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

February 29, 2012

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

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ADDITION OF AN ANALYTICAL METHODOLOGY TO CORE OPERATING LIMITS REPORTS FOR THE CRITICAL HEAT FLUX CORRELATION (TAC NOS. ME4262 AND ME4263)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos.266 and 247 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (NAPS), Units 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated July 19, 2010, as supplemented September 9, 2010, January 26, May 16, and June 23, 2011.

These amendments revise TS 5.6.5.b, "Core Operating Limits Report (COLR)," to include "Appendix C, Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code." This would allow NAPS to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 correlation pairs to perform licensing calculations with Westinghouse Robust Fuel Assembly (RFA-2) fuel in the cores.

TS 5.6.5, documents the approved methodologies used in the development of the COLR operating limits. It identifies the current approved methodologies used for the current core reload. Therefore, the NRC expects that the licensee will submit an administrative change to TS 5.6.5 removing outdated references which are not used explicitly in the development of core operating limits. Specifically, the AREVA topical reports no longer used to support COLR listed operating limits.

TS 4.2.1, "Fuel Assemblies," is intended to identify the fuel assembly designs used in the current core reloads. Therefore, the NRC expects that the licensee will maintain these TS up to date.

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

Sincerely,

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

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No.266 to NPF-4
2. Amendment No.247 to NPF-7
3. Safety Evaluation

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 266
Renewed License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated July 19, 2010, as supplemented September 9, 2010, January 26, May 16, and June 23, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 266, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: February 29, 2012

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
Renewed License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated dated July 19, 2010, as supplemented September 9, 2010, January 26, May 16, and June 23, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 247, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: February 29, 2012

ATTACHMENT

TO LICENSE AMENDMENT NO. 266

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND

TO LICENSE AMENDMENT NO. 247

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

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

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

(2) Pursuant to the Act and 10 CFR Part 70, VEPCO to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or component; and

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

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

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal).

(2) Technical Specifications

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

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations as set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2893 megawatts (thermal).

(2) Technical Specifications

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

(3) Additional Conditions

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

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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."

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

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

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

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 266

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 247

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By application dated July 19, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102020165), and supplemented by additional information dated September 9, 2010, January 26, 2011, and May 16, 2011 (ADAMS Accession Nos ML102560291, ML110270089, ML111370137), Virginia Electric Power Company (the licensee) requested changes to the Technical Specifications (TSs) of North Anna Power Station, Units 1 and 2 (NAPS). The proposed change would revise TS 5.6.5.b, "CORE OPERATING LIMITS REPORT" to include the U.S. Nuclear Regulatory Commission (NRC)-approved methodology, Appendix C of the Dominion Fleet Report DOM-NAF-2-A, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code." The addition of the above-approved methodology to TS 5.6.5.b would allow the licensee to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 correlation pairs to perform licensing calculations with Westinghouse Robust Fuel Assembly (RFA-2) fuel in the NAPS cores.

The licensee intends to reload the NAPS cores with Westinghouse 17x17 RFA-2 fuel assemblies commencing with Cycle 23 for both units. The Westinghouse RFA-2 fuel product is a replacement for the resident fuel product, the AREVA Advanced MARK-BW (AMBW) fuel assemblies. The proposed amendment would permit the licensee to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 core/correlation pairs to perform licensing calculations for the Westinghouse RFA-2 fuel in the NAPS cores, using the deterministic design limits (DDLs) documented in Appendix C of the DOM-NAF-2-A Fleet Report (Reference 4) and the statistical design limit (SDL) documented in Reference 5.

The supplement dated September 9, 2010, and the additional information dated January 26, 2011, and May 16, 2011, contained clarifying information only and did not change the initial no significant hazards determination or expand the scope of the initial application.

2.0 REGULATORY EVALUATION

2.1 Background

VIPRE-D is the licensee version of the computer code VIPRE-01 that was originally developed for Electric Power Research Institute (EPRI) by the 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) for reactor cores. VIPRE-01 was approved by the NRC (Reference 14). VIPRE-D is based on VIPRE-01 and is customized by the licensee to fit specific needs of Dominion's nuclear plants and fuel products. VIPRE-D was previously approved by the NRC staff (References 4 and 6). In Appendix C of DOM-NAF-2 topical report, the NRC staff approved the use of the Westinghouse WRB-2M CHF correlation (Reference 7) for the Westinghouse RFA fuel design with or without the intermediate flow mixer (IFM) grids. The NRC staff has determined that the WRB-2M correlation implemented in the VIPRE-D code provided a conservative predicted CHF value with no bias or trends in the prediction of CHF. In addition, the licensee was approved to use VIPRE-D/W-3 code/correlation pair (Reference 6), since W-3 was one of the CHF correlations contained in the NRC-approved generic version of VIPRE-01 (Reference 21), and it has already been approved for use with VIPRE-D code.

2.2 Proposed Technical Specification Core Operating Limits Report Change

Analysis of the RFA-2 fuel design at NAPS units will require a revision to the plant Technical Specification (TS) 5.6.5.b, Core Operating Limits Report (COLR) by adding the licensee-specific analytical method used in the determination of core operating limits that are applicable to the Westinghouse RFA-2 fuel product:

> *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.*

2.3 Regulatory Requirements and Guidance Documents

The Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800), Section 4.2, "Fuel System Design," provides regulatory guidance for the review of fuel rod cladding materials and the fuel system. In addition, the SRP provides guidance for compliance with the applicable General Design Criteria (GDC) in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50. According to SRP Section 4.2, the fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOO),
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

The NRC staff reviewed the LAR to evaluate the applicability of the WRB-2M and W-3 CHF correlations to the NAPS TSs, confirm that the use of this methodology is within NRC approved ranges of applicability, and verify that the results of the analyses are in compliance with the requirements of the following GDC specified in Appendix A to 10 CFR Part 50:

- GDC-10, "Reactor Design," requiring the reactor design (reactor core, reactor coolant system (RCS), control, and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs.
- GDC-15, "Reactor Coolant System Design," requiring the RCS and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the RCS boundary are not exceeded during any condition of normal operation, including AOOs.
- GDC-20, "Protection System Functions," requiring the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.
- GDC-35, "Emergency Core Cooling," requiring a system to provide abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

3.0 TECHNICAL EVALUATION

The licensee is requesting approval for the addition of Appendix C of the Fleet Report DOM-NAF-2-A to the NAPS TS 5.6.5.b list of USNRC-approved methodologies used to determine core operating limits. Approval of this LAR will enable the licensee to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pairs to perform licensing calculations for Westinghouse RFA-2 fuel in NAPS 1 and 2 cores.

Starting with Cycle 17 of Unit 2 and Cycle 18 of Unit 1, the licensee transitioned to Advanced Mark-BW fuel supplied by Framatome ANP. Beginning with Cycle 19 for Unit 2 and Cycle 20 for Unit 1, NAPS core were fully loaded with Advanced Mark-BW fuel. The licensee plans to use Westinghouse RFA-2 fuel in NAPS Units commencing with Unit 1, Cycle 23 (Spring 2012) and with Unit 2, Cycle 23 (Spring 2013). The licensee has requested to modify the NAPS TS to include Appendix C of the Fleet Report, DOM NAF-2-A (Reference 6) to the TS 5.6.5.b COLR list of NRC approved methodologies used to determine the core operating limits. The license amendment request (LAR) would enable the licensee to use the VIPRE-D/WRB-2M and VIPRE-D/W-3 code/correlation pair to perform licensing calculations of Westinghouse RFA-2 fuel in NAPS 1 and 2 cores using the DDLs in Appendix C of Reference 6 and the SDLs documented in Reference 5.

The NRC staff has reviewed the LAR in conjunction with the supplemental information, the responses to the staff's requests for additional information (RAIs) to: (1) evaluate the acceptability of the licensee transition to Westinghouse RFA-2 fuel, (2) evaluate the use of the associated Westinghouse and Dominion methodologies for licensing applications, and (3) confirm adequate technical basis for the proposed TS changes. In addition, the staff performed an audit (Reference 8) and reviewed the NAPS specific safety analyses, calculation notebooks and associated fuel transition methodologies.

3.1 Westinghouse Robust Fuel Assembly – 2 (RFA-2) Fuel Design

Topical Report WCAP-12488-A (Reference 9) incorporates a process and criteria to apply changes or improvements in existing Westinghouse fuel designs that does not require NRC review and approval when these criteria are satisfied. The Fuel Criteria Evaluation Process (FCEP) assesses the safety significance and assures the NRC's approval of pressurized water reactor (PWR) fuel system mechanical design changes by confirming that appropriate criteria related to the fuel system are satisfied. Implementation of this process allows licensees to make design changes provided it meets the criteria specified within WCAP-12488-A. In 1998, Westinghouse notified the NRC of the introduction of the RFA with modified low-pressure drop (LPD) structural mixing vane mid-grid and modified intermediate flow mixer (IFM) grid used on the design (Reference 10). The RFA design introduced fuel for the 17x17 12-foot core (Reference 11). The original RFA design was a modified V5H mid-grid that restored departure from nucleate boiling margins and eliminated fuel assembly vibration. The RFA fuel significantly reduced the grid to fuel rod fretting that was observed in the V5H fuel assembly design. The RFA-2 grid is a modification to RFA to further improve its resistance to fuel rod fretting wear. Reference 11 addresses the design categories, peak rod average burnup limit and associated parameters in the Westinghouse FCEP (Reference 9) and has shown that the RFA-2 design changes have an insignificant impact on these parameters.

3.1.1 Fuel Mechanical-Thermal Performance Analysis

Westinghouse used their fuel performance code, PAD 4.0 for thermal and mechanical design analysis of RFA-2 fuel. The Westinghouse PAD model is a best estimate fuel rod performance model used for both fuel rod performance analysis and safety analysis input (Reference 17). The PAD code consists of several fuel rod performance models integrated to predict fuel temperature, rod pressure, fission gas release, cladding elastic and plastic behavior, cladding growth, cladding corrosion, fuel densification, and fuel swelling as a function of linear power and time.

Westinghouse made several improvements in the PAD 4.0 code compared to its previous version, PAD 3.4. These model changes are to the cladding irradiation growth, Zr-4 and ZIRLO clad thermal conductivity, Zr-oxide thermal conductivity, equation of state gas pressure, the oxide metal ratio, and Zr-4 clad gas absorption models. Audit calculations were performed with the NRC-developed FRAPCON-3 code (References 18 and 19) for comparison to the examples provided in PAD 4.0 licensing analysis. The staff upon review of the Westinghouse improved PAD 4.0 fuel performance code as described in WCAP-15063 (Reference 17), concluded that PAD 4.0 is acceptable for fuel licensing application up to a rod average burnup of 62 GWd/MTU. However, the PAD 4.0 was found to predict lower temperatures than FRAPCON-3 at high burnups due the lack of a burnup dependent fuel thermal conductivity model in PAD 4.0.

3.1.2 Fuel Thermal Conductivity Model in PAD 4.0

As indicated in Section 3.1.1 of this safety evaluation, PAD 4.0, the Westinghouse fuel performance code lacks a fuel thermal conductivity model that is fuel burnup dependent. A comparison of the FRAPCON-3 calculated centerline and average fuel temperatures to those from PAD 4.0 at LHGRs typical for LOCA initialization demonstrates that PAD 4.0 predicts higher temperatures very early in core life. This difference is reduced with increasing burnups such that the PAD 4.0 code prediction is similar at moderate burnups, and PAD predicts lower fuel temperatures than FRAPCON-3 at high burnups. The reason why the PAD 4.0 code thermal predictions are lower at high burnups is because the FRAPCON-3 code has a fuel thermal conductivity model that is burnup dependent (lower fuel conductivity with increasing burnup) while the PAD 4.0 code has a thermal conductivity model with no burnup dependence.

One of the audited documents associated with "PAD 4.0, Fuel Temperatures for North Anna Units 1 and 2 RSTR Program," indicates that the RFA-2 fuel temperatures are generated using PAD 4.0 fuel performance code using fuel design parameters and transient limits (Reference 8). NRC staff's safety evaluation of topical report, WCAP-15063 (Reference 16) indicates that a comparison of NRC audit code FRAPCON-3 calculated centerline and average fuel temperatures to those from PAD 4.0 show higher temperatures at early in core life. Further, the staff concluded that PAD 4.0 thermal predictions are lower at high burnups. The staff concluded that while FRAPCON-3 has a burnup dependent fuel thermal conductivity model, the PAD 4.0 code has no burnup dependent fuel thermal conductivity model.

Pursuant to Appendix K to 10 CFR 50 and Criterion 10 and 35 of Appendix A to 10 CFR 50, the licensee is expected to incorporate a methodology for calculating the highest cladding and fuel temperatures and thereby the highest calculated stored energy in the fuel during any condition of normal operation including AOOs and for ECCS evaluation models. This methodology should include the evaluation of thermal conductivity of the fuel as a function of burnup and temperature taking into consideration all of the effects that take place in the fuel during irradiation including but not limited to solid fission product buildup both in solution and as precipitates, porosity, and fission gas-bubble formation. This evaluation shall include the effects of thermal conductivity on all fuel rod thermal-mechanical analyses (e.g. rod internal pressure) and inputs to downstream safety analyses (e.g. loss-of-coolant accident (LOCA) stored energy).

3.1.3 Thermal Conductivity Degradation Evaluation

Westinghouse initiated modeling changes in PAD 4.0 code to address the impact of thermal conductivity degradation (TCD) on fuel performance and the overall impact on LOCA and non-LOCA transients. The model change that addresses the TCD as recommended by the Halden Project has been incorporated in to the Westinghouse STAV fuel performance code (Reference 24). The impact of the TCD on PAD was assessed by using the functional form and the coefficients in the STAV code (References 24 and 25). Westinghouse evaluated the impact of TCD on fuel rod design criteria, such as, fuel melt limit, rod internal pressure criteria, transient clad stress/strain, steady state strain, clad fatigue.

The NRC staff conducted a regulatory audit of all Westinghouse and NAPS calculation documents that were impacted by the TCD with burnup model. During the audit, the staff reviewed the following Westinghouse calculations supporting the NAPS TCD safety assessment and fuel thermal-mechanical design (Reference 25).

The licensee performed a comprehensive review of all safety analyses potentially impacted by TCD (Reference 25). Analysis areas that showed minimal or no impact are: fuel thermal hydraulic design, core physics and core design, fuel rod mechanical design, non-LOCA transients excluding control rod ejection events, small break LOCA, long term core cooling after a LOCA, containment pressure for main steam line break (MSLB), radiological consequences, and steam generator tube rupture (SGTR).

The analyses that were impacted by TCD were identified as containment pressure for LOCA, fuel centerline temperature and hot spot fuel enthalpy for reactivity insertion accidents (RIA) such as spectrum of control rod ejection events, and LBLOCA (Reference 25).

Based on the NRC staff's review of the audit documents related to the impact of TCD on NAPS mechanical and thermal performance of Westinghouse fuel, impact of TCD on transients and accident analyses at North Anna, the staff concludes that the predicted results are acceptable subject to requirements described in Section 3.7.

3.1.4 Conclusion

Upon review of the evaluation of Westinghouse FCEP for RFA-2 fuel design change and their evaluation of RFA-2 using the modified fuel performance code, PAD 4.0 (PADTCD), the NRC staff concludes that NAPS be permitted to load RFA-2 fuel with the peak rod average burnup limit of 62 GWd/MTU.

3.2 Critical Heat Flux Correlations

In References 10 and 11, Westinghouse notified the NRC of the applicability of WRB-1 and WRB-2 DNB correlations (Reference 12) to the 17x17 modified LPD mid-grids and modified IFM grid used on the RFA fuel design. Additional CHF tests were performed for the new design and a new critical heat flux CHF correlation, WRB-2M was developed (Reference 13) from Westinghouse rod bundle test data. WRB-2M incorporates a simple functional multiplier applied to the original WRB-2 correlation presented in Reference 12. Therefore the WRB-2M can be considered as a minor perturbation of WRB-2 and more accurately reflects the test data from the rod bundles containing the modified grids.

The NRC staff approved the licensee's request to use the WRB-2M correlation along with VIPRE-D thermal-hydraulic code to predict the CHF behavior of modified 17x17 V5H fuel with or without the modified IFM grids, referred to as RFA-2 fuel design (Reference 4). VIPRE-D code is the licensee version of the original VIPRE-01 code (Reference 14). The NRC staff has accepted DNBR limits that ensure a 95% probability that CHF will not occur with a confidence of 95% for the hottest pins of the reactor core. A DNBR limit of 1.14 was derived for the VIPRE-D code using the WRB-2M CHF correlation to meet this criterion.

The range of applicable fluid and fuel parameters is defined in the original WRB-2M topical report (Reference 13).

The WRB-2M range of applicability is reproduced in Table 1 below.

Table 1: WRB-2M Range of Applicability

Parameter	WRB-2M Range
Pressure (psia)	1495 - 2425
Local mass velocity (Mlbm/hr-ft^2)	0.97 – 3.1
Local quality	0.1 – 0.29
Heat length, inlet to CHF location (ft)	≤ 14
Grid spacing (inches)	10 – 20.6
Equivalent hydraulic diameter (inches)	0.37 – 0.46
Equivalent heated diameter (inches)	0.46 – 0.54

The VIPRE-D/WRB-2M code/correlation pair together with the approved statistical DNBR methodology (Reference 5) will be applied to Condition I and II events (except rod withdrawal from subcritical, RWFS), 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 bypass flow (Reference 8), measurement component of nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$)[1] and engineering enthalpy rise hot channel factor ($F_{\Delta H}^{E}$)[2] uncertainties. The application of statistical methodology in the NAPS calculations are briefly discussed in Section 3.4 of this safety evaluation. The NRC staff has verified during the regulatory audit (Reference 8) that the uncertainties are convoluted statistically into the DNBR limit.

In addition to the WRB-2M correlation, NAPS was approved to use W-3 CHF correlation in the DNBR calculations for the Westinghouse RFA-2 product (References 14 and 15). For NAPS, the W-3 correlation will be used when conditions fall outside the range of the WRB-2M correlation. Specifically, the W-3 correlation will be applied to the lower portion of the fuel assemblies in the RWFS event because of the bottom peaked axial power profile assumed and the MSLB event because of the lower pressures encountered. For the MSLB event, the DNBR limit of 1.45 will be used for pressures 500 to 1000 psia and a limit of 1.30 will be used for pressures above 1000 psia (Reference 16). The W-3 CHF correlation shall always be used deterministically. Details of application of the WRB-2M and W-3 correlations and the corresponding retained DNBR margin are summarized in Section 3.4 of this safety evaluation.

The NRC staff concludes that based on the information provided by the licensee in the LAR,

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1 Nuclear enthalpy rise hot channel factor is the ratio of enthalpy rise in hottest core channel to the average enthalpy rise in core channel.

2 Engineering enthalpy rise hot channel is an allowance on heat flux required for manufacturing tolerances. The engineering factor accounts for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. The engineering factor for NAPS core is quantified as a normal probability distribution with a magnitude of ±3.0%.

additional information and the documents presented during the regulatory audit (Reference 8), the proposed addition to the TS 5.6.5.b. list of NRC approved methodologies COLR of Appendix C of Dominion fleet report, DOM-NAF-2-A is acceptable.

3.3 Uncertainty Analysis

3.3.1 North Anna Plant Parameter Uncertainties

The licensee has submitted detailed results from its calculations of uncertainties in the statistically treated plant parameters in accordance with the condition number 2 in the safety evaluation report for Reference 5. Consistent with the approved statistical DNBR methodology, inlet temperature, pressurizer pressure, core thermal power, reactor vessel flow rate, core bypass flow, the $F_{\Delta H}^{N}$ and the $F_{\Delta H}^{E}$ were selected as the statistically treated parameters in the implementation analysis (Reference 3). The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and core inlet temperature were determined 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 a two-sided 95% probability range.

Details of the uncertainty and margin results are listed in Reference 3. The NRC staff has reviewed the detailed licensee calculations during the regulatory audit (Reference 8). Results of these uncertainty calculations are listed in Table 3.2-1 of Reference 1 and repeated below in Table 2.

Table 2 North Anna Parameter Uncertainties

PARAMETER	Nominal Value	Standard Deviation	Uncertainty	Distribution
Pressure (psia)	2250 psi	15.31	±30.0 psi at 2σ	Normal
Temperature (°F)	553.7	2.143°F	±4.2°F at 2σ	Normal
Power [MWt]	2940	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%	Normal
Bypass [%]	5.5	N/A	±1.0%	Normal

3.3.2 CHF Correlation Uncertainty Factor

In compliance with the Section 2.4 of Statistical DNBR Evaluation Methodology (VEP-NE-2-1) (Reference 5), each of the calculated DNBRs is required to be multiplied by a random variable to include the effect of the correlation uncertainty. The WRB-2M CHF correlation was developed from CHF data obtained at the Columbia University HTRF using full-scale, electrically heated rod bundle test sections. The licensee qualification of WRB-2M in VIPRE-D was performed against the same test data from the Columbia-EPRI CHF database for Westinghouse 17x17 fuel. The licensee used the CHF experimental data used by Westinghouse to develop the WRB-2M CHF correlation (Reference 4).

Each test section was modeled for analysis with the VIPRE-D thermal-hydraulic computer code as a full assembly model following the modeling methodology. For each set of bundle data, VIPRE-D produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc.) at every axial node along the heated length of the test section. The ratio of measured-to-predicted CHF (M/P) is the variable that was used to evaluate the thermal-hydraulic performance of a code/correlation pair. The measured CHF is the local heat flux at a given location, while the predicted CHF is calculated by the code using the WRB-2M CHF correlation. Typically, the M/P distribution is found to be normally distributed, but not the reciprocal DNBR distribution itself. As a result, the randomizing factor was to reflect the normality of the M/P distribution.

The correlation uncertainty factor was applied to all 2000 calculated DNBRs at each of the nominal statepoints (pressurizer pressure, inlet temperature, power, flow rate, and nuclear enthalpy rise factor) to generate the Randomized DNBR distribution. The nine nominal statepoints that are listed in Table 3.6-1 of Reference 1 cover the full range of normal operation and anticipated operational occurrences (transients) AOOs. The statepoints span the range of conditions over which the statistical methodology is 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 (62.98%) was also included.

This detailed Monte-Carlo analysis consisted of 2000 calculations performed around each of the nine nominal statepoints. The standard deviation at each nominal statepoint was augmented by the code/correlation uncertainty (Section 3.3.2), the small sample correction factor, and the code uncertainty (Section 3.3.3) to obtain a total DNBR standard deviation. The randomized DNBR distribution from the unrandomized DNBR results is obtained by correcting for the code/correlation uncertainty using the equation:

$$DNBR_{Randomized} = \frac{DNBR_{Unrandomized}}{[1.0 + s\left(\frac{M}{P}\right) * K * NormalRandomNumber]}$$

Where:

- S(M/P) is the standard deviation of the code/correlation M/P database for the WRB-2M CHF correlation taken from Appendix C of DOM-NAF-2-A (Reference 4),
- K is the correction factor that depends on the size of the experimental database used to obtain the code/correlation deterministic DNB limit which is calculated as:

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

The 95% upper confidence limit on the CHF correlation is represented by the product of the standard deviation of the code/correlation M/P database and the sample correction factor in the equation for DNBR $_{Randomized}$.

Table 3 lists the CHF code/correlation Data (Reference 1)

Table 3: CHF Code/Correlation Data

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

The NRC staff has reviewed detailed calculations during the regulatory audit. The staff finds the methodology and results acceptable.

3.3.3 Code Uncertainty

The thermal hydraulic code uncertainty is applied to account for the effect of analyzing a full core with a correlation, which was based only upon steady state test bundle data, and the effect of performing analyses with the a specific code and a specific correlation. The code uncertainty of 5% discussed in Section 2.5 of Reference 5 was originally based on comparisons between the licensee's COBRA code and another NRC-approved Westinghouse thermal-hydraulic code (THINC) and comparisons made between COBRA and the W-3 correlation experimental database. The VIPRE-D code was compared against AREVA's LYNXT thermal-hydraulic code in Section 5.0 of Reference 4 with the maximum difference in predicted DNBRs between the two codes being less than 5%. In Appendices of DOM-NAF-2, comparisons have been performed between VIPRE-D and different CHF correlation experimental databases; such as Figures A.4-1 in Appendix A, Figure B.6-2 of Appendix B, and Figure C.5-1 in Appendix C. Based on the comparison of VIPRE-D/LYNXT and VIPRE-D/CHF experimental database, the 5% code uncertainty is found appropriate for use with VIPRE-D code.

3.3.4 Model Uncertainty

A model uncertainty accounts for the difference between the simple core thermal-hydraulics model with which the correlation DNBR limit was derived and the sophisticated model which is normally used for production calculations. This model uncertainty satisfies the Condition 3 in the SER for the Statistical DNBR Evaluation Methodology topical report, VEP-NE-2-A. The VIPRE-D 20-channel production model was used for North Anna with the 17x17 RFA-2 fuel assemblies for the development of the VIPRE-D/WRB-2M code/correlation pair SDL for North Anna. Since this production model is intended to be used for all NAPS evaluations, no additional uncertainty is necessary, and the model uncertainty term is set to zero for the calculation of the total DNBR standard deviation.

3.3.5 Total Uncertainty and Full-Core DNB Probability Summation

The Monte Carlo analysis consisted of 2000 calculations performed around each of the nine nominal statepoints. 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 uncertainty, s_{TOTAL} is obtained using the root-mean-square (RMS) according to equation 3.2 of Reference 1, and reproduced below:

$$S_{TOTAL} = [\ s^2_{DNBR}\ \{\ 1.0 + \surd[\surd(\surd(n\text{-}1)/x^2) - 1.0]^2 + (1/N)\}^2 + F_c^{\ 2} + F_M^{\ 2}]^{1/2}$$

Where:

- S_{DNBR} is the standard deviation for the Randomized DNBR distribution.
- The factor $[\surd(\surd(n\text{-}1)/x^2)$ is the uncertainty in the standard deviation of the 2000 Monte-Carlo simulations, and provides a 95% upper confidence limit on the standard deviation.
- 1/N is 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, defined as 5% (see Section 3.3.3).
- F_M is the model uncertainty, which is 0.0 (see Section 3.3.4).

The limiting peak fuel rod SDL was calculated to be 1.243 for VIPRE-D/WRB-2M code/correlation pair (Table 3.6.2 of Attachment 4, Reference 1).

The location of the minimum DNBR depends on the axial power profile and the value of the DNBR depends on the enthalpy rise to that point. The maximum value of the rod integral is used to identify the most likely rod for minimum DNBR. Section 3.7 of Attachment 4 to Reference 1, supported by the calculations reviewed at the audit (Reference 8), describes the procedure that analyzes the DNB sensitivity to rod power. A representative fuel rod census curve is used for the determination of the SDL of DNBR for a maximum nuclear enthalpy rise factor, $F\Delta h$ of 1.587 (Table 3.7.2 of Attachment 4 to Reference 1). The full-core DNB probability summation is reevaluated on a cycle reload basis to verify the applicability of the fuel rod census used in the implementation of the DNBR limit. This procedure resulted in an SDL limit of 1.247 for VIPRE-D/WRB-2M code/correlation pair
(Table 3.7.3 of Attachment 4 to Reference 1).

The NRC staff has determined that the licensee's procedure in the determination of CHF correlation uncertainties and their appropriateness is acceptable.

3.4 <u>Application of VIPRE-D/WRB-2M/W-3 to NAPS</u>

VIPRE-D/WRB-2M code/correlation pair together with the Statistical DNBR Evaluation Methodology is applied to Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWSC), and to the Complete Loss of Flow event and the Locked Rotor Accident. 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. W-3 CHF correlation is used when the thermodynamic conditions are outside the range of validity of the WRB-2M CHF correlation, such as, the MSLB evaluation where there are reduced pressure and temperature.

Table 4 lists the DNBR limits for WRB-2M and W-3 CHF correlations.

Table 4: 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 (≥1000psia)	1.30
DDL (≤ 1000psia)	1.45
SAL (≥ 1000psia)	1.44
SAL ((≤ 1000psia)	1.61

The SDL for NAPS cores that contain Westinghouse 17x17 RFA-2 fuel is 1.25 (Section 3.3.5). The SDL provides a peak fuel rod protection with at least 95% probability at 95% confidence level and a 99.9% DNB protection for the full core. This SDL is plant specific since it includes all the applicable uncertainties for the key parameters treated in the statistical DNBR evaluation methodology (Reference 5). Therefore, this limit shall be applied to statistical DNB events (Table 3.9-1 of Attachment 4 to Reference 1) with the Westinghouse RFA-2 fuel.

For deterministic DNB analyses, (Table 3.9-1 of Attachment 4 to Reference 1) the design DNBR limit shall be set equal to DDL. The DDLs and SDLs are fixed and any changes to their values require NRC review and approval. The safety analysis limits (SALs) for deterministic and statistical DNB analyses (SAL_{DET} and SAL_{STAT}, respectively) may be changed without prior NRC approval since they are stipulated in the COLR.

A SAL is equal to 1.55 has been selected for 17x17 RFA-2 fuel at NAPS with the VIPRE-/WRB-2M code/correlation pair. This SAL shall be 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 (Reference 8).

The retained DNBR margins that are listed in Table 5 are the differences between the SALs and the design limits.

Retained DNBR Margin (%) = (SAL – DDL)/SAL.

Table 5: DNBR Limits and Retained DNBR Margin for Deterministic and Statistical 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
Statistical DNB Applications			
DNB Correlation	SDL	SAL_{STAT}	Retained DNBR Margin (%)
WRB-2M	1.25	1.55	19.3

The licensee has stated in Reference 1 (Attachment 1 Section 4.0, and Attachment 4, Section 5) that the existing Reactor Core Safety Limits and protection functions (Overtemperature Delta-T (OTΔT), Overpower Delta-T (OPΔT), and FΔI) did not require revision as a consequence of implementation of the proposed LAR. However, the licensee has determined that a change is required to the FΔI reset function for the OTΔT trip function in order to accommodate representative reload power shapes for Westinghouse RFA-2 fuel with the VIPRE-D code and the Westinghouse WRB-2M and W-3 CHF correlations (Reference 20). In Westinghouse designed reactors such as North Anna Units, these trip functions are designed to provide protection against fuel centerline melting by limiting the linear heat generation rate (LHGR) and DNBR. These trip functions also assure that the vessel temperature rise is proportional to the core power during postulated transients.

The bases for these trip functions are the core thermal limit line, axial offset envelopes, and other reactor coolant system and plant parameters. The core thermal limit lines are defined as most limiting vessel exit boiling, hot channel exit quality, and core DNB considerations. The existing axial offset envelopes are based on analysis of bounding power shapes from previous analytical experience with Westinghouse 17x17 Vantage 5H fuel and AREVA AMBW fuel with the applicable CHF correlations and core thermal hydraulic analyses. For cycle specific evaluations, representative reload power shapes for future operating cycles were evaluated for the Westinghouse 17x17 RFA-2 fuel with the VIPRE-D code and the Westinghouse WRB-2M and W-3 CHF correlations (Reference 20). The licensee's evaluation resulted in a significant number of unbounded bottom-skewed power shapes. This resulted in a change of the FΔI reset function (NAPS 1 and 2 Technical Specifications, Table 3.3.1-1 and) for OTΔT trip function in order to accommodate these reload power shapes for the Westinghouse RFA-2 fuel with the VIPRE-D and the Westinghouse WRB-2M and W-3 correlations.

The values that define the reset FΔI function appear in the NAPS COLR. Therefore, the licensee intends to implement the change to the FΔI function under the provisions of 10 CFR 50.59 using Dominion's NRC approved reload design methodology in VEP-FRD-42-A (Reference 22) and the NRC approved methodology in WCAP-8745-P-A (Reference 23). The NRC staff found the implementation of the change to the FΔI reset function acceptable.

3.5 Thermal-Hydraulic Compatibility and Transition Core Penalties

The proposed new RFA-2 fuel will be co-resident with AREVA Advanced Mark-BW (AMBW) fuel that has different thermal and hydraulic characteristics from that of the RFA-2 fuel design. During the audit, the NRC staff reviewed all the documents which are listed in Attachment 1, Item Number 2 for RFA Fuel Transition in the Regulatory Audit Report (Reference 8). The analyses reviewed by the NRC staff are axial fit-up, grid growth, axial and cross flow, lift force, fuel temperature, core stored energy, and flow stability. The NRC staff reviewed the calculations and found that the analyses confirm that thermal-hydraulic compatibility criteria are met for limiting core configurations.

NAPS Cycles 23 and 24 will have mixed core configurations which consist of RFA-2 fuel assemblies co-resident with AREVA Advanced Mark-BW fuel assemblies. RFA-2 and AMBW fuel designs exhibit different thermal-hydraulic characteristics, including pressure drop and grid characteristics. Section 4.3 of the license amendment request (Reference 1) indicates that the available retained DNBR margin is the difference between the self-imposed safety analysis limit and the design limit. DNBR penalties that contribute to retained margin are classified as: (1) generic fuel design issues (e.g., fuel rod bow, transition core), (2) cycle-specific violations of limits (e.g., unbounded power shapes or peaking factors), and (3) plant operating conditions. Reload thermal hydraulic evaluations that were available at the regulatory audit (Attachment 1 of Reference 8) included loss coefficients, mixed core penalty, thimble bypass flow and AREVA's calculations for mixed core penalty. The NRC staff reviewed the calculations at the audit and find that the analyses confirm appropriate margin in the DNBR calculations to account for the mixed core penalty.

3.6 Impact of August 23, 2010, Earthquake on NAPS Fuels and Related Systems

On August 23, 2011, with the NAPS, operating at 100% power, the site experienced ground motion from a seismic event (a Magnitude 5.8 earthquake reported by the U.S. Geological Survey) in Mineral, Virginia, approximately 10 miles from NAPS. Shortly after the earthquake both units tripped, and there was a loss of offsite power to the station. Following the earthquake, both units were stabilized, taken to a safe shutdown condition, and offsite power was restored. Subsequent analysis indicated that the spectral and peak ground accelerations for the Operating Basis and Design Basis Earthquakes (OBE and DBE), respectively, for NAPS were exceeded at certain frequencies for a short period of time.

The NRC staff's assessment utilized the guidance provided in Regulatory Guide (RG) 1.167, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event," which endorses, with exceptions, the Electric Power Research Institute's (EPRI's) NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake" (Reference 26).

Following the earthquake, the NRC dispatched an Augmented Inspection Team (AIT) to NAPS to better understand the event and the licensee's response. The team's findings stated that no significant damage to the plant was identified and safety system functions were maintained. The NRC also sent a team of inspectors to the NAPS to provide an assessment of the licensee's inspection and testing program and the licensee's readiness for restart. The NRC staff also conducted an audit of the post-seismic fuel inspections at the NAPS in September 2011 (Reference 26). The purpose of the NAPS audit was (1) to discuss the scope of the post-seismic fuel inspection along with the procedures and criteria for judging the condition of the fuel assemblies, and (2) to witness the actual pool-side inspections to understand the capabilities of these inspections to identify fuel damage in support of NRC review of the licensee's restart submittal. A second audit to review Westinghouse and AREVA engineering calculations were also conducted to ensure operability of the fuel assemblies and control rods.

Based on the inspections and audit conducted, the NRC staff found that the performance of the AREVA and Westinghouse fuel assemblies during future normal operation and postulated accidents including during a similar seismic event remains acceptable (Reference 26) and therefore, no impact on the issuance of this amendment.

3.7 Technical Summary and Conclusion

The NRC staff has reviewed the LAR, in conjunction with the supplemental and additional information and reviewed supporting documents at the regulatory audits to evaluate the acceptability of the NAPS transition to Westinghouse RRFA-2 fuel with Dominion and Westinghouse safety analysis and core design methodologies. Based on its review, the NRC staff has determined that the licensee provided adequate technical basis to support the proposed TSs changes. Specifically, the NRC staff finds that the licensee has demonstrated that (1) Dominion complies with the staff limitations and conditions imposed for application of the topical reports, (2) Dominion-specific codes and methods are applicable for NAPS, and (3) the proposed TSs changes are acceptable, subject to the requirements described below.

The licensee is not requesting approval of the PAD4TCD model to support future cycles of NAPS. Instead the licensee is using this unapproved code to assess the impact of TCD on ECCS Performance, fuel mechanical design, and non-LOCA safety analyses. Currently approved Westinghouse methods will be maintained in the NAPS Technical Specifications. The audit (Reference 25) documents the staff's preliminary review of the licensee's assessment of this non-conformance. In accordance with 10 CFR 50.46(a)(3) reporting requirements, the licensee will submit a 30-day notification following startup of the reactor. This report is expected to provide further details and provide the opportunity for further interaction. The audit serves to provide a level of assurance that the units will startup and operate in accordance with Section 50.46(b)(1) to (5) criteria and that the TCD related error, while significant (i.e. >50°F), does not pose an immediate risk to public health and safety. A further level of assurance is provided by the fact that all of the Westinghouse fuel is fresh and therefore not immediately impacted by TCD (which builds with continued exposure).

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (76 FR 44618). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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23. WCAP-8745-P-A, "Design Bases for the Thermal Overpower DT and the Thermal Overtemperature DT Trip Function," Westinghouse, September 1986 (ADAMS Accession No. ML073521507).

24. WCAP-15386-P-A, Volumes I and II, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," April 2006.

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26. "Technical Evaluation by the Office of Nuclear Reactor Regulation Related to Plant Restart after the Occurrence of an Earthquake Exceeding the Level of the Operating Basis and Design Basis Earthquakes," US Nuclear Regulatory Commission, November 11, 2011 (ADAMS Accession No. ML061220455).

Principal Contributor: Mathew Panicker, NRR/DSS/SNPB

Date: February 29, 2012

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

Sincerely,

/ra/

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

Docket Nos. 50-338 and 50-339
Enclosures:
1. Amendment No. 266 to NPF-4
2. Amendment No. 247 to NPF-7
3. Safety Evaluation
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