April 4, 2006

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

SUBJECT: MILLSTONE POWER STATION, UNIT NOS. 2 AND 3 (MILLSTONE 2 AND 3), NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 (NORTH ANNA 1 AND 2), AND SURRY POWER STATION, UNIT NOS. 1 AND 2 (SURRY 1 AND 2) - APPROVAL OF DOMINION'S FLEET REPORT DOM-NAF-2, "REACTOR CORE THERMAL-HYDRAULICS USING THE VIPRE-D COMPUTER CODE" (TAC NOS. MC4571, MC4572, MC4573, MC4574, MC4575, AND MC4576)

Dear Mr. Christian:

By letter dated September 30, 2004, as supplemented by letters dated January 13, June 30, and September 8, 2005, Dominion Nuclear Connecticut, Inc., and Virginia Electric and Power Company (the licensees), requested approval for the generic application of Fleet Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code." The NRC staff has defined the term 'fleet report' as a report that can be used by the licensees' nuclear facilities.

In their submittal, the licensees stated that they are using the COBRA IIIc/MIT computer code to perform thermal hydraulic analyses. However, due to the need for enhanced core thermal-hydraulic capabilities, the licensees requested to use VIPRE-D to analyze multiple fuel types. The licensees developed VIPRE-D to fit the needs of the licensees nuclear plants and fuel products.

Although, the September 30, 2004, submittal identified the docket number for each of the licensees' plants, the Nuclear Regulatory Commission (NRC) staff was requested to approve of this fleet report on a generic basis. The licensees stated that plant-specific applications to implement this fleet report, including applicable appendixes, would be submitted to the NRC staff for review and approval under separate correspondence.

The enclosed Safety Evaluation (SE) documents the basis for the NRC staff's conclusion's that Fleet Report DOM-NAF-2, was found to be acceptable for the licensees' nuclear facilities. The SE defines the basis for acceptance of the report.

In accordance with the guidance provided on the NRC website, the NRC requests that the licensees publish an accepted version of this fleet report within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain, in appendices, historical review information, such as questions and accepted responses, and original report pages that were replaced. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

If the NRC's criteria or regulations change such that its conclusions as to the acceptability of the fleet report are invalidated, then the licensees will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

/RA/

Christopher I. Grimes, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket Nos. 50-336, 50-423, 50-338,
50-339, 50-280, and 50-281

Enclosure: Safety Evaluation

cc w/encl: See next page

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D. Christian

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO FLEET REPORT DOM-NAF-2

MILLSTONE POWER STATION, UNIT NOS. 2 AND 3

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-336, 50-423, 50-338, 50-339, 50-280, AND 50-281

1.0 INTRODUCTION

By letter dated September 30, 2004 (Reference 1), as supplemented by letters dated January 13 (Reference 2), June 30 (Reference 13), and September 8, 2005 (Reference 14), Dominion Nuclear Connecticut, Inc., and Virginia Electric and Power Company (the licensees), submitted a request for Nuclear Regulatory Commission (NRC) staff approval for the application of Fleet Report DOM-NAF-2, “Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code,” Appendix A, "Qualification of the Framatome Advanced Nuclear Power (F-ANP) BWU Critical Heat Flux (CHF) Correlations," and Appendix B "Qualification of the Westinghouse WRB-1 CHF Correlations in the Dominion VIPRE-D Computer Code." Appendix A includes the VIPRE-D code and correlation departure from nucleate boiling ratio (DNBR) design limits, and Appendix B provides an evaluation of DNBR for the Westinghouse WRB-1 CHF correlations that are applicable to the Westinghouse 15x15 optimized fuel assembly (OFA) fuel bundle.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations,* Part 50, Section 50.90, requires licensees to submit an application to the NRC whenever they desire to amend the license.

The VIPRE-01 computer code is a core thermal hydraulics computer program developed by the Electric Power Research Institute (EPRI) and approved generically by the NRC staff for the purpose of evaluating departure from nucleate boiling (DNB) for pressurized water reactor (PWR) systems. Since this generic approval did not include specific applications of VIPRE-01 to any particular fuel design, NRC staff review and approval is necessary in order to apply this methodology to a specific fuel design. Therefore, this review addresses the specific application of VIPRE-01 by the licensees to the Framatome and Westinghouse fuel types in the licensees' nuclear steam supply systems (NSSS).

VIPRE-D is the licensees' version of VIPRE-01, which has been enhanced by the addition of several vendor-specific CHF correlations. The licensees intend to utilize the VIPRE-D computer code to assess the DNBR for the Framatome BWU-N, BWU-Z, and BMU-ZM CHF fuel correlations. Additionally, the licensees intend to apply the VIPRE-D code to assess the Westinghouse WBR-1 CHF correlation for the 15x15 OFA fuel design. The licensees have previously used the COBRA IIIc/MIT computer code (Reference 3) to perform the thermal hydraulic analyses and is submitting this fleet report to replace COBRA IIIc/MIT computer code with the VIPRE-D computer program along with the new CHF correlations for the various Framatome and Westinghouse fuel designs. The NRC staff's technical evaluation of the VIPRE-D code and the new CHF fuel correlations is given below.

3.0 TECHNICAL EVALUATION

In order to evaluate DNB in the licensees' NSSS for the Framatome and Westinghouse fuel types, the NRC staff reviewed the application of the VIPRE-D code along with the various pertinent code correlations and models, fuel-specific CHF correlations, and DNBR design limits.

The VIPRE-D code is a modified version of the VIPRE-01 code which is a finite volume subchannel thermal hydraulics code with the specific capability to model a three-dimensional core and other component geometries. With the appropriate boundary conditions from a systems code such as RETRAN, VIPRE-01 computes the flow, void, pressure, and temperature distribution of the fluid through the core to ultimately compute the minimum DNB for steady state and transient conditions. The VIPRE-01 code also contains a fuel rod model that computes the radial and axial temperature distribution that is coupled to the cladding surface heat transfer coefficient correlations and CHF correlations that are particular to a given fuel rod and bundle design with the objective of determinating DNB following a non-loss-of-coolant accident (LOCA) transient event.

In order to compute the single and two-phase flow conditions that develop during transients undergoing a potential DNB, various two-phase flow models for handling subcooled and bulk boiling are available for use in the code, as well as convective heat transfer correlations for single and two-phase flow conditions. Correlations are also included in the code to deal with turbulent mixing, axial and cross-flow resistance, and form loss coefficients. As such, the NRC staff's review consisted of reviewing the CHF correlations and the various fluid flow and heat transfer options in the code to assure the correlations and models were validated over the range of conditions for those transients for which DNB is to be evaluated.

It is also noted that the licensees did not modify any of the phenomenological models or correlations in VIPRE-01. The licensees only added the new CHF correlations (Reference 1, Appendix A and Reference 2, Appendix B) to accommodate the DNBR assessments of the Framatome and Westinghouse fuel types. No other changes were made to VIPRE-01 in constructing the new VIPRE-D code.

3.1 Code Usage

The licensees indicated it plans to use the VIPRE-D code for the following applications.

(1) Perform an analysis of 14x14, 15x15, and 17x17 fuel in PWR reactors.

(2) Perform an analysis of DNBR for statistical and deterministic transients in the Updated Final Safety Analysis Report (UFSAR), as identified in Table 1, below. Additional DNBR transients that are plant specific may be analyzed in a plant-specific application that would be submitted to the NRC staff for review and approval.

(3) Perform steady state and transient DNB evaluations.

(4) Develop reactor core safety limits or core thermal limit lines (CTL).

(5) Provide the basis for reactor protection setpoints.

(6) Establish or verify the deterministic code/correlation DNBR design limits of the various DNB correlations in the code. Each one of these DNBR limits would be documented in an addendum or appendix to the original VIPRE-D document.

3.2 Code Applications

The licensees intend to implement Fleet Report DOM-NAF-2 (VIPRE-D) in its plant-specific applications through the following methods.

(1) Changes to the technical specifications (TSs) to add Fleet Report DOM-NAF-2 and Appendices A and B to the plant Core Operating Limit Report for that particular plant.

(2) Changes to the Statistical Design Limit(s) for the relevant code and correlation(s).

(3) Any TS changes related to over temperature delta T (OTΔT), over power delta T (OPΔT), enthalpy rise factor (FΔH) or other reactor protection function, as well as revised reactor core safety limits.

(4) Changes to the list of UFSAR transients for which the code and correlations apply, as shown in Table 1.

Table 1: UFSAR Transients Analyzed with VIPRE-D

1	Accidental depressurization of the main steam system
2	Accidental depressurization of the reactor cooling system
3	Excessive heat removal due to feedwater system malfunction
4	Excessive load increase
5	Inadvertent operation of emergency core cooling system during power operation
6	Locked reactor coolant pump rotor or shaft break
7	Loss of external electrical load and/or turbine trip
8	Loss of forced reactor coolant flow
9	Loss of normal feedwater
10	Major rupture of a main feedwater pipe
11	Rod cluster control assembly misalignment/dropped rod/bank
12	Rod cluster control assembly bank withdrawal at power
13	Rod cluster control assembly bank withdrawal from subcritical
14	Rupture of a main steam pipe
15	Single rod cluster control assembly withdrawal at full power
16	Startup of an inactive reactor coolant loop
17	Uncontrolled boron dilution

3.3 Compliance with the VIPRE-01 Safety Evaluation Report (SER)

In order to meet the NRC staff's requirements listed in the VIPRE-01 SER (References 4 and

5), the licensees will apply the VIPRE-D code for PWR licensing applications under the following conditions:

(1) The application of VIPRE-D is limited to PWR licensing calculations with heat transfer regimes up to CHF. VIPRE-D cannot be used for post-CHF calculations or for boiling-water-reactor calculations.

(2) VIPRE-D analyses will use only those DNB correlations reviewed and approved by the NRC staff in this SER. These correlations include the Framatome BWU-N, BWU-Z, and BMU-ZM CHF and the Westinghouse WRB-1 fuel CHF correlations.

(3) The Framatome BWU CHF correlations, which have been specifically developed for use with the Framatome Advanced Mark-BW fuel, were used in the 12-channel model. There are three BWU CHF correlations that constitute the licensing basis for the Framatome Advanced Mark-BW fuel assembly. These correlations use the same basic equation, but are fit to different databases (References 6 and 7). VIPRE-D applies different BWU correlations at different axial levels, according to the following guidelines:

> - <u>BWU-N</u>, which is only applicable in the presence of non-mixing vane grids (MVG), is used from the beginning of the heated length to the leading edge of the first structural MVG (Reference 6).
>
> - <u>BWU-Z</u>, which is the enhanced mixing vane correlation, is used from the leading edge of the first structural MVG to the leading edge of the second structural MVG (Reference 6).
>
> - <u>BWU-ZM</u>, which is just BWU-Z with a multiplicative enhancement factor and is applicable in the presence of mid-span mixing grids (MSMGs), is used from the leading edge of the second structural MVG to the leading edge of the last structural MVG (Reference 7).
>
> - <u>For the uppermost span</u>, in which the end of heated length occurs less than one grid span beyond the last MVG, the BWU-Z correlation is used with a grid spacing equal to the effective grid spacing (the distance from the last grid to the end of the heated length) (Reference 6).

(4) As required by the NRC staff in Reference 4, the following model options were reviewed and justified by the licensees for use in the DNB evaluation of the Framatome fuels.

> - Radial Nodalization: The licensees utilize 1/8th core symmetry and the model is applicable to the 14x14, 15x15, and 17x17 fuel arrays. These guidelines are consistent with the previously approved COBRA models (Reference 3). Benchmark calculations with the Framatome LYNXT code (References 8 and 9) verified this modeling approach.
>
> - Axial Nodalization: Node size is limited to a maximum of 6 inches.
>
> - Fuel Rod Model: The licensees will use the dummy fuel rod model which requires the

surface heat flux as input, computed by the RETRAN code. RETRAN accounts for the fuel conduction, gap conductance, and associated delayed energy transport effects. This approach is consistent with previously approved licensees' methodologies (Reference 10). Also, the analysis assumes that 97.4 percent of the reactor power is generated in the fuel while 2.6 percent is generated in the coolant, consistent with the previously approved COBRA modeling techniques.

- Power Distribution: A chopped cosine axial power shape is typically used. The power distribution is modeled to limit the cross flow and mixing in the hot channel since the peak F ΔH is also applied to the thimble and hot cell. This results in a conservative calculation of DNBR. Also since the data is limited with respect to top peaked axial profiles, the licensees utilize the Tong F-factor to correct for non-uniform axial power shapes, which has been previously approved by the NRC staff. The licensees also performed benchmark comparisons between VIPRE-D/BWU and LYNXT/BWU and VIPRE-D/WRB-1 with COBRA/WRB-1 using symmetric and non-symmetric axial power shapes that show no dependency on the shape of the power distribution.

- Turbulent Mixing: The turbulent mixing factor is 0.0 as opposed to the VIPRE Manual recommended value of 0.8. This produces a conservative calculation since momentum mixing is precluded with this assumption. The turbulent mixing for single-phase fluid in single channels is set to 0.038 (range 0.0 to 0.1). This is the default model approved in the original generic VIPRE SER. For flow paths connected to lumped channels, turbulent mixing is set to zero for conservatism.

- Axial Hydraulic Losses and Cross-Flow Resistance: For axial cross flow, the McAdams correlation is used to approximate the Colebrook smooth pipe formulation for single-phase axial friction. Lateral resistance is computed by the Idle Chik empirical correlation (Reference 10) for bundle circular tubes in a vertical column.

- Form Loss Coefficients: These are obtained from the vendor for the particular fuel bundle designs. VIPRE-D properly places the losses at the top of the cell, or at the boundaries between the cells where the grids are located. Varying the location of the grid resistance upward or downward showed an insignificant change in DNBR (much less than the 5 percent uncertainty associated with thermal-hydraulic codes in this application).

- Two-Phase Flow and Heat Transfer Correlations: The licensees will use the following models to compute CHF for the specific fuel types: EPRI Subcooled Void Model, EPRI Bulk Boiling Void Model, and the EPRI Two-Phase Friction Multiplier. No hot wall friction correlation is used. Results of the comparisons of VIPRE-D with LYNXT justify this choice of correlations and models since this combination produced the lowest standard deviation in DNBR with a value of 0.89 percent. The slip model is not to be employed and cannot be used. The Dittus-Boelter single-phase heat-transfer correlation is also used.

-Engineering Factors: The licensees include the following factors which adversely affect DNBR: Local Heat Flux Hot Channel Factor, Engineering Enthalpy-Rise Hot Channel Factor, Stack Height Reduction, and Inlet Flow Reduction. These factors are fuel product dependent.

- CHF Correlations: See the "Correlations and DNBR Limits" Section 3.5 below.

- For transient analysis, appropriate time steps are selected to ensure numerical stability and accuracy. The Courant number, which is based on flow velocity, time step and axial node size, is set to be greater than one in VIPRE-D transient calculations whenever a subcooled void model is used.

3.4 Benchmarks

VIPRE-D benchmark calculations were performed with the Framatome LYNXT code and the 12-channel model created by Framatome to model the North Anna Power Station, Unit Nos. 1 and 2, cores containing Framatome Advanced Mark-BW fuel assemblies. This benchmark uses 173 state points obtained from the UFSAR Chapter 15 events including the reactor core safety limits, axial offset envelopes (AO's), rod withdrawal at power (RWAP), rod withdrawal from subcritical (RWSC), control rod misalignment, loss of flow accident (LOFA), and locked rotor accident (LOCROT) events to compare the performance of VIPRE-D and LYNXT. These various limits and events provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power FΔH), pressure and temperature (reactor core safety limits); (b) AOs; (c) elevated hot rod power (misaligned rod); and (d) LOFA and LOCROT. The 173 state points cover the full range of conditions and axial offsets in the North Anna UFSAR Chapter 15 evaluations except for main steamline break (MSLB), which is discussed in Section 5.2 of Reference 1. These results were specifically selected to challenge the three BWU CHF correlations.

This benchmark study showed an average deviation between VIPRE-D and LYNXT of less than 0.14 percent in DNBR, with a maximum deviation of 2.2 percent. These results are well within the uncertainty typically associated with thermal-hydraulic codes, which has been quantified to be 5 percent (References 9 and 12), and these results justify the model selections in Section 4 of Reference 1. The close comparison of VIPRE-D to LYNXT over the full range of conditions expected for UFSAR transients justifies the applications of VIPRE-D to the transients identified in Table 1, above. The range of conditions for the benchmarks is given below in Table 2.

Table 2: Range of VIPRE-D / LYNXT 173 Benchmark State points

VARIABLE	RANGE
Pressure [psia]	1860 to 2400
Power [percent of 2942.2 MWt]	66 to 135
Inlet Temperature [°F]	506.6 to 626.2
Flow [percent of Minimum Measured Flow]	64 to 100
FΔH	1.49 to 1.945
Axial Offset [percent]	-48.7 to 57.9

The 12-channel model discussed in Section 5.1 of Reference 1 does not allow the modeling of the peaking and inlet boundary conditions in the fuel assemblies adjacent to the hot assembly, which is necessary for the analysis of some accidents, such as MSLB. Consequently, a

14-channel model was created to more accurately simulate the behavior of the core during an MSLB event.

The VIPRE-D 14-channel model for a North Anna core containing Framatome Advanced Mark-BW fuel assemblies consists of 14 channels (10 subchannels and 4 lumped channels) and 16 rods. The two additional channels provide adequate detail of the flow field in the vicinity of the hot assembly and allow for the modeling of the peaking and inlet boundary conditions in the fuel assemblies adjacent to the hot assembly.

In order to verify the accuracy, the licensees compared the results from the VIPRE-D 14-channel model to the results from the Framatome LYNXT model for high flow (with offsite power) and low flow (without offsite power) MSLB evaluations. The results obtained showed a maximum deviation of 2.12 percent in DNBR. These results demonstrated that VIPRE-D provides results similar to those of other approved codes accepted for analysis of an MSLB event, provided the model has sufficient detail surrounding the hot assembly, such as the 14-channel model described in Reference 1.

In addition, the results of the 14-channel model comparison with the DNBR results of the 173 state points obtained with the VIPRE-D 12-channel model showed that there was essentially no difference between the 12-channel and the 14-channel models (the average deviation in DNBR was 0.03 percent), which indicates that VIPRE-D models were created following the methodology discussed in Section 4 of Reference 1 and are acceptable.

3.5 Correlations and DNBR Limits

The BWU-Z, BWU-ZM and BWU-N correlations have been qualified with the licensees' VIPRE-D computer code. Table 3 summarizes the DNBR design limits for VIPRE-D/BWU-Z, VIPRE-D/BWU-ZM and VIPRE-D/BWU-N that yield a 95 percent non-DNB probability at a 95 percent confidence level. Table 3 summarizes the applicability and the ranges of validity for all three CHF correlations.

Table 3: VIPRE-D DNBR Limits for BWU-Z, BWU-ZM and BWU-N

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

These correlations are to be used over the following thermal hydraulic conditions:

Table 4: Range of validity for BWU-Z, BWU-ZM and BWU-N

	BWU-Z	**BWU-ZM**	**BWU-N**
Pressure [psia]	400 to 2,465	400 to 2,465	788 to 2,616
Mass Velocity [Mlbm/hr-ft^2]	0.36 to 3.55	0.47 to 3.55	0.25 to 3.83
Thermodynamic Quality at CHF	Less than 0.74	Less than 0.68	Less than 0.70
Applicability	Mixing Vane Grids	Mid-Span Mixing Grids	Non-Mixing Vane Grids

The WRB-1 correlation is applicable to the Westinghouse 15x15 OFA fuel assemblies at Surry Power Station, Unit Nos. 1 and 2. The DNBR limit was found to be 1.17 and was the same as the limits computed using the previously approved methodologies of the licensees (COBRA of Reference 11) and Westinghouse (THINC and VIPRE-01). The range of applicability of the WRB-1 correlation is summarized below in Table 5.

Table 5: Range of VIPRE-D / WRB-1 Benchmark State points

VARIABLE	**RANGE**
Pressure [psia]	1440 to 2490
Mass Velocity [Mlbm/hr-ft^2]	0.9 to 3.7
Thermodynamic Quality at CHF	#0.30
Local Heat Flux [Mbtu/hr-ft^2]	#1.00
Mixing Vane Grid [in]	> 13.0

By letter dated January 13, 2005, the licensees imposed the following additional restrictions on the use of the VIPRE-D/WRB-1 correlation.

(1) VIPRE-D/WRB-1 will not be used when the local heat flux exceeds 1.0 Mbtu/hr-ft^2, and

(2) VIPRE-D/WRB-1 will not be used for fuel with less than a 13-inch mixing vane grid spacing.

The licensees imposed these restrictions as a result of the constraints the NRC staff placed on the use of Reference 11, in its letter dated July 25, 1989.

The previously approved W-3 correlation will be used when conditions fall outside the range of the WRB-1 correlation. Specifically, the W-3 correlation will be applied to the lower portion of the fuel assemblies in the RWSC event because of the bottom peaked axial power profile assumed and the MSLB event because of the low pressures encountered. The W-3 will use a limit of 1.3 for the rod withdrawl event. For the MSLB, the limit of 1.45 will be used for pressures 500 to 100 psia and the limit of 1.3 will be used for pressures above 1000 psia. Benchmarking of the VIPRE-D code with the results of the COBRA code for the events listed in Table 1 above (except the MSLB event) showed an average deviation of less than 0.6 percent in DNBR with a maximum deviation of 3.75 percent. This is within the uncertainty for thermal hydraulic codes used to perform analyses of this nature. For the MSLB, the comparison with COBRA using the W-3 correlation, showed the maximum deviation was 1.5 percent.

The licensees utilized a One-Sided Tolerance theory for the VIPRE-D fuel correlation DNBR design limits given above. This theory allows the licensees to calculate a DNBR limit such that values equal to the design limit avoids DNB with a 95-percent probability at a 95-percent confidence level. All of the statistical techniques utilized in the design limit determinations assumed that the original data distribution is normal. As such the licensees verified that the overall measured-to-predicted CHF ratios were also normally distributed evaluated through the use of a "D" normality test.

Following the review of References 1 and 2, Requests for Additional Information (RAIs) were sent to the licensees requesting supplemental information regarding the review of the VIRPE-D code model options and usage, the statistical evaluation of the DNBR design limits specific to each fuel type, and the benchmarking evaluations. The RAI responses are documented in Reference 13 and the staff found these responses to be acceptable.

Lastly, an error was uncovered by Framatome in their LYNXT computer code, the results of which, were used by the licensees to qualify portions of the licensees' VIPRE-D code. The licensees' assessment of the impact of the error, reported to the NRC staff in Reference 14, shows that the error does not affect the LYNXT/BWU code or correlation limits. Furthermore, the maximum change in any numerical value reported in Reference 1, Section 5, regarding benchmark DNBR calculations between LYNXT and VIPRE-D, was found to be 0.02 percent. Appendix B of Reference 2 is not affected by this error. The NRC staff agrees that the impact of the error has a negligible effect on the calculated differences between the VIPRE-D and LYNXT DNBR benchmarking calculations.

4.0 CONCLUSION

The NRC staff finds the proposed use of the VIPRE-D code to evaluate DNBR for selected PWR transients is acceptable. Furthermore, the NRC staff finds the modifications to VIPRE-D to evaluate the Framatome BWU fuel using the BWU-Z, BWU-ZM, and BWU-N CHF correlations as well as the Westinghouse 15x15 OFA fuel using the WRB-1 correlation to also be acceptable. The VIPRE-D fuel design limits are also found to be acceptable by the NRC staff for the Framatome and Westinghouse fuel types listed herein. The use of the licensees' VIPRE-D code is limited to only these CHF correlations. The VIPRE-D code can be used subject to the models and options specified in DOM-NAF-2, Rev. 0, Sections 4.0 through and including, Section 4.12 (Reference 1). Evaluation of the Framatome fuel using the BWU-Z, BWU-ZM, and BWU-N CHF correlations is subject to the DNBR limits and ranges given in Section A.5 of DOM-NAF-2, Rev. 0 (Reference 1). Use of the VIPRE-D code is also approved for evaluating the Westinghouse 15x15 OFA fuel using the WRB-1 CHF correlation subject to the DNBR limits and evaluation ranges given in Tables B.8-1 and B.8-2 of DOM-NAF-2, Rev. 0.0 Appendix B (Reference 2). The WRB-1 correlation is limited by the following restrictions: (1) VIPRE-D/WRB-1 will not be used when the local heat flux exceeds 1.0 MBTU/hr-ft^2, and (2) VIPRE-D/WRB-1 will not be used for fuel with less than a 13-inch mixing vane grid spacing, as discussed in Reference 2 Section B.3. The W-3 correlation will also be used when the conditions fall outside the range of the WRB-1 correlation as discussed in Section B.3, last paragraph of Reference 2. The VIPRE-D code is further restricted for application to those transients listed in Table 2.1-1 of DOM-NAF-2, Rev. 0.0 (Reference 1) and the uses and applications listed in Section 2.1 entitled "VIPRE-D Application."

5.0 REFERENCES

1. Letter from Leslie N. Hartz (Dominion Nuclear Connecticut, Inc.) to the USNRC, "Virginia Electric and Power Company (Dominion), Dominion Nuclear Connecticut, Inc. (DNC), North Anna and Surry Power Stations Units 1 and 2, Millstone Power Station Units 2 and 3, Request for Approval of Topical Report DOM-NAF-2 Reactor Core Thermal-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," dated September 30, 2004.

2. Letter from Eugene S. Grecheck (Dominion Nuclear Connecticut, Inc.) to the USNRC, "Virginia Electric and Power Company (Dominion), Dominion Nuclear Connecticut, Inc. (DNC), North Anna and Surry Power Stations Units 1 and 2, Millstone Power Station Units 2 and 3, Request for Approval of Appendix B of Topical Report DOM-NAF-2 Qualification of the Westinghouse WRB-1 CHF Correlations in the Dominion VIPRE-D Computer Code," dated January 13, 2005.

3. Topical Report, VEP-FRD-33-A, "VEPCO Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIc/MIT Computer Code," F. W. Sliz and K. L. Basehore, October 1983.

4. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," Volumes 1, 2, 3 and 4, dated May 1, 1986.

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Principal Contributor: L. Ward

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Virginia Electric and Power Company
cc:

Ms. Lillian M. Cuoco, Esq.
Senior Counsel
Dominion Resources Services, Inc.
Building 475, 5th Floor
Rope Ferry Road
Waterford, Connecticut 06385

Mr. Donald E. Jernigan
Site Vice President
Surry Power Station
Virginia Electric and Power Company
5570 Hog Island Road
Surry, Virginia 23883-0315

Senior Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
5850 Hog Island Road
Surry, Virginia 23883

Chairman
Board of Supervisors of Surry County
Surry County Courthouse
Surry, Virginia 23683

Dr. W. T. Lough
Virginia State Corporation Commission
Division of Energy Regulation
Post Office Box 1197
Richmond, Virginia 23218

Dr. Robert B. Stroube, MD, MPH
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
Post Office Box 2448
Richmond, Virginia 23218

Office of the Attorney General
Commonwealth of Virginia
900 East Main Street
Richmond, Virginia 23219

Mr. Chris L. Funderburk, Director
Nuclear Licensing & Operations Support
Innsbrook Technical Center
Dominion Resources Services, Inc.
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Mr. Jack M. Davis
Site Vice President
North Anna Power Station
Virginia Electric and Power Company
Post Office Box 402
Mineral, Virginia 23117-0402

Mr. C. Lee Lintecum
County Administrator
Louisa County
Post Office Box 160
Louisa, Virginia 23093

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
1024 Haley Drive
Mineral, Virginia 23117