VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10CFR50.90

June 23, 2011

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

Serial No.	11-349
NL&OS/ETS	R0
Docket Nos.	50-338/339
License Nos.	NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
PROPOSED LICENSE AMENDMENT REQUEST(LAR)
ADDITION OF ANALYTICAL METHODOLOGY TO COLR

In a July 19, 2010 letter (Serial No. 10-404) supplemented by letters dated September 9, 2010 letter (Serial No. 10-523), January 26, 2011 (Serial No. 11-019), and May 16, 2011 (Serial No. 11-279) Dominion requested amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station (NAPS) Units 1 and 2, respectively. The proposed amendment requested 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," as a referenced analytical methodology into Technical Specification 5.6.5.b. Plant specific application of the methodology also requires approval of the Statistical Design Limit (SDL) for the relevant code/correlation pair. Consequently, in addition to including Appendix C of Fleet Report DOM-NAF-2-A into TS 5.6.5.b, Dominion also requested NRC review and approval for the use of the Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," with the Westinghouse RFA-2 fuel at North Anna and the VIPRE-D/WRB-2M code/correlation pair, as well as the SDL.

In a June 15, 2011 letter, the NRC requested additional information (RAI) to complete the review of the proposed licensing actions. The response to this RAI is provided in the attachment to this letter.

The information provided in the attachment to this letter does not impact the conclusion of the significant hazards consideration determination as defined in 10 CFR 50.92 or the evaluation for eligibility for categorical exclusion as set forth in 10 CFR 51.22(c)(9).

Dominion is currently planning to use Westinghouse RFA-2 fuel in NAPS Units 1 and 2 commencing with NAPS Unit 1, Cycle 23 (Spring 2012) and NAPS Unit 2, Cycle 23 (Spring 2013). Therefore, Dominion continues to request approval of the proposed amendments by July 21, 2011 to complete analysis work required to support operation with the Westinghouse RFA-2 fuel. Dominion also continues to request 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

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

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 23rd day of June, 2011.

My Commission Expires: May 31, 2014

Vicki L. Hull
Notary Public

Attachment: Response to Request for Additional Information

Commitments made in this letter: None

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ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PROPOSED LICENSE AMENDMENT REQUEST (LAR)
ADDITION OF ANALYTICAL METHODOLOGY TO COLR

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

Background

In a letter dated July 19, 2010 (Serial No. 10-404), supplemented by letters dated September 9, 2010 (Serial No. 10-523), January 26, 2011 (Serial No. 11-019), and May 16, 2011 (Serial No. 11-279) Virginia Electric and Power Company (Dominion) requested amendments in the form of changes to the Technical Specifications (TS) to the Facility operating License numbers NPF-4 and NPF-7 for North Anna Power Station (NAPS) Units 1 and 2, respectively. The license amendment request (LAR) also requested the NRC's review and approval of the use of Dominion's Topical Report VEP-NE-2-A using the VIPRE-D/WRB-2M code/correlation with the Westinghouse 17x17 RFA-2 fuel and the resulting Statistical Design Limit (SDL).

In reviewing Dominion's submittal related to qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code for NAPS Units 1 and 2, the NRC staff determined that the following information is needed in order to complete their review:

NRC Question 1

Based on the guidance specified in Generic Letter 88-16, (GL 88-16), each approved methodology listed in Technical Specifications (TSs) Section 5.6.5.b should support a calculation for a cycle-specific core operating limit in TS Section 5.6.5.a. In other words, the methodologies listed should identify its supporting role for the cycle-specific parameters in order to be listed in TS Section 5.6.5.b.

a. Explain why no date of approval and use of the approved methodologies are proposed for TS Section 5.6.5.b.
b. For each approved methodology listed in TS Section 5.6.5.b, identify the cycle-specific parameter listed in TS Section 5.6.5.a that relates to a real application to the current reload analysis.
c. Provide justification that all the proposed methodologies meet the guidance of GL 88-16 to be listed in TS Section 5.6.5.b.

Dominion's Response

a. Amendments 146 and 130, dated June 7, 1991 for North Anna Units 1 and 2, respectively, removed the cycle specific parameters from the Technical Specifications and incorporated COLR reporting requirements in accordance with GL 88-16.

   During North Anna's conversion to Improved Standard Technical Specifications in 2002, Dominion adopted TSTF-363, which removed additional information (dates, revision numbers, and applicable variables) from the list of analytical methods listed in TS Section 5.6.5. The NRC SER approved the removal of the dates and revisions of the approved analytical methods consistent with TSTF-363. In the NRC SER dated April 5, 2002 (Amendments 231 and 212 for Units 1 and 2, respectively) this information was categorized as "less restrictive removal of detail" and the staff stated:

"The Staff has concluded that these types of detailed information and specific requirements do not need to be included in the ITS to ensure the effectiveness of the ITS to adequately protect the health and safety of the public. Accordingly, these requirements may be moved to one of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

- Bases controlled in accordance with ITS 5.513, "Technical Specifications (TS) Bases Control Program."
- UFSAR (which references TRM) controlled by 10 CFR 50.59.
- Programmatic documents required by ITS Section 5.5 and controlled by ITS Section 5.4.
- Inservice Inspection (ISI) and IST Programs controlled by 10 CFR 50.55a.
- ODCM controlled by ITS 5.5.1.
- COLR controlled by ITS 5.6.4.
- QA Plan, as approved by the NRC and referenced in the UFSAR, controlled by 10 CFR Part 50, Appendix B, and 10 CFR 50.54(a).
- Site Emergency Plan controlled by 10 CFR 50.54(q)."

"To the extent that information has been relocated to licensee-controlled documents, such information is not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. Further, where such information is contained in LCOs and associated requirements in the CTS, the staff has concluded that they do not fall within any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii) and discussed in the Final Policy Statement (see Section 2.0 of this SE). Accordingly, existing detailed information, such as generally described above, may be removed from the CTS and not included in the ITS."

b. The following is a list of the NRC approved methodologies in the current North Anna TS and the associated cycle specific parameters from TS 5.6.5.a that each method is used to develop.

   1. VEP-FRD-42-A, "Reload Nuclear Design Methodology." **Rev. 2.1-A, August 2003**

      Methodology for TS 3.1.1 – Shutdown Margin, TS 3.1.3 – Moderator Temperature Coefficient, TS 3.1.4 – Rod Group Alignment Limits, TS 3.1.5 – Shutdown Bank Insertion Limit, TS 3.1.6 - Control Bank Insertion Limits, TS 3.1.9 – Physics Test Exceptions-Mode 2, TS 3.2.1 - Heat Flux Hot Channel Factor, TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor, TS 3.5.6 – Boron Injection Tank (BIT) and TS 3.9.1- Boron Concentration

2. ~~WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."~~

   Methodology no longer used. Methodology is being removed in the Best Estimate (BE) LBLOCA license amendment request dated October 21, 2010.

3. ~~WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."~~

   Methodology no longer used. Methodology is being removed in the BE LBLOCA license amendment request dated October 21, 2010.

4. ~~WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."~~

   Methodology no longer used. Methodology is being removed in the BE LBLOCA license amendment request dated October 21, 2010.

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code." **August 1985.**

   This methodology is for Westinghouse fuel and is currently not being used. The methodology is being maintained as contingency if a significant issue was discovered with the resident AREVA fuel that would prohibit reuse of certain AREVA fuel. In that case, previously irradiated Westinghouse fuel could be reinserted to support continued operation. This methodology is also applicable to Westinghouse RFA-2 fuel. **Methodology for TS 3.2.1 - Heat Flux Hot Channel Factor.**

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code." **August 1985.**

   This methodology is for Westinghouse fuel and is currently not being used. The methodology is being maintained as contingency if a significant issue was discovered with the resident AREVA fuel that would prohibit reuse of certain AREVA fuel. In that case, previously irradiated Westinghouse fuel could be reinserted to support continued operation. This methodology is also applicable to Westinghouse RFA-2 fuel. **Methodology for TS 3.2.1 - Heat Flux Hot Channel Factor.**

7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT." **April 1995.**

   This methodology is for Westinghouse fuel and is currently not being used. The methodology is being maintained as contingency if a significant issue was discovered with the resident AREVA fuel that would prohibit reuse of certain AREVA fuel. In that case, previously irradiated Westinghouse fuel could be reinserted to support continued operation. This methodology is also applicable to Westinghouse RFA-2 fuel. **Methodology for TS 2.1.1 Reactor Core Safety Limits, TS 3.2.1 - Heat Flux Hot Channel Factor**.

8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology." **Rev. 0, June 1987**

   Methodology for TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor and TS 3.4.1 – RCS Pressure, Temperature and Flow DNB Limits

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

   Methodology no longer used. Methodology is being removed in the BE LBLOCA license amendment request dated October 21, 2010.

10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology Associated FQ Surveillance Technical Specifications." **Rev. 0.1, August 2003.**

    Methodology for TS 3.2.1 – Heat Flux Hot Channel Factor and TS 3.2.3 – Axial Flux Difference

11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function." **September 1986**

    Methodology for TS 2.1.1 – Reactor Core Safety Limits and TS 3.3.1 – Reactor Trip System Instrumentation

12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report." **January 1999**

    Methodology for TS 2.1.1 – Reactor Core Safety Limits, TS 3.1.1 – Shutdown Margin, TS 3.1.4 – Rod Group Alignment Limits, TS 3.1.9 – Physics Test Exceptions-Mode 2, TS 3.3.1 – Reactor Trip System Instrumentation, TS 3.4.1 – RCS Pressure, Temperature, and Flow DNB Limits, TS 3.5.6 – Boron Injection Tank (BIT) and TS 3.9.1 – Boron Concentration

13. BAW-10227-P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel." Structural Material (M5) in PWR Reactor Fuel." **Rev. 0, February 2003.**

    Methodology for TS 2.1.1 – Reactor Core Safety Limits, TS 3.2.1 - Heat Flux Hot Channel Factor

14. BAW-10199-P-A, "The BWU Critical Heat Flux Correlations." **Rev. 0, August 1996.**

Note there are 2 applicable addenda:
Addendum 1-A, "Application to the Mark B11 and Mark BW17 MSM Designs," December 2000;
Addendum 2-A, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," June 2002;

This methodology was used for AREVA fuel during fuel transition for TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor and TS 3.4.1 – RCS Pressure, Temperature and Flow DNB Limits. The current core uses the approved DOM-NAF-2-A (VIPRE-D) to establish the limits on these cycle specific parameters. However, the methodology is being maintained to also permit vendor evaluation of AREVA fuel.

15. BAW-10170-P-A, "Statistical Core Design for Mixing Vane Cores." **Rev. 0, December 1988.**

This methodology was used for AREVA fuel during fuel transition for TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor and TS 3.4.1 – RCS Pressure, Temperature and Flow DNB Limits. The current core uses the approved DOM-NAF-2-A (VIPRE-D) to establish the limits on these cycle specific parameters. However, the methodology is being maintained to also permit vendor evaluation of AREVA fuel.

16. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." **Rev. 0 April 2003.**

Methodology for TS 3.2.1 - Heat Flux Hot Channel Factor

17 EMF-96-029 (P)(A), "Reactor Analysis System for PWRs." **Rev. 0, January 1997**

Methodology for TS 3.2.1 - Heat Flux Hot Channel Factor

18 BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models). **Rev. 3, December 1996**

Methodology for TS 3.2.1 - Heat Flux Hot Channel Factor

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." **Rev. 0.2, August 2010.**

Methodology for TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor and TS 3.4.1 – RCS Pressure, Temperature and Flow DNB Limits

20 WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO" (Westinghouse Proprietary), **July 2006.**

Methodology added to support fuel transition to Westinghouse RFA-2 fuel. Methodology for TS 2.1.1 – Reactor Core Safety Limits, TS 3.2.1 - Heat Flux Hot Channel Factor.

c. Most of the methodologies included in the TS list are currently used or will be used to develop the cycle specific parameter for the existing AREVA fuel to be used in the mixed cores during the fuel transition. Five of the methodologies currently in the list are being maintained on the list for contingencies. (Items 5, 6, and 7 for Westinghouse fuel and Items 14 and 15 for AREVA fuel)

The methodologies listed in TS 5.6.5.b are presented consistent with GL 88-16 as modified by approved TSTF-363, which permitted the removal of the document revision number and date.

NRC Question 2

It appears that DOM-NAF-2, Rev. 0.1-A, Appendix C, "Qualification of Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," is an approved code since it is part of approved document DOM-NAF-2, Revision 0.1-A (with Appendixes A, B, and C), "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," dated July 2009.

a. Provide the rational for requesting NRC review and approval of the implementation of the Dominion Topical Report VEP-NE-2A, "Statistical DNBR Evaluation Methodology" for Westinghouse RFA-2 fuel.
b. Provide in a flow chart or table a description of the relationship among DOM-NAF-2A, Appendix C to DOM-NAF-2A, and Statistical Design Limit (SDL) including their supporting role to each other.
c. Provide verification and validation data to show the applicability of DOM-NAF-2A to Westinghouse 17x17 RFA-2 fuel.
d. Identify any deviations from the approved methodologies in the submittal dated July 19, 2010.

Dominion's Response

To clarify, the most up to date NRC-approved version of DOM-NAF-2 is Rev. 0.2-P-A, dated August 2010 (Reference 2a).

a. For Dominion to perform thermal-hydraulic calculations to support the determination of departure from nucleate boiling (DNB), Dominion is required to have an NRC-approved critical heat flux (CHF) correlation, an NRC-approved code, and an NRC-approved Statistical Design Limit (SDL). Dominion provided a flow chart in the supplemental information submitted to the NRC in the letter dated September 9, 2010 (Reference 2b). This flow chart and supporting discussion specified the relationship between the NRC-approved VIPRE-D code, the NRC-approved WRB-2M CHF correlation, and the SDL.

   The Westinghouse WRB-2M correlation is an NRC-approved CHF correlation for the determination of departure from nucleate boiling ratio (DNBR) for application to the Westinghouse RFA-2 fuel product. WRB-2M was reviewed and approved by the NRC in WCAP-15025-P-A (Reference 2c). The application of the WRB-2M correlation to the RFA-2 fuel product was performed using the Fuel Criteria Evaluation Process (FCEP) defined in the NRC-approved methodology report WCAP-12488-A (Reference 2d). This methodology allows Westinghouse to perform assessments of changes to fuel products under specific conditions to determine whether the changes can be incorporated without further NRC approvals. The approved RFA-2 fuel product represents the evolution of fuel product updates, starting with the 17x17 Vantage 5H product (Reference 2e). In References 2f through 2k, Westinghouse issued a series of notification letters under the FCEP process to report assessments of the individual fuel changes. Reference 2k specifically documents the applicability (i.e., verification and validation data) to show the applicability WRB-2M CHF correlation to the RFA-2 fuel product. This progression is shown in the first column of the flow chart in Reference 2b.

   VIPRE (Versatile Internals and Components Program for Reactors - EPRI), was 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 2l through 2p). VIPRE-01 was approved by the USNRC (References 2q and 2r). VIPRE-D, the NRC-approved Dominion version of the computer code 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. However, Dominion has not made any modifications to the NRC-approved constitutive models and algorithms contained in VIPRE-01. Dominion submitted Fleet Report DOM-NAF-2 to the NRC for generic review and approval in September 2004 (Reference 2s). DOM-NAF-2 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. DOM-NAF-2-A was reviewed and approved by the NRC in April 2006 (Reference 2t). The most recent issuance of the NRC-approved Fleet Report DOM-NAF-2 is Rev. 0.2-P-A, dated August 2010 (Reference 2a).

The qualification of the NRC-approved VIPRE-D code with the NRC-approved WRB-2M correlation was documented in Appendix C of Fleet Report DOM-NAF-2-A. The NRC reviewed and approved this qualification in a letter dated April 22, 2009 (Reference 2u). The most recent issuance of the NRC-approved Fleet Report DOM-NAF-2 is Rev. 0.2-P-A, dated August 2010 (Reference 2a). This progression is shown in the second column of the flow chart referenced above (Reference 2b).

The final step in this progression is to obtain NRC approval of a SDL for the plant specific application of VIPRE-D/WRB-2M code/correlation set for the Westinghouse RFA-2 fuel at North Anna. The development of the SDL is performed using the NRC-approved methodology of VEP-NE-2-A (Reference 2v). This is shown in the third column of the referenced flow chart.

The Dominion LAR submitted in a letter dated July 19, 2010 requested NRC approval of:

1. Inclusion of Fleet Report DOM-NAF-2-A including Appendix C 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)).
2. The SDL from the implementation of the Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for Westinghouse RFA-2 fuel as per 10 CFR 50.59(c)(2)(vii) it constitutes a Design Basis Limit for a Fission Product Barrier (DBLFPB).

These approvals will allow Dominion to use the VIPRE-D/WRB-2M code/correlation pair to perform 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 design.

b. Dominion provided a flow chart in the supplemental information that was provided to the NRC in a letter dated September 9, 2010 (Reference 2b). This flow chart and the supporting discussion provided the relationship among the NRC-approved VIPRE-D code, the NRC-approved WRB-2M CHF correlation, and the SDL.

c. Refer to the response to Item 2.a.

d. There are no deviations from the approved methodologies in this submittal.

References

2a. Fleet Report, DOM-NAF-2, Rev. 0.2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," August 2010. (Non-Proprietary version ADAMS Accession No. ML102390419).

2b. Letter from J. A. Price (Dominion) to Document Control Desk (USNRC), "Supplemental Information to Support Proposed License Amendment Request (LAR) – Addition of Analytical Methodology to the Core Operating Limits Report (COLR)," September 9, 2010, Serial No. 10-523 (ADAMS Accession No. ML102560291).

2c. Technical Report, WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M for predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," L.D. Smith, et al, April 1999.

2d. Topical Report, WCAP-12488-P-A, 'Westinghouse Fuel Criteria Evaluation Process," October 1994.

2e. Topical Report, WCAP-10444-P-A, "Reference Core Report, Vantage 5 Fuel Assembly," July 1985; Addendum 1-A, March 1986; Addendum 2-A, "Vantage 5H Fuel Assembly," November 1988.

2f. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Transmittal of Presentation Material from NRC/Westinghouse Fuel Design Change Meeting on April 15, 1996," NSD-NRC-96-4694, April 22, 1996.

2g. Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (NRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications," NSD-NRC-97-5189, June 30, 1997.

2h. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application," NSD-NRC-98-5618, March 25, 1998.

2i. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly (RFA) with IFM Grid Design," NSD-NRC-98-5796, October 13, 1998.

2j. 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.

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

2l. 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.

2m. 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.

2n. 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.

2o. Technical Report, EPRI NP-2511-CCM-A Volume 4, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 4: Applications," April 1987.

2p. Technical Report, EPRI NP-2511-CCM Volume 5, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. Volume 5: Guidelines," March 1988.

2q. 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.

2r. 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.

2s. Letter from L. N. Hartz (Dominion) to Document Control Desk (USNRC), "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," September 30, 2004 (ADAMS Accession No. ML042800118).

2t. Letter from C. I. Grimes (USNRC) to D. A. Christian (Dominion), "Approval of Dominion's Fleet Report DOM-NAF-2, 'Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code' (TAC NOS. MC4571, MC4572, MC4573, MC4574, MC4575, AND MC4576)," April 4, 2006 (ADAMS Accession No. ML060790496).

2u. Letter from D. N. Wright (NRC) to D. A. Christian (Dominion), "Appendix C to 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, MD8706, MD8707, MD8708, MD8709)," dated April 22, 2009 (ADAMS Accession No. ML091030634).

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

NRC Question 3

Describe the details of the deterministic design limits (DDLs) and the statistical design limit (SDL) including their definition, relationship, and applicability to the proposed technical specification change.

Dominion's Response

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, deterministic design limits (DDLs) are set equal to each of the applicable code/correlation limits. The DDLs for VIPRE-D/WRB-2M and VIPRE-D/W-3 are shown in Table 4.3-1 of the July 19, 2010 LAR. DOM-NAF-2-A (Reference 3a) describes the method for determining the DNBR design limit for a code/correlation such that DNB will be avoided with a 95% probability at a 95% confidence level for a DNBR equal to this limit. As described in VEP-NE-2-A (Reference 3b), for statepoint and transient analyses that are analyzed deterministically, the initial condition for each parameter (pressure, temperature, flow, power, etc.) is assumed to be simultaneously and continuously at the worst point in its uncertainty range with respect to the DNBR. As stated in Attachment 1, Section 1 of the LAR, the DDLs documented in Appendix C of DOM-NAF-2-A for the VIPRE-D code and the WRB-2M and W-3 critical heat flux (CHF) correlation sets have already been approved by the NRC.

For statistical DNB analyses, Attachment 4, Section 3 of the LAR describes the development of the statistical design limit (SDL) for application of VIPRE-D/WRB-2M to the Westinghouse RFA-2 fuel design at North Anna. The method for determining the SDL as described in Attachment 4 of the submittal uses the NRC-approved methodology in VEP-NE-2-A. A SDL of 1.25 is defined in the LAR. Even though the SDL is larger than the DDL, its use is advantageous as the Statistical DNBR Evaluation Methodology permits the use of nominal values for operating conditions instead of requiring the application of evaluated uncertainties to the initial conditions for statepoint and transient analysis. As stated in Attachment 4, Section 5 of the LAR, the SDL is a design basis limit for a fission product barrier (DBLFPB). As such a change to the SDL requires NRC review and approval consistent with 10 CFR 50.59(c)(2)(vii). Approval of the SDL and inclusion of Appendix C of DOM-NAF-2-A into the North Anna COLR will allow Dominion to perform licensing calculations with the VIPRE-D/WRB-2M and W-3 CHF correlations for Westinghouse RFA-2 fuel at North Anna Power Station Units 1 and 2.

Safety analysis limits (SALs) are used in the performance of DNB thermal-hydraulic evaluations. These self-imposed SALs are set above the applicable design DNBR limit (SDL or DDL) to give a set amount of retained DNBR margin by the following formula:

$$Retained\ DNBR\ Margin = \frac{(SAL - DNBR\ limit)}{SAL} \cdot 100\%$$

The SALs and Retained DNBR Margins are described in Attachment 4, Section 4.3 of the LAR. This method of defining retained DNBR margin allows for the retained DNBR margin to be found in a single, clearly defined location. The retained DNBR margin can be used to offset DNBR penalties to account for the DNB effect due to changes in the fuel product (e.g. transition core penalties), plant operating conditions, or analysis methodology (e.g., fuel rod bowing).

VEP-NE-2-A describes the application of retained DNBR margin to thermal hydraulic calculations. As stated in the LAR, 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 the applicable statistical and deterministic DNB analyses.

The DDLs and SDL are two of the DBLFPB described in Reference 3c. The DDLs and SDLs are fixed and any changes to their value require USNRC review and approval. However, the SALs for deterministic and statistical DNB analyses may be changed without prior USNRC review and approval provided the changes meet the criteria established in Reference 3c.

References

3a. Fleet Report, DOM-NAF-2, Rev. 0.2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," August 2010. [Non-Proprietary version ADAMS Accession No. ML102390419]

3b. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

3c. Technical Report, NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," Nuclear Energy Institute, November 2000.

NRC Question 4

It appears that there are various SDLs and DDLs resulting from [sic, used in] transient analysis. Describe how to apply these values to finalize the DNBR for North Anna Unit 1 Cycle 23 and Unit 2 Cycle 23 operation.

Dominion's Response

See Dominion's Response to NRC Question #3 for a discussion of DDLs, SDL, and SALs.

The Statistical DNBR Evaluation Methodology of VEP-NE-2-A (Reference 4a) is applied to the 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 of the July 19, 2010 LAR. This table also indicates which events are analyzed deterministically; RWFS, accidental depressurization of the main steam system, and rupture of a main steam pipe (i.e., main steam line break or MSLB).

Dominion assesses the impact of reload operation on NSSS accident analyses using the methodology of Topical Report VEP-FRD-42, Revision 2.1-A, "Reload Nuclear Design Methodology" (Reference 4b). This methodology defines a set of key analysis parameters that fully describe a valid conservative safety analysis - the "reference analysis." If the key analysis parameters for a reload core are bounded by the corresponding parameters in the reference analysis, the reference safety analysis is bounding, and further analysis of the reload core is unnecessary. When a key analysis parameter for the reload is not bounded, further evaluation is necessary to ensure that the required safety margin is maintained. This latter determination is made either through a complete reanalysis of the transient, or through a simpler, though conservative, evaluation process using known parametric sensitivities. Should a re-analysis be required, the DNBR limits identified in the LAR would be used as discussed above.

References

4a. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

4b. Topical Report, VEP-FRD-42-A, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.