D/WRB-2M code/correlation pair. The Monte Carlo Statepoint analysis is summarized in Table 3.6-2.

Table 3.6-2: Peak Pin SDL Results for North Anna 17x17 RFA-2 Fuel with VIPRE-D/WRB-2M

STATEPOINT	Randomized DNB s_{DNBR}	Total DNB s_{TOTAL}	Pin Peak $SDL_{95/95}$
A	0.1281	0.1405	1.231
B	0.1223	0.1344	1.221
C	0.1302	0.1426	1.235
D	0.1268	0.1391	1.229
E	0.1287	0.1411	1.232
F	0.1228	0.1349	1.222
G	0.1245	0.1367	1.225
H	0.1238	0.1360	1.224
I	0.1348	0.1475	**1.243**

3.7 Full Core DNB Probability Summation

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

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

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

in which:

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

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

Table 3.7-1: ∂(DNBR)/∂(1/FΔh) Estimation for WRB-2M

STATEPOINT	β	se(β)	β^*	R^2
A	5.24294	0.00491	5.23486	99.9%
B	5.10419	0.00514	5.09574	99.9%
C	5.42769	0.00414	5.42088	99.9%
D	5.28417	0.00474	5.27637	99.9%
E	5.48612	0.00507	5.47778	99.9%
F	5.28143	0.00494	5.27330	99.9%
G	5.36615	0.00443	5.35887	99.9%
H	5.23293	0.00454	5.22546	99.9%
I	5.70647	0.00736	5.69437	99.8%

A representative fuel rod census curve used for the determination of the SDL is listed in Table 3.7-2. The full core DNB probability summation will be re-evaluated on a reload basis to verify the applicability of the fuel rod census ($F_{\Delta H}^{N}$ versus % of core with $F_{\Delta H}^{N}$ greater than or equal to a given $F_{\Delta H}$ limit) used in the implementation analysis. The limiting full-core DNB probability summation resulted in an SDL of 1.247. The DNB probability summation for VIPRE-D/WRB-2M code/correlation pair is summarized in Table 3.7-3.

Table 3.7-2: Representative Fuel Rod Census
for a Maximum Peaking Factor FΔh = 1.587

MAXIMUM % OF FUEL RODS IN CORE WITH FΔh ≥ to:	FΔh LIMIT
0.0	1.5870
0.1	1.5866
0.2	1.5860
0.3	1.5855
0.4	1.5850
0.5	1.5840
0.6	1.5829
0.7	1.5815
0.8	1.5803
0.9	1.5786
1.0	1.5770
1.5	1.5682
2.0	1.5580
2.5	1.5500
3.0	1.5430
4.0	1.5330
5.0	1.5240
6.0	1.5150
7.0	1.5060
8.0	1.4970
9.0	1.4900
10.0	1.4860
20.0	1.4560
30.0	1.4100
40.0	1.3500
PEAK	**1.5870**

Table 3.7-3: Full Core DNB Probability Summation for 17x17 RFA-2 Fuel with VIPRE-D/WRB-2M

STATEPOINT	S_{TOTAL}	% of Rods in DNB	Full Core $SDL_{99.9}$
A	0.1405	0.09861	1.238
B	0.1344	0.09887	1.226
C	0.1426	0.09923	1.240
D	0.1391	0.09849	1.234
E	0.1411	0.09876	1.236
F	0.1349	0.09880	1.225
G	0.1367	0.09853	1.228
H	0.1360	0.09873	1.228
I	0.1475	0.09906	**1.247**

3.8 Verification of Nominal Statepoints

Condition 1 of the USNRC's SER for VEP-NE-2-A (Reference 7) requires that the Nominal Statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may potentially be applied.

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

Table 3.8-1 shows the R^2 coefficients obtained for the verification of the nominal statepoints. The largest linear curve fit R^2 coefficient is 53.65%, thus validating that there is no dependence.

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

Table 3.8-1: R^2 Coefficients for the Verification of the Nominal Statepoints for North Anna 17x17 RFA-2 Fuel with VIPRE-D/WRB-2M

	R^2 - Linear Regression
Pressure	9.83%
Temperature	41.02%
Flow Rate	53.65%
Power	1.34%

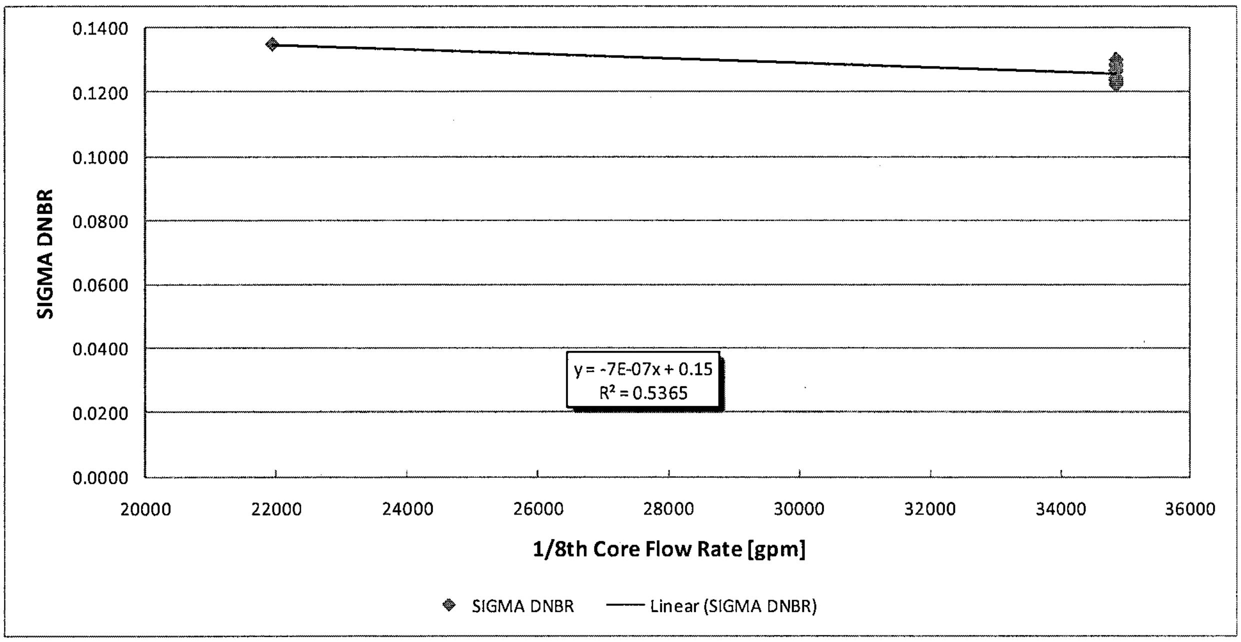

Figure 3.8-1: Variation of the Unrandomized Standard Deviation with Flow Rate for the WRB-2M CHF Correlation

3.9 Scope of Applicability

This section is included herein to satisfy Condition 4 in the SER (Reference 7) of VEP-NE-2-A (Reference 1).

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

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

Table 3.9-1: UFSAR Transients Analyzed with VIPRE-D/WRB-2M/W-3 for North Anna

ACCIDENT	NAPS USAR SECTION	APPLICATION
Uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition	15.2.1	DET-DNB
Uncontrolled rod cluster control assembly bank withdrawal at power	15.2.2	STAT-DNB
Rod cluster control assembly misalignment (System Malfunction or Operator Error)	15.2.3	STAT-DNB
Uncontrolled boron dilution	15.2.4	STAT-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 to the station auxiliaries	15.2.9	STAT-DNB
Excessive heat removal due to feedwater system malfunctions	15.2.10	STAT-DNB
Excessive load increase incident	15.2.11	STAT-DNB
Accidental depressurization of the reactor coolant system	15.2.12	STAT-DNB
Accidental depressurization of the main steam system	15.2.13	DET-DNB
Spurious operation of the safety injection system at power	15.2.14	STAT-DNB
Complete loss of forced reactor coolant 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 feed water pipe	15.4.2.2	non-DNB
Locked reactor coolant pump rotor	15.4.4	STAT-DNB

3.10 Summary of Analysis

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

In accordance with the Statistical DNBR Evaluation Methodology, 2,000 random statepoints are generated about each nominal statepoint and VIPRE-D is then executed to obtain MDNBRs. The standard deviation for the distribution of 2,000 MDNBRs is referred to as the unrandomized standard deviation. At the limiting Nominal Statepoint (I), the standard deviation of the randomized DNBR distributions, which is the unrandomized corrected for CHF correlation uncertainty, was found to be 0.1348. This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation of 0.1475, as listed in Table 3.6-2. The use of 0.1475 in Equation 3.1 yields a peak pin DNBR limit of 1.243 with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core of 1.247, which occurs at Nominal Statepoint (B). Therefore the VIPRE-D/WRB-2M code/correlation pair SDL for North Anna 17x17 RFA-2 fuel is set to 1.25.

4. Application of VIPRE-D/WRB-2M/W-3 to NAPS

VIPRE-D/WRB-2M code/correlation pair together with the Statistical DNBR Evaluation Methodology will be applied to Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWFS), and to the Complete Loss of Flow event and the Locked Rotor Accident. The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}{}^{N}$ (measurement component) and $F_{\Delta H}{}^{E}$ uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

The WRB-2M and W-3 CHF correlations are used for the calculation of DNBRs in the Westinghouse 17x17 RFA-2 fuel product. The W-3 correlation is only used below the first mixing grid or when the local thermodynamic conditions are outside of the range of validity of the WRB-2M CHF correlation, such as the main steam-line break evaluation, where there are reduced temperature and pressure. The W-3 CHF correlation is always used deterministically.

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

4.1 VIPRE-D/WRB-2M SDL for North Anna

The SDL for North Anna cores containing Westinghouse 17x17 RFA-2 fuel with the VIPRE-D/WRB-2M code/correlation pair was derived in Section 3 of this report. The SDL for VIPRE-D/WRB-2M code/correlation pair is determined to be 1.25. The SDL limit provides a peak fuel rod DNB protection with at least 95% probability at a 95% confidence level and a 99.9% DNB protection for the full core. This SDL is plant specific as it already includes the North Anna specific uncertainties for the key parameters accounted for in the application of the Statistical DNBR Evaluation Methodology. Therefore, this limit is applicable to the analysis of statistical DNB events of Westinghouse 17x17 RFA-2 fuel in North Anna cores with the VIPRE-D/WRB-2M code/correlation pair.

4.2 Safety Analysis Limits (SAL)

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

For deterministic DNB analyses, the design DNBR limit is set equal to the applicable code/correlation limit and it is termed the DDL. For statistical DNB analyses, the design DNBR limit is set equal to the applicable SDL. These design limits are two of the DBLFPB described in

Reference 15. The DDLs and SDLs are fixed and any changes to their value require USNRC review and approval. However, the safety analysis limits for deterministic and statistical DNB analyses (SAL_{DET} and SAL_{STAT}, respectively) may be changed without prior USNRC review and approval, provided the changes meet the criteria established in Reference 15.

A deterministic and statistical SAL equal to 1.55 has been selected for 17x17 RFA-2 fuel at NAPS with the VIPRE-D/WRB-2M code/correlation pair. 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.

Table 4.2-1: DNBR Limits for WRB-2M and W-3

VIPRE-D/WRB-2M	
DDL	1.14
SDL	1.25
SAL	1.55
VIPRE-D/W-3	
DDL (≥1000 psia)	1.30
DDL (<1000 psia)	1.45
SAL (≥1000 psia)	1.42
SAL (<1000 psia)	1.58

4.3 Retained DNBR Margin

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

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

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

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

DETERMINISTIC DNB APPLICATIONS			
DNB CORRELATION	**DDL**	**SAL_{DET}**	**RETAINED DNBR MARGIN [%]**
WRB-2M	1.14	1.55	26.4
W-3 (< 1000 psia)	1.45	1.61	9.9
W-3 (≥1000 psia)	1.30	1.44	9.7

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

STATISTICAL DNB APPLICATIONS			
DNB CORRELATION	**SDL**	**SAL_{STAT}**	**RETAINED DNBR MARGIN [%]**
WRB-2M	1.25	1.55	19.3

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

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

4.4 Verification of Existing Reactor Core Safety Limits, Protection Setpoints and NAPS UFSAR Chapter 15 Events

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

To demonstrate that the DNB performance of the Westinghouse 17x17 RFA-2 fuel is acceptable, Dominion performed calculations for full-core configurations of Westinghouse 17x17 RFA-2 fuel. The calculations were performed using the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs and selected statepoints including: the reactor core safety limits (RCSL), axial offset limits (AO), rod withdrawal from subcritical (RWFS), rod withdrawal at power (RWAP), loss of flow (LOFA), locked rotor events (LOCROT), hot zero power steam line break (MSLB), dropped rod limit line (DRLL), and static rod misalignment (SRM). These various statepoints provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power FΔh), pressure and temperature (RCSL); (b) limiting axial flux shapes at several axial offsets (AO); and (c) low flow (LOFA and LOCROT). The statepoints for the RWFS and MSLB were evaluated with deterministic DNB methods. The remaining statepoints were evaluated using statistical DNB methods. The evaluation criterion for these analyses is that the minimum DNBR must be equal to or greater than the applicable safety analysis limit (SAL) listed in Table 4.2-1.

The results of the calculations demonstrate that the minimum DNBR values are equal to or greater than the applicable safety analysis limit for the analyses that are performed to address statepoints of the Reactor Core Safety Limits, the OTΔT, OPΔT and FΔI trip setpoints, as well as all the evaluated Chapter 15 events (including the LOFA and LOCROT) with an FΔh of 1.587 (COLR limit of 1.65 divided by the measurement uncertainty of 1.04 = 1.587) at a Rated Thermal Power of 2940 MWt.

5. Conclusions

Dominion's Statistical DNBR Evaluation Methodology has been used to derive an SDL. This application employs the VIPRE-D code with the Westinghouse WRB-2M CHF correlation (VIPRE-D/WRB-2M code/correlation pair) for the thermal-hydraulic analysis of Westinghouse 17x17 RFA-2 fuel product at NAPS. 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. In particular, Dominion seeks the review and approval of the SDL of 1.25 documented herein as per 10 CFR 50.59(c)(2)(vii) since it constitutes a DBLFPB.

Dominion is also seeking the approval for the inclusion of Fleet Report DOM-NAF-2-A, Appendix C, 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 COLR). This would allow Dominion the use of the VIPRE-D/WRB-2M code/correlation pair to perform licensing calculations for the Westinghouse 17x17 RFA-2 fuel in North Anna's cores, using the DDL qualified in Appendix C of Fleet Report DOM-NAF-2-A, and the SDL documented herein. In addition, DOM-NAF-2-A provides justification of the normality of the WRB-2M CHF M/P distributions, their means and standard deviations, as required by the SER to Reference 1.

6. References

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

2. Fleet Report, DOM-NAF-2, Rev. 0.1-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," R. S. Brackmann, July 2009 (ADAMS Accession No. ML092190894).

3. Letter from H. A. Sepp, (Westinghouse) to J. S. Wermiel (NRC), "Fuel Criterion Evaluation Process (FCEP) Notification of the RFA-2 Design, Revision 1 (Proprietary)," LTR-NRC-02-55, November 13, 2002 (ADAMS Accession No. ML023190181).

4. 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.

5. 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.

6. Letter from D. Wright (NRC) to D. A. Christian (Dominion), "Kewaunee Power Station, Millstone Power Station, Units 2 and 3, North Anna Power Station, Unit Nos. 1 and 2, and Surry Power Station, Unit Nos. 1 and 2 – Appendix C to the Dominion Fleet Report DOM-NAF-2, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code" (TAC Nos. MD8703, MD8704, MD8705, MD8707, MD8708, MD8709)," Serial No. 09-290, April 22, 2009.

7. 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.

8. Letter from E. S. Grecheck (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specification Changes, Addition of Analytical Methodology to COLR," Serial No. 05-419, July 5, 2005 (ADAMS Accession No. ML051890034).

9. Letter from W. R. Mathews (Dominion) to USNRC, "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)," Serial No. 06-142, March 30, 2006 (ADAMS Accession No. ML060900631).

10. Letter from E. S. Grecheck (Dominion) to USNRC, "Virginia Electric and Power Company (Dominion), North Anna Power Station Units Nos. 1 and 2, Proposed Technical Specifications Changes on Addition of Analytical Methodology to the Core Operating Limits Report, Administrative Correction," Serial No. 06-142A, April 13, 2006 (ADAMS Accession No. ML061040062).

11. Letter from E. S. Grecheck (Dominion) to USNRC, "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)," Serial No. 06-142B, May 11, 2006 (ADAMS Accession No. ML061310495).

12. Letter from S. Monarque (NRC) to D. A. Christian (Dominion), "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)," Serial No. 06-643, July 21, 2006 (ADAMS Accession No. ML062020005).

13. Technical Report, WCAP-12488-P-A, "Westinghouse Fuel Criteria Evaluation Process," October 1994.

14. Letter from H. A. Sepp (Westinghouse) to J. S. Wermiel (NRC), "Fuel Criterion Evaluation Process (FCEP) Notification of the RFA-2 Design (Proprietary)," LTR-NRC-01-44, December 19, 2001.

15. Technical Report, NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," Nuclear Energy Institute, November 2000.

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