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United States Nuclear Regulatory Commission
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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

VIPRE is a core thermal-hydraulics computer code developed by EPRI and approved by the NRC, which is currently in use throughout the nuclear industry. VIPRE-D is the Dominion version of VIPRE, which has been enhanced by the addition of several vendor specific CHF correlations. Dominion has validated VIPRE-D with extensive code benchmark calculations, and the accuracy of VIPRE-D has been demonstrated through comparisons with other NRC-approved methodologies.

In an April 15, 2004 letter (Serial No. 04-196), Dominion submitted for NRC review and approval Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," and Appendix A to the Topical Report DOM-NAF-2, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code." In a July 9, 2004 letter, the NRC requested a superseding submittal addressing completeness of the supporting materials.

A public meeting was held between the NRC and Dominion on August 4, 2004 to discuss the proposed resubmission of Topical Report DOM-NAF-2. Dominion is resubmitting Topical Report DOM-NAF-2, incorporating the information identified in the September 3, 2004 letter "Summary of August 4, 2004, Meeting with VEPCO on Topical Report DOM-NAF-2". In summary, the attached Topical Report DOM-NAF-2 has been revised as follows:

1. A discussion of the differences between VIPRE-01, MOD.2 and VIPRE-01, MOD.2.1, and how the NRC approval for VIPRE-01, MOD.2 is fully applicable to VIPRE-01, MOD.2.1 is included in Section 3.1 of the Topical Report. In addition, Section 3.2 discusses the differences between VIPRE-D and VIPRE-01, and how the NRC approval of VIPRE-01 is fully applicable to VIPRE-D.

2. Section 4 of DOM-NAF-2 discusses and justifies all the modeling assumptions and parameters of interest to be used in the creation of VIPRE-D models.
3. Section 2.1 of DOM-NAF-2 includes a list of the UFSAR transients for which Dominion plans to use the VIPRE-D code for DNBR analysis.
4. Sections 5.1 and 5.2 of DOM-NAF-2 demonstrate that the VIPRE-D models, which were created using the selections and modeling guidelines described in Section 4, provide close comparison with other NRC approved subchannel codes. In particular, VIPRE-D and Framatome ANP's LYNXT results for 173 state points obtained from the UFSAR are compared in these sections.
5. Section 5.3 of DOM-NAF-2 includes specific results and boundary conditions for two sample transients and the limiting steady state statepoints selected for each transient.

There are no changes to Appendix A from the previous submittal.

Thus, Dominion is re-submitting Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," provided in Attachment 1 for NRC review and approval. In addition, Dominion is resubmitting Appendix A to the Topical Report DOM-NAF-2, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," provided in Attachment 2 for NRC review and approval. Appendix A documents the qualification of the Framatome ANP BWU CHF correlations (BWU-Z, BWU-ZM and BWU-N) with the VIPRE-D code, and the code/correlation DNBR design limits. Please note that future references to the Topical Report DOM-NAF-2 are considered to consist of the main body of the report and any approved appendixes.

Although the docket number is identified for each Dominion/DNC unit, Dominion is requesting the approval of the generic application of this topical report. Plant specific applications of this topical report, including applicable appendixes, will be submitted to the NRC for review and approval, in accordance with Section 2.1 of DOM-NAF-2. In fact, Dominion plans to reference this Topical Report, including Appendix A, in a License Amendment Request that will be submitted in the first quarter of 2005.

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

Very truly yours,

Leslie N. Hartz
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Attachments

Commitments made in this letter: None

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Attachment 1

Topical Report DOM-NAF-2

REACTOR CORE THERMAL-HYDRAULICS USING THE VIPRE-D COMPUTER CODE

Virginia Electric and Power Company (Dominion)
Dominion Nuclear Connecticut (DNC)

DOM-NAF-2, Rev. 0.0

Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code

NUCLEAR ANALYSIS AND FUEL DEPARTMENT
DOMINION
RICHMOND, VIRGINIA
September, 2004

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CLASSIFICATION/DISCLAIMER

ABSTRACT

As part of a continuing effort to improve core thermal-hydraulics methods, Dominion (Virginia Electric and Power Company) is updating its capability for performing nuclear reactor analyses in support of its nuclear power stations. VIPRE is a core thermal-hydraulics computer code currently in wide use throughout the nuclear industry. VIPRE-D is the Dominion version of VIPRE, which has been enhanced by the addition of several vendor specific CHF correlations. Dominion has validated VIPRE-D with extensive code benchmark calculations, and the accuracy of VIPRE-D has been demonstrated through comparisons with other NRC-approved methodologies. VIPRE-D has been shown to meet or exceed the same standards for accuracy as methodologies currently being used by Dominion.

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ACRONYMS AND ABBREVIATIONS

AMBW	Advanced Mark-BW
AO	Axial Offset
BWR	Boiling Water Reactor
CHF	Critical Heat Flux
CTL	Core Thermal Limit
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EPRI	Electric Power Research Institute
F-ANP	Framatome Advanced Nuclear Power
FLC	Form Loss Coefficients
FTM	Turbulent Momentum Factor
LOCROT	Locked Rotor Accident
LOFA	Loss Of Flow Accident
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MSLB	Main Steam Line Break
MSMG	Mid-Span Mixing Grid
MVG	Mixing Vane Grid
NMVG	Non-Mixing Vane Grid
NAPS	North Anna Power Station
PWR	Pressurized Water Reactor
RWAP	Rod Withdrawal At Power
RWSC	Rod Withdrawal from Subcritical
SER	Safety Evaluation Report
UFSAR	Updated Final Safety Analysis Report
USNRC	US Nuclear Regulatory Commission
VIPRE	Versatile Internals and Components Programs for Reactors - EPRI

1.0 INTRODUCTION

The basic objective of core thermal-hydraulic analysis is the accurate calculation of reactor coolant conditions to verify that the fuel assemblies constituting the reactor core can safely meet the limitations imposed by departure from nucleate boiling (DNB) considerations. DNB, which could occur on the heating surface of the fuel rod, is characterized by a sudden decrease in the heat transfer coefficient with a corresponding increase in the surface temperature. DNB is a concern in reactor design because of the possibility of fuel rod failure resulting from the increased rod surface temperature.

In order to preclude potential DNB related fuel damage, a design basis is established and is expressed in terms of a minimum departure from nucleate boiling ratio (MDNBR). The departure from nucleate boiling ratio (DNBR) is the ratio of the predicted heat flux at which DNB occurs (i.e. the critical heat flux, CHF) and the local heat flux of the fuel rod. By imposing a DNBR design limit, adequate heat transfer between the fuel cladding and the reactor coolant is assured. DNBRs greater than the design limit indicate the existence of thermal margin within the reactor core. Thus, the purpose of core thermal-hydraulic DNB analysis is the accurate calculation of DNBR in order to assess and quantify core thermal margin.

Dominion (Virginia Power) has used the COBRA IIIc/MIT computer code (Reference 8) to perform the thermal-hydraulic analyses discussed above. COBRA is licensed to evaluate the thermal margin for North Anna Power Station (NAPS) and Surry Power Station cores containing Westinghouse fuel. However, Dominion's nuclear assets and fuel products require enhanced core thermal-hydraulic capabilities. As a consequence, Dominion has decided to implement a new thermal-hydraulic analysis computer program to analyze multiple fuel types.

VIPRE-D is the Dominion version of the computer code VIPRE (Versatile Internals and Components Program for Reactors - EPRI), developed for EPRI (Electric Power Research Institute) by Battelle Pacific Northwest Laboratories in order to perform detailed thermal-hydraulic analyses to predict CHF and DNBR of reactor cores (References 1 through 5). VIPRE-01 has been approved by the U.S. Nuclear Regulatory Commission (USNRC) (References 6 and 7). VIPRE-D, which is based upon VIPRE-01, MOD-02.1, was developed by Dominion to fit the specific needs of Dominion's nuclear plants and fuel products by adding vendor specific CHF correlations and customizing its input and output. Dominion, however, has not made any modifications to the NRC-approved constitutive models and algorithms in VIPRE-01.

This report describes Dominion's use of the VIPRE-D code, including modeling and qualification for Pressurized Water Reactors (PWR) thermal-hydraulic design. This report demonstrates that the VIPRE-D methodology is appropriate for PWR licensing applications.

This report is organized into six sections. Section 2 provides a description of VIPRE-D methodology and intended applications, including a discussion on VIPRE-D compliance with the VIPRE-01 Safety Evaluation Report (SER). Section 3 describes the VIPRE-D code and its capabilities. Section 4 describes the VIPRE-D modeling of PWR cores and fuel rods. Section 5 provides VIPRE-D benchmark calculations against other subchannel codes for PWR DNB analyses, such as Framatome ANP (F-ANP) LYNXT (Reference 14). Conclusions and references are presented in succeeding sections. The topical allows for a series of appendixes, each one containing the verification and qualification of additional CHF correlations with the VIPRE-D code.

2.0 TOPICAL METHODOLOGY

2.1 VIPRE-D APPLICATION

The intended VIPRE-D applications are consistent with the Dominion COBRA applications for PWRs using USNRC approved methodologies (Reference 8). The VIPRE-D applications include DNB analyses to define PWR core safety limits that provide the basis for reactor protection setpoints, and to perform DNBR calculations in reactor transients. While VIPRE-D is able to model Boiling Water Reactors (BWR), its BWR features and capabilities are not discussed for qualification in this report. Furthermore, the rod conduction model present in VIPRE-D will not be used. All VIPRE-D models will employ the dummy rod model.

Dominion plans to use the VIPRE-D code for:

1) Analysis of 14x14, 15x15 and 17x17 fuel in PWR reactors.

2) Analysis of DNBR for statistical and deterministic transients in the Updated Final Safety Analysis Report (UFSAR), as identified in Table 2.1-1. Additional DNBR transients that are plant specific may be analyzed in a plant specific application that would be submitted to the USNRC for review and approval.

3) Steady state and transient DNB evaluations.

4) Development of reactor core safety limits (also known as core thermal limit lines, CTL).

5) Providing the basis for reactor protection setpoints.

6) Establishing or verifying 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 appendix to this document.

Plant specific applications of VIPRE-D would include:

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

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

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

4) List of UFSAR transients for which the code/correlations apply (see Table 2.1-1).

Table 2.1-1: 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

2.2 COMPLIANCE WITH VIPRE-01 SER

In order to meet the USNRC's requirements listed in the VIPRE-01 SER (References 6 and 7), Dominion 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 regime up to CHF. VIPRE-D will not be used for post-CHF calculations or for BWR calculations.

2) VIPRE-D analyses will only use DNB correlations that have been reviewed and approved by the USNRC. The VIPRE-D DNBR calculations will be within the USNRC approved parameter ranges of the DNB correlations, including fuel assembly geometry and grid spacers. The correlation DNBR design limits will be derived or verified using fluid conditions predicted by the VIPRE-D code. Each DNB correlation will be qualified or verified in appendixes to this report.

3) This report provides the necessary documentation to describe the intended uses of VIPRE-D for PWR licensing applications. The report provides justification for Dominion's specific modeling assumptions, including the choice of two-phase flow models and correlations, heat transfer correlations and turbulent mixing models. Dominion only applies models and correlations already existing in VIPRE-01 and previously approved by the USNRC (Section 4).

4) 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.

5) VIPRE-D is maintained within Dominion's 10CFR50, Appendix B Quality Assurance program.

3.0 CODE DESCRIPTION

3.1 VIPRE-01

VIPRE is a computer code developed for EPRI by Battelle Pacific Northwest Laboratories in order to perform detailed thermal-hydraulic analyses of reactor cores (References 1 through 5). VIPRE-01, MOD-02 was previously approved by the USNRC (References 6 and 7). The code errors reported and verified since the release of VIPRE-01, MOD-02, as well as some documentation changes and other minor enhancements, were incorporated into version VIPRE-01, MOD-02.1 of the code, which was released in May 2001. These changes did not alter the basic models, equations and algorithms in the code, and it was verified that all significant differences between the results of the VIPRE Standard Testcases for MOD-02.1 and MOD-02 were accounted for and were the result of error corrections.

VIPRE-01 uses the subchannel analysis concept where a reactor core is divided into a number of flow channels that communicate laterally by crossflow and turbulent mixing. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, crossflow, and momentum pressure drop. A detailed description of the VIPRE-01 subchannel equations can be found in Reference 1. The VIPRE-01 flow field is assumed to be incompressible and homogeneous. It is assumed that any lateral flow is directed by the gap through which it flows, and it loses its sense of direction after leaving the region. Since crossflow is assumed to exist only between two adjacent channels, no external lateral boundary conditions are required.

The VIPRE-01 heat transfer model is capable of solving the conduction equation for the temperature distribution within the fuel rods and provides the heat source term for the fluid energy equation. The full boiling curve can be incorporated into the heat transfer model, from single-phase convection through nucleate boiling to the DNB point, and from transition boiling to the film boiling regime. A detailed description of the VIPRE-01 heat transfer model can be found in Reference 1.

VIPRE-01 offers two numerical solution options: the upflow solution, which is similar to the one in COBRA-IIIC; and the recirculation solution scheme adapted from COBRA-WC. Both solution schemes iteratively solve the same finite difference equations and use the same model and correlations for heat transfer, wall friction, fluid state and two-phase flow. The difference between them is in the numerical method used to obtain the flow and pressure fields. Both solution schemes yield essentially the same results (Reference 4, Section 7.3). However, the recirculation solution scheme is applicable to core conditions having flow reversal and recirculation. Either solution scheme can be used for PWR analysis.

VIPRE-01 modeling of a PWR core is based on the one-pass modeling approach (Reference 1), in which hot channels (subchannels with the highest enthalpy rise) and their adjacent region are modeled in detail, while the remainder of the core is modeled simultaneously on a relatively coarse mesh. A reactor core can be modeled in a small number of channels while still maintaining sufficient detail and accuracy around the hot channels. A one-pass model contains lumped channels that comprise total flow area and heated and wetted perimeters of the individual subchannels. The lumped channel gives uniform conditions over the entire flow area of the channel. Some input parameters of the lateral momentum equation in the VIPRE-01 code are adjusted in order to obtain the correct crossflow for the lumped channel. The VIPRE-01 one-pass modeling has been approved by the USNRC (References 6 and 7).

3.2 VIPRE-D

VIPRE-D, which is based upon VIPRE-01, MOD-02.1, was developed by Dominion to fit the specific needs of Dominion's nuclear plants and fuel products by adding vendor specific CHF correlations and customizing its input and output. Dominion, however, has not made any modifications to the NRC-approved constitutive models and algorithms in VIPRE-01 and the computational philosophy of VIPRE-D remains unchanged from VIPRE-01. Therefore, the VIPRE-01 qualification is fully applicable to VIPRE-D.

In addition to minor formatting changes and corrections to reported code errors, the main enhancement made to VIPRE-01, MOD-02.1 to obtain VIPRE-D is the addition of several vendor proprietary CHF correlations. Additional customizations were made in VIPRE-D's input and output to integrate it seamlessly into Dominion's thermal hydraulic methodologies. Additional CHF correlations may be added to the code in the future. Each one of these DNB correlations will be qualified or verified in its own appendix to this report, and submitted to USNRC for review and approval, prior to licensing use.

The VIPRE-D coding changes do not alter the fundamental computational method and solution scheme of the VIPRE-01 code. It has been demonstrated by running the VIPRE Standard Testcases that the additions and modifications made to create VIPRE-D have been correctly implemented into the code and have not affected in any way the original internal models and algorithms in the code. VIPRE-D has been developed and is maintained in accordance with Dominion's 10CFR50 Appendix B Quality Assurance program. The VIPRE-01 User's Manual (Reference 2) is fully applicable to VIPRE-D, but it has been augmented with an in-house User's Manual that clarifies the selection of VIPRE-D specific CHF correlations and enhancements.

4.0 VIPRE-D MODELING

The goal of Section 4 of this report is to comply with Condition 3 of the VIPRE-01 SER (References 6 and 7), which requires that each organization using VIPRE-01 provide justification for specific modeling assumptions, including the choice of two-phase flow models and correlations, heat transfer correlations and turbulent mixing models. As such, the methodology and guidelines used to create the VIPRE-D model for a typical Dominion reference plant core are described in this section.

As discussed in Section 3 of this report, no substantive modifications have been made to VIPRE-D. All the models selected and discussed herein were previously approved in VIPRE-01. The modeling choices described below (Table 4.0-1), which are not plant-specific, were developed in a manner consistent with the USNRC approved Dominion COBRA models (Reference 8) and with standard industry practice. These modeling choices will be used for the qualification or verification of all the CHF correlations included in the Appendixes to this report, unless otherwise specified in the particular Appendix.

Sections 4.1 (radial nodalization), 4.2 (axial nodalization), 4.4 (power distribution), 4.7 (form loss coefficients), 4.9 (CHF correlations), 4.10 (engineering factors) and 4.11 (boundary conditions) describe modeling areas that are fuel or accident dependent and would have to be determined based on the particular core and the type of analysis to be performed. The remaining sections, listed in Table 4.0-1, describe modeling choices that are independent of the fuel type.

Section 5.0 of this report describes a specific example applying these guidelines to a North Anna Power Station core containing F-ANP Advanced Mark-BW (AMBW) fuel assemblies. Extensive code benchmark calculations have confirmed that the VIPRE-D models specified in sections 4.1 through 4.12 in this report produce essentially the same results as equivalent F-ANP LYNXT models (Reference 13).

Table 4.0-1: VIPRE-D Modeling Summary

VIPRE-01 MODEL	DOMINION SELECTION	SECTION
Fuel Rod Modeling	"Dummy" rod model	Section 4.3
Turbulent Mixing	No momentum mixing ABETA fuel dependent	Section 4.5
Axial Friction Losses	McAdams Correlation	Section 4.6
Crossflow Resistance	Idel'Chik Correlation	Section 4.6
Two Phase Flow	EPRI Correlations	Sections 4.8 & 5.4
Heat Transfer	Dittus-Boelter Correlation	Section 4.8
Run Control Parameters	Default Options with Courant > 1 for transients	Section 4.12

4.1 RADIAL NODALIZATION

While the techniques used in formulating the hydraulic representation of a typical core are applicable in general to all PWRs, the specifics of the model change with the type of fuel present in the particular core and the type of analysis being performed. In general it is assumed that the core presents 1/8th symmetry, and thus it is only necessary to model 1/8th of the core. It is also assumed that the hot assembly is located at the center of the core, and therefore, the 1/8th core model will contain 1/8th of the hot assembly. The adequate number of channels to model a given core must allow simulating the entire core, while having a detailed subchannel model surrounding the hot channels. A set of subchannels surrounding the hot channels (i.e., hot thimble cell and hot typical cell) is sufficient to provide adequate solution detail of the flow field in the vicinity of the hot subchannels (Reference 2). If the model is used for the analysis of main steam line break (MSLB) events, it is also necessary to account for the core inlet enthalpy maldistribution when defining the number of channels. This modeling methodology is applicable to 14x14, 15x15 and 17x17 PWR fuel.

These modeling guidelines are consistent with the USNRC approved Dominion COBRA models (Reference 8) and with standard industry practice. The adequacy of using a one-eighth core model and the above modeling guidelines has been verified through benchmark calculations with the F-ANP LYNXT code (References 13 and 14), and will be discussed in Section 5.0.

4.2 AXIAL NODALIZATION

The finite differences methods used in VIPRE-D require that sufficient axial nodes be provided to resolve the details of the flow field and the axial power profiles. Dominion models use an axial nodalization scheme that places all the mixing and non-mixing vane grids at the upper edges of the axial nodes for better numerical convergence, while preserving the actual grid spacing. This is important because VIPRE-D applies the pressure loss associated with a node at the top edge of the node. Therefore, it is important to create a nodal distribution that ensures that the axial locations where the pressure losses are applied match the actual axial locations for each spacer grid.

VIPRE-D allows a PWR core to be modeled with variable axial nodal length. VIPRE-D offers a great deal of control and flexibility by allowing the user to define both the geometry and the axial power shapes with as much detail as needed in the critical areas of the model and with not so much detail in less critical areas. Dominion models use typical node lengths of 2 inches. A maximum node length of 6 inches will be used in the models. Selection of a very small node length is not reasonable since an excessive number of nodes will add significantly to the run time of the problem and the memory required to store the results without actually improving the precision. The subchannel model qualification in Section 5.0 demonstrates the acceptability of using maximum node lengths of 6 inches with axial node lengths of about 2 inches in the MDNBR region.

The length of the axial nodes should also be taken into account when running transient problems in order to satisfy the Courant number limit (The Courant number is defined as the axial velocity u times the numerical approximation of the time derivative - $u\Delta t/\Delta x$). The VIPRE-D solution methods are generally fully implicit and have no time step size limitations for numerical stability. However, solution instability could occur in transient calculations using a subcooled void model that was developed based on steady state data, such as the EPRI subcooled void model. In these cases, and to avoid numerical instabilities, appropriate time step sizes and axial node sizes are selected in transient heat flux and DNBR calculations to ensure that the Courant number is greater than one.

These modeling guidelines are consistent with the USNRC approved Dominion COBRA models (Reference 8) and with standard industry practice. Dominion VIPRE-D axial nodalizations are created according to these guidelines.

4.3 FUEL ROD MODELING

A typical VIPRE-D model defines the number of rods appropriate for the number of channels selected in the radial nodalization (Section 4.1), normally in accordance with the type of fuel present in the core, and uses the "dummy" rod model to represent them. In the dummy rod model there is no calculation of the heat transfer and the temperature distribution within the fuel rod, and the surface heat flux for each rod is specified as an input parameter. Unheated rods, such as instrument tubes and guide tubes, do not need to be modeled as rods. They are taken into account when calculating the flow area, the wetted and heated perimeters, and the crossflow gaps in the appropriate channels, but they are not modeled as separate entities. Dominion does not plan to use the conduction model present in the code.

The VIPRE-D model accounts for a fraction of the core power being generated directly in the coolant due to gamma heating and neutron absorption. For the safety analysis, it is assumed that 97.4% of the reactor power is generated within the fuel rods, and the remaining 2.6% is generated directly in the coolant. VIPRE-D fuel rod modeling and the treatment of the gamma heating is consistent with the USNRC approved Dominion COBRA production models (Reference 8).

4.4 POWER DISTRIBUTION

In the VIPRE-D model, an axial power profile is entered to specify the power generated by each axial node relative to the average. A radial power factor that determines the rod power relative to the average core power is assigned to each rod.

DNBR calculations are typically performed with reference axial power shapes. For example, the typical reference axial power shape used in establishing core thermal limits is a chopped cosine shape with a peak-to-average value of 1.55. This reference power shape is supplemented by other axial shapes skewed to the bottom or to the top of the core to determine the reduction of trip setpoints on excessive axial power imbalance. Dominion's VIPRE-D model interpolates in the axial power table using the spline fit option, as opposed to the default linear interpolation option. The spline fit option was added to VIPRE-01, MOD02.1 and provides a slightly smoother axial power profile integration. A sensitivity analysis of the impact of this option was performed by Dominion, and virtually identical MDNBR results were obtained with both options.

The radial power distribution is specified by assigning to each dummy rod a radial power factor that specifies the rod power relative to the average core power. The power distributions provide a gradual power gradient with the highest peaking around the hot channels (i.e., hot thimble cell and hot typical cell) to reduce the benefit of crossflow into the hot channel. The VIPRE-D models apply the peak $F\Delta H$ to a rod in the hot thimble cell and the hot typical cell. This radial modeling results in a conservative evaluation of DNBR in the hot channel and hot pin, since the mixing effects in the center of the core are significantly reduced. A typical radial power distribution for a 1/8th core model of 157 17x17 fuel assemblies, adjusted for a 1.587 maximum peaking factor, is described in Table 4.4-1.

Table 4.4-1. Typical Radial Peaking Factors for a 1/8th Core Model of 157 17x17 Fuel Assemblies Modeled with 12 Channels and 14 Rods

Rod Number	Relative Power f_i	Number of rods N_i	Statistical Maximum $F\Delta H$
			1.587
1	1.0	0.5	1.587
2	0.99748	0.5	1.583
3	0.993699	0.5	1.577
4	0.994959	1	1.579
5	0.986767	0.5	1.566
6	0.988658	1	1.569
7	0.996219	1	1.581
8	0.988028	0.25	1.568
9	0.986767	0.5	1.566
10	0.991178	0.5	1.573
11	0.983617	0.5	1.561
12	0.980466	0.125	1.556
13	0.982987	26.125	1.560
14	$\frac{\sum_{i=1}^{i=14} N_i - \sum_{i=1}^{i=13} F_{\max \Delta H} \cdot f_i \cdot N_i}{N_{14}}$	5148	0.99639

4.5 TURBULENT MIXING

The VIPRE-01 turbulent mixing model accounts for the exchange of energy and momentum between adjacent subchannels due to turbulence. This is not a turbulence model, but an attempt to empirically account for the effect of turbulent mixing. The following inputs are needed to setup this model:

- Turbulent Momentum Factor (FTM), which can range from 0.0 to 1.0, measures how efficiently the turbulent crossflow mixes momentum. The VIPRE-01 User's Manual (Reference 2) recommends a value of 0.8 for FTM and explains that VIPRE is not very sensitive to the value of FTM. In Dominion models FTM has been conservatively set to 0.0, which indicates that the turbulent crossflow mixes enthalpy only and not momentum. This modeling approach is consistent with USNRC approved Dominion COBRA models (Reference 8).

- The model for turbulent mixing chosen for single phase mixing describes the mixing as w'=A x S x G, where A is an empirical mixing coefficient (the variable ABETA in VIPRE-D) entered by the user, S is the rod-to-rod gap width (ft), and G is the average mass velocity in the channels linked by a given gap (lbm/ft^2-s). This coefficient ABETA, which depends on the particular fuel type and can range from 0.0 to 0.1, is typically set to 0.038. The two phase turbulent mixing is computed in the same way as the single phase. This is the default model in the code and it is consistent with USNRC approved Dominion COBRA models (Reference 8).

Since turbulent mixing is a subchannel phenomenon, the value of the turbulent mixing coefficient needs to be corrected for lumped channels to reflect the effect of lumping together many rod-to-rod gaps. The value of ABETA for the flow path between a subchannel and a lumped channel is defined as:

$$ABETA_{lumped} = ABETA_{subchannel} \times \frac{\text{SubchannelCentroidDistance}}{\text{LumpedChannelCentroidDistance}} \qquad [4.5.1]$$

The impact of correcting the value of the turbulent mixing coefficient for the flow paths connecting to lumped channels has been quantified with a sensitivity analysis, which demonstrated that both approaches yield essentially the same results. This methodology is consistent with standard industry practice. In larger lumped regions, on the order of a bundle or larger, turbulent mixing tends to be smeared out by the effect of averaging on both flow and enthalpy. As a consequence, the turbulent mixing coefficients for the flow paths between lumped channels are set to zero (Reference 4, Section 7.2).

4.6 AXIAL HYDRAULIC LOSSES AND CROSSFLOW RESISTANCE

Axial friction losses are calculated with the McAdams correlation, which has been shown to provide an excellent approximation to the Colebrook smooth pipe formulation for single phase axial friction factor for the range $3{\cdot}10^4 < Re < 2{\cdot}10^6$ (Reference 11). This is the same correlation used in the USNRC approved Dominion COBRA models (Reference 8).

$$F = MAX\,(0.184 * Re^{-0.2} + 0.0\ [\text{turbulent}],\ 64.0 * Re^{-1.0} + 0.0\ [\text{laminar}]) \qquad [4.6.1]$$

Lateral resistance for a subchannel is calculated in both the turbulent and laminar regions with a Blasius-type function of the gap Reynolds number, where the coefficient A is calculated using the Idel'Chik empirical correlation for a bundle of circular tubes in vertical columns (Reference 12, p.332).

$$K_G = A \cdot \mathrm{Re}_{lateral}^{\ -0.2} \qquad [4.6.2]$$

where A is defined as: $$A = 1.52 \cdot \left[\frac{SubchannelPitch}{FuelRodOD} - 1\right]^{-0.5} \qquad [4.6.3]$$

In order to correctly calculate the effective crossflow resistance for the lumped channels, the subchannel crossflow resistance is multiplied by the ratio of the lumped channel centroid distance and the subchannel centroid distance. This treatment is consistent with the USNRC SER for VIPRE-01 (Reference 6).

4.7 FORM LOSS COEFFICIENTS

The local form loss coefficients (FLC) associated with a given fuel assembly type are obtained by the vendor from full-scale hydraulic tests of the fuel assemblies. These form losses are specified for each fuel component (non-mixing grids, mixing grids, mid-span mixing grids, etc.) and for each type of subchannel (unit cell, corner cell, etc). Thus, VIPRE-D allows the definition of different FLCs for different channels and at different axial locations.

In the VIPRE-D models, the FLCs are axially placed at the upper edges of the axial nodes immediately below the corresponding component (mixing vane grids, mid-span mixing vane grids, etc). VIPRE-D places the pressure loss associated with a node at the top edge of the node, thus applying the pressure losses at the actual axial locations for each spacing grid. The impact of slightly varying (upward and downward) the axial location where the FLCs are applied was studied with a sensitivity analysis, which showed an insignificant change in DNBR.

4.8 TWO-PHASE FLOW AND HEAT TRANSFER CORRELATIONS

VIPRE-01 has a number of empirical correlations available to simulate two-phase flow effects (Reference 1). These correlations can be grouped in three major categories: 1) two-phase friction multipliers; 2) subcooled void correlations; and 3) bulk boiling void correlations. In Reference 4, a sensitivity study was performed to assess the differences in the performance of the various correlations and, although significant differences were not found, the EPRI models were defined as the default models for VIPRE-01. The USNRC, in Reference 6, concluded that the EPRI void models and EPRI correlation for two-phase friction are acceptable for licensing calculations.

Dominion performed yet another sensitivity study to verify that this set of two-phase flow correlations provided results approximate to results already approved by USNRC for the F-ANP AMBW fuel product. The set of two-phase flow correlations listed below was shown to provide the closest comparison to the USNRC approved F-ANP LYNXT code for F-ANP AMBW fuel products (Section 5.4) and was deemed to be the most suitable for Dominion applications. Dominion will apply this set of two-phase flow correlations for all applications unless future fuel types necessitate the use of a different set. In those cases, the selection of two-phase flow correlations will be described and justified in the appendix where the CHF correlations associated to that particular fuel type are qualified or verified.

The selections are:

- Subcooled Void Model: EPRI
- Bulk Boiling Void Model: EPRI
- Two-Phase Friction Multiplier: EPRI
- Hot Wall Friction Correlation : NONE

VIPRE-D also requires the user to select the heat transfer correlations that describe the boiling curve. These selections (except the Single Phase Forced Convection Correlation), however, are only applied to the heat transfer solution if the conduction model is used. Since Dominion VIPRE-D models described herein use the "dummy" rod model (Section 4.3), the conduction model is ignored.

The Single Phase Forced Convection is modeled with the standard Dittus-Boelter correlation, which is commonly used for this type of configuration (Reference 2).

$$h_{DB} = 0.023 \cdot \mathrm{Re}_l^{0.8} \cdot \mathrm{Pr}^{0.4} \cdot \frac{k}{D_e} \qquad [4.8.1]$$

where Re_l is the Reynolds number for the liquid, Pr is the Prandtl number, k is the thermal conductivity of the fluid (Btu/s-ft-°F) and D_e is the hydraulic diameter in ft. This selection is consistent with the USNRC approved Dominion COBRA models (Reference 8) and with standard industry practice.

4.9 CRITICAL HEAT FLUX CORRELATIONS

VIPRE-D currently includes several CHF correlations applicable to various F-ANP and Westinghouse fuel types. Dominion intends to add appendixes to the present report qualifying various CHF correlations for fuel products to be used within the Dominion nuclear units. This modular approach will allow simple submittals of additional CHF correlations for new fuel types in the future. The critical heat flux correlation to be used for a particular fuel type will be qualified in one of the appendixes and will have been approved by the USNRC for use with such fuel product.

The VIPRE-D CHF correlations will be used within the USNRC approved parameter ranges of the CHF correlations, including fuel assembly geometry and grid spacers. The DNBR design limits applied to each CHF correlation will be derived or verified using fluid conditions predicted by the VIPRE-D code.

4.10 ENGINEERING FACTORS

Variations in the fuel fabrication and core flow adverse to DNB margin are also considered in the VIPRE-D models. Typical VIPRE-D models account for engineering hot channel factors for both enthalpy-rise and heat flux, as well as for inlet flow maldistribution. These engineering factors are fuel product dependent.

Local Heat Flux Engineering Hot Channel Factor, F_Q^E:

F_Q^E accounts for pellet-to-pellet variations in enrichment, density and burnable absorber plus the effects of pellet-to-clad eccentricity and variations in the clad outer diameter. Used in the evaluation of the maximum linear heat generation rate, F_Q^E has been determined to have negligible effect on DNB, and it is not used for most fuel types. F_Q^E will be applied according to fuel vendor approved methodologies.

Engineering Enthalpy-Rise Hot Channel Factor, $F_{\Delta H}^E$:

$F_{\Delta H}^E$ accounts for variations in the fuel enrichment, density, rod dimensions and pin pitch that affect the heat generation rate along the flow channel. Uncertainties in these variables are determined from sampling of manufacturing data. For deterministic analyses, $F_{\Delta H}^E$ is incorporated in the model as a multiplier to the energy input to the hot channel without affecting the surface heat flux. In statistical DNBR methods, $F_{\Delta H}^E$ is statistically convoluted into the DNBR design limit.

Stack Height Reduction:

Active fuel stack height varies during reactor operation due to the combined effects of fuel densification, swelling and thermal expansion. However, the treatment of this phenomenon is vendor specific and fuel specific. VIPRE-D models comply with the treatment specified by the fuel vendor.

Inlet Flow Reduction:

Core inlet flow maldistribution accounts for non-uniform flow distribution into each fuel assembly at the core inlet. Consistent with the USNRC approved Dominion COBRA methodology for PWR applications (Reference 8), a 5% flow reduction (maldistribution) to the hot assembly is applied in VIPRE-D models.

4.11 BOUNDARY CONDITIONS

The VIPRE-D models require the following parameters as the input or the boundaries for calculations:

- Core inlet temperature or enthalpy
- Core average power
- System pressure
- Core inlet flow rate

- Core power distributions

The core inlet temperature and inlet flow may be uniform or non-uniform, depending on the core conditions being analyzed. The core power defines the thermal energy entering the fluid through the fuel rods. The system pressure is assumed to be uniform throughout the VIPRE-D model. The core inlet flow conservatively excludes flow through bypass leakage, such as through the guide tubes.

The core boundary conditions for VIPRE-D transient calculations can be obtained from system computer codes and neutronic codes. For example, the system code provides time-dependent reactor coolant system pressure, core average power, core flow rate and core inlet temperature for transient DNBR calculations. The neutronic codes provide core power distributions and nuclear peaking factors such as FΔH.

4.12 RUN CONTROL PARAMETERS

The run control parameters determine the maximum and minimum number of iterations to be performed to find a solution, as well as the convergence limits and the damping factors used. After a careful review, these values have been set to the defaults provided by the code (Reference 2). In a few occasions, when convergence problems have been reported by the code, the damping factors and/or the convergence limits have been adjusted in the models to allow the code to converge. These convergence problems do not necessarily mean bad results or false convergence, just some numerical instability. Indeed, in most occasions, the results obtained by the code with the adjusted convergence limits or damping factors are nearly identical to the non-converging results (Reference 6, Section 2.1).

The VIPRE-01 solution methods are generally fully implicit and have no time step size limitations for numerical stability. However, solution instability could occur in transient calculations using a subcooled void model that was developed based on steady state data, such as the EPRI subcooled void model. In these cases, and to avoid numerical instabilities, appropriate time step sizes and axial node sizes are selected in transient heat flux and DNBR calculations to ensure that the Courant number is greater than one. This modeling guideline is consistent with VIPRE-01 SER Restriction #4 (see Reference 6 and Section 2.2 herein).

5.0 QUALIFICATION OF THE VIPRE-D SUBCHANNEL MODEL

The analyses shown in this section demonstrate that Dominion VIPRE-D models created using the selections and modeling guidelines described in Section 4 of this report provide close comparison to other USNRC approved subchannel codes. This section is provided as an example to demonstrate in sufficient detail the validity of the methodology discussed herein, and it is not meant to be linked to a specific plant or fuel product.

5.1 STEADY STATE APPLICATION

Dominion created a 12-channel model for F-ANP AMBW fuel at North Anna Power Station in accordance with the methodology described in Section 4 of this report. This VIPRE-D model of the 1/8th North Anna core consists of 12 channels (10 subchannels and 2 lumped channels) and 14 rods, as shown in Figure 5.1-1. The axial nodalization used in this model has been customized for F-ANP AMBW fuel assemblies and contains 87 non-uniform axial nodes with typical node lengths of 2 inches and a maximum node length of 6 inches. The reference axial power profile (1.55 chopped cosine) was defined as an axial power profile table with 37 points. All other axial power shapes are defined as axial power profile tables with 32 points.

The AMBW fuel assembly consists of 264 fuel rods with an outside diameter of 0.374 inches arranged in a 17x17 matrix with a pin pitch of 0.496 inches. The AMBW fuel contains several advanced design features, such as mixing vane grids (MVG) and mid-span mixing grids (MSMG) in the upper two thirds of the heated length (Reference 13). The local FLCs used in this VIPRE-D 12-channel model were developed by F-ANP from full-scale hydraulic tests.

The F-ANP BWU CHF correlations, which have been specifically developed for use with the AMBW fuel, were used in the 12-channel model. There are three BWU CHF correlations that constitute the licensing basis for the F-ANP AMBW fuel assembly. These correlations use the same basic equation, but are fit to different databases (References 9 and 10). VIPRE-D applies different BWU correlations at different axial levels, according to the following guidelines:

- BWU-N, which is only applicable in the presence of non-mixing vane grids (NMVG), is used from the beginning of the heated length to the leading edge of the first structural MVG (Reference 9).
- BWU-Z, 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 9).
- BWU-ZM, which is just BWU-Z with a multiplicative enhancement factor and is applicable in the presence of MSMGs, is used from the leading edge of the second structural MVG to the leading edge of the last structural MVG (Reference 10).

- For the uppermost span, 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 heated length) (Reference 9).

VIPRE-D benchmark calculations were performed with the F-ANP LYNXT code and the 12-channel model created by F-ANP to model North Anna Power Station cores containing AMBW fuel assemblies. This benchmark uses 173 statepoints 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) axial power shapes (AOs); (c) elevated hot rod power (misaligned rod); and (d) low flow (LOFA and LOCROT). The 173 statepoints cover the full range of conditions and axial offsets in the North Anna UFSAR Chapter 15 evaluations (except for MSLB that is discussed in Section 5.2), and were specifically selected to challenge the three BWU CHF correlations (Table 5.1-1).

This benchmark study showed an average deviation between VIPRE-D and LYNXT of less than 0.14% in DNBR, with a maximum deviation of 2.2%. These results are well within the uncertainty typically associated with thermal-hydraulic codes, which has been quantified to be 5% (Reference 15), and justify the model selections in Section 4. Figure 5.1-2 shows graphically the performance of VIPRE-D versus LYNXT for the 173 statepoints. 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 2.1-1 (MSLB will be discussed in Section 5.2).

Table 5.1-1: Range of VIPRE-D / LYNXT 173 Benchmark Statepoints

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

Figure 5.1-1. Typical North Anna VIPRE-D 12-Channel Model
for F-ANP AMBW Fuel Assemblies

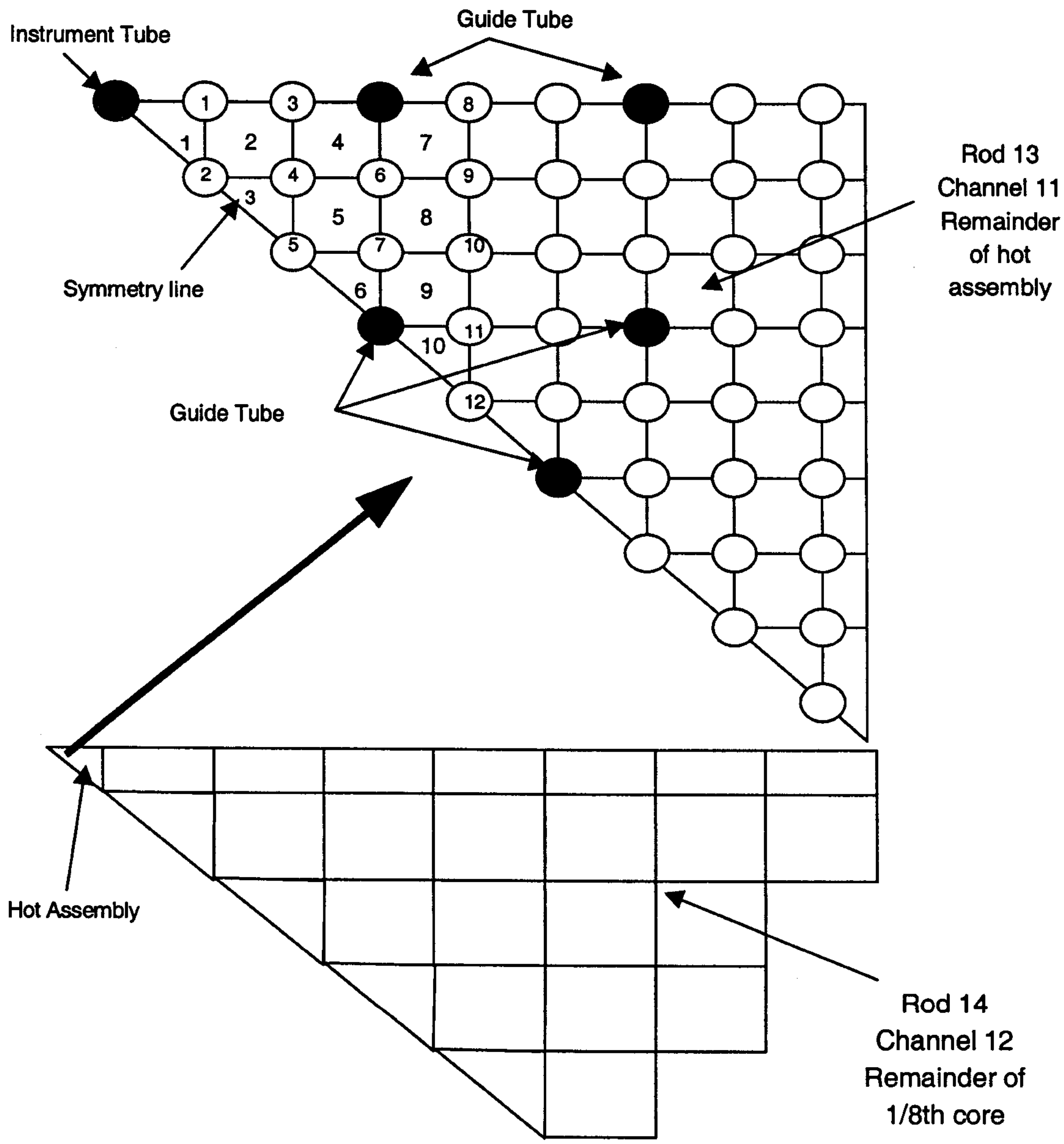
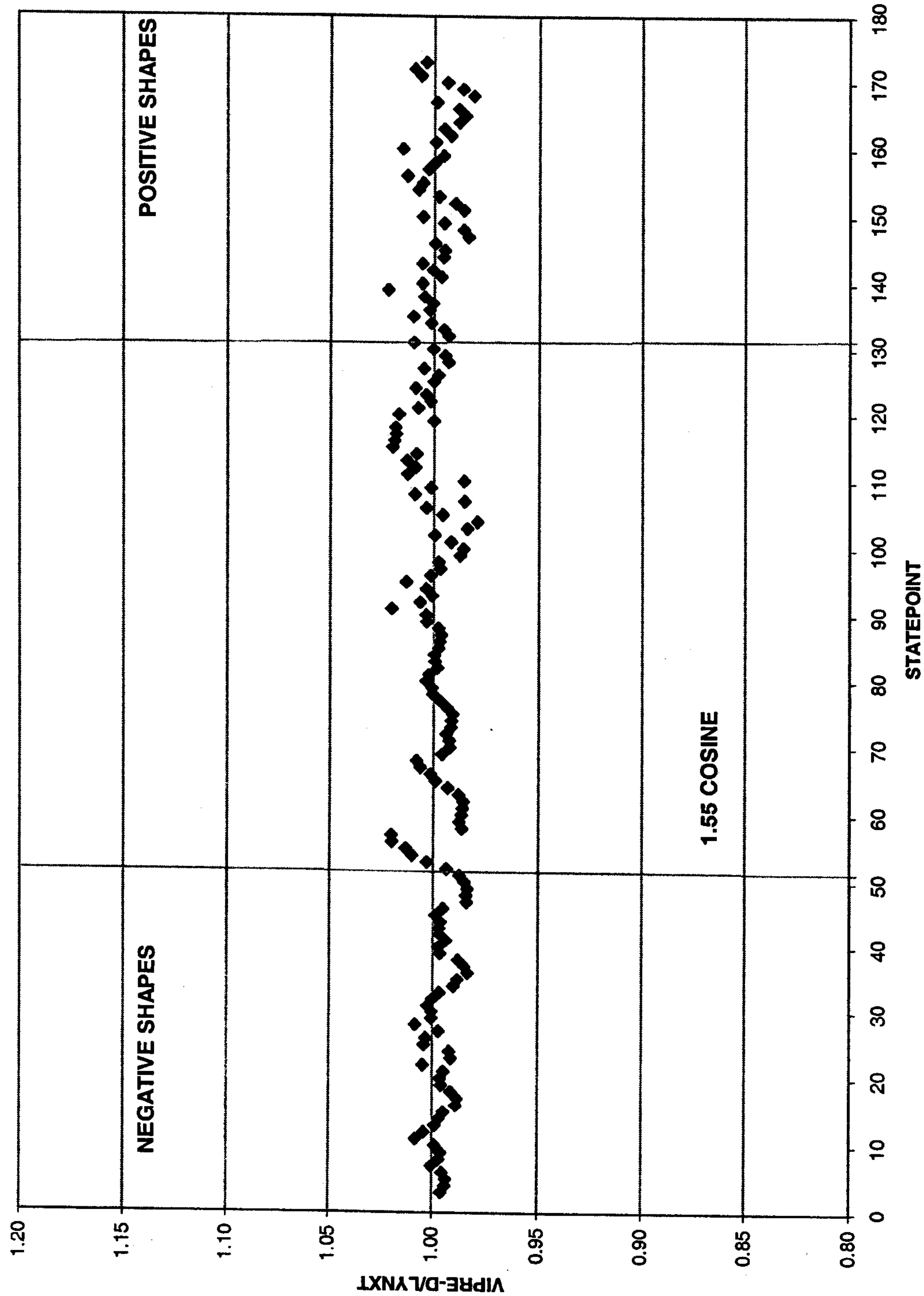


Figure 5.1-2: VIPRE-D vs. LYNXT for the 173 Statepoints



5.2 MAIN STEAM LINE BREAK APPLICATION

The 12-channel model discussed in section 5.1 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 a MSLB event.

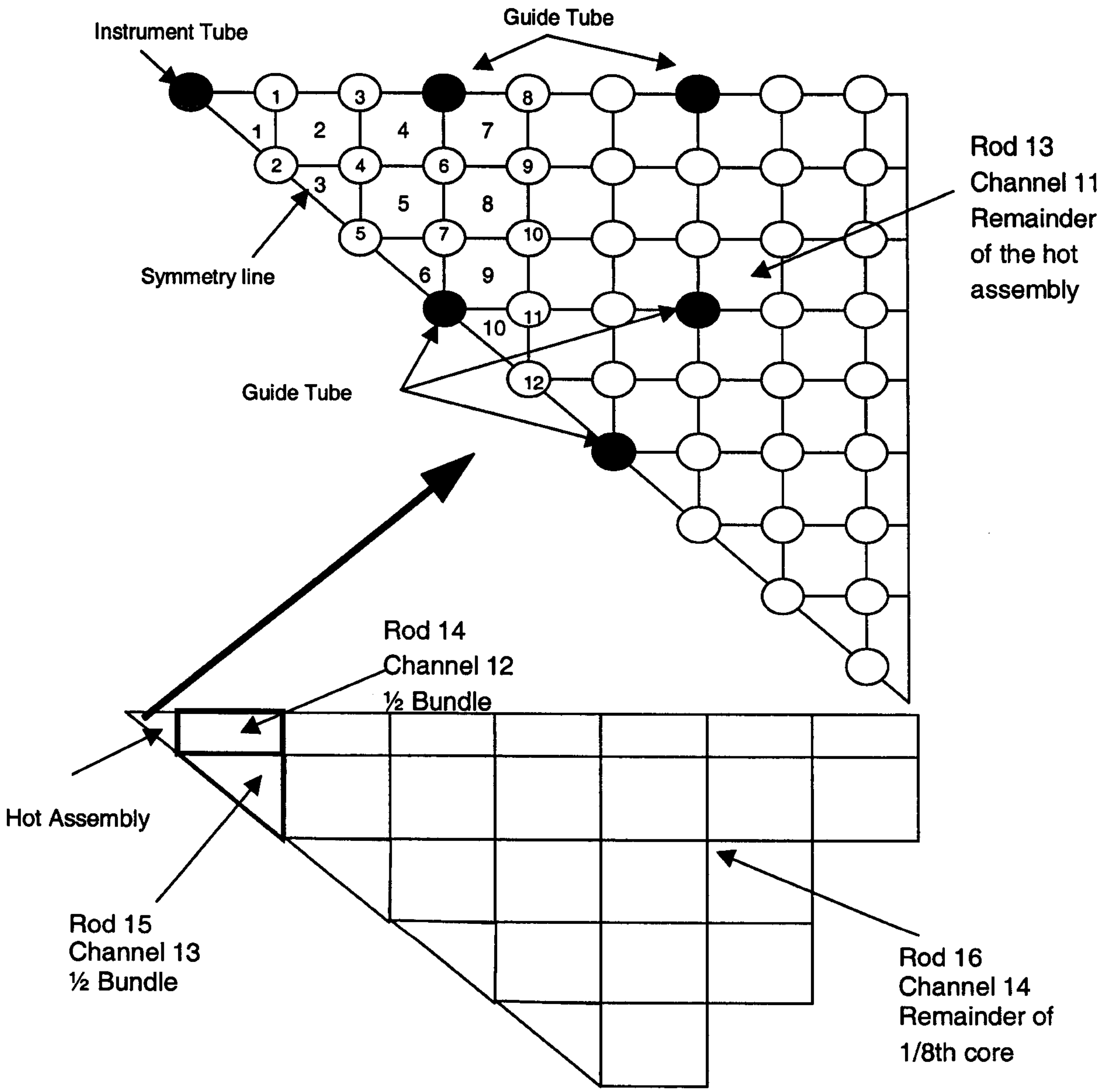
The VIPRE-D 14-channel model for a North Anna core containing F-ANP AMBW fuel assemblies consists of 14 channels (10 subchannels and 4 lumped channels) and 16 rods as shown in Figure 5.2-1. The two additional channels provide adequate solution detail of the flow field in the vicinity of the hot assembly and allow the modeling of the peaking and inlet boundary conditions in the fuel assemblies adjacent to the hot assembly.

The 14-channel model defines the inlet temperature for each one of the 14 channels. In addition, the inlet flow fraction is also specified for each of 14 channels. This modeling choice is of key importance for MSLB events, since the inlet temperature may change for each channel and it is then necessary to adjust the flow fraction to obtain the appropriate values of core inlet flow rate and channel flow rate.

The results from the VIPRE-D 14-channel model were compared to the results from a F-ANP LYNXT model for high flow (with offsite power) and low flow (without offsite power) MSLB evaluations. The results obtained show a maximum deviation of 2.12% in DNBR. These results demonstrate that VIPRE-D can analyze a MSLB event, provided the model has sufficient detail surrounding the hot assembly, such as the 14-channel model described here.

In addition, the accuracy of the 14-channel model was demonstrated through comparison with the DNBR results of the 173 statepoints obtained with the VIPRE-D 12-channel model. As discussed in Section 5.1, this set of statepoints is consistent with the list of intended applications of the VIPRE-D code discussed in Section 2.1. (Table 2.1-1). This comparison shows that there is essentially no difference between the 12-channel and the 14-channel models (the average deviation in DNBR is 0.03%), which indicates that VIPRE-D models created following the methodology discussed in Section 4 of this report are adequate.

Figure 5.2-1. Typical VIPRE-D 14-Channel Model for North Anna Cores with F-ANP AMBW Fuel



5.3 TRANSIENT APPLICATION

VIPRE-D has the capability to perform transient calculations by using boundary conditions obtained from a reactor systems code or a neutronic code. The reactor systems code provides time-dependent forcing functions for pressure, core average power, core flow rate and core inlet temperature and the neutronics code provides core power distributions and nuclear peaking factors.

VIPRE-D transient capability was tested by performing several sample transient calculations, two of which are described in this report. These two transient calculations were only intended to be samples designed to exercise the transient capabilities of the VIPRE-D code and a typical VIPRE-D model created according to the guidelines discussed in Section 4. In both cases, the behavior of the VIPRE-D results was successfully compared to the behavior to the COBRA analysis of record in the UFSAR. In addition, the VIPRE-D transient results were benchmarked against the steady state analysis of the most limiting statepoint in the transient. Two statepoints were selected in each case, the statepoint with the highest value of the power to mass flow ratio, and the limiting statepoint determined in the transient calculation (if different).

As discussed in Section 4.12, a numerical instability could occur in transient calculations using a subcooled void model that was developed based on steady state data, such as the EPRI model. For that reason, in order to avoid numerical instabilities, the time steps used for these transient simulations were selected to ensure that the Courant number is greater than one.

The damping factors and the convergence limits were set to the defaults provided by the code (Section 4.12). In a few occasions, when convergence problems were reported by the code, the damping factors and/or the convergence limits were adjusted in the models to allow the code to converge. These convergence problems do not necessarily mean bad results or false convergence, just some numerical instability. Indeed, in most occasions, the results obtained by the code with the adjusted convergence limits or damping factors were nearly identical to the non-converging results.

The first sample transient selected to verify the capabilities of the VIPRE-D code and the 12-channel model was the RWAP accident. Forcing functions for the RWAP transient were obtained from a NAPS UFSAR case (Dominion COBRA analysis of record for Westinghouse fuel). The length of the transient was 4.0 seconds, with a 0.05-second timestep. VIPRE-D results show similar behavior to the COBRA analysis of record in the UFSAR, but the MDNBR results are different because the analyses use different fuel types and CHF correlations (see Figure 5.3.1). Comparison with the results of the steady state calculation of the limiting statepoint show MDNBR values that are essentially the same as the results obtained in the transient.

The second sample transient selected to perform this verification was the LOFA. Forcing functions for the LOFA transient were obtained from the NAPS UFSAR. In particular, COBRA forcing functions were obtained for a F-ANP uprated core tripping on reactor coolant pump undervoltage. The length of the transient was 20.4 seconds, with a 0.1-second timestep. COBRA analysis of record and VIPRE-D calculations exhibited similar behavior, but the MDNBR results are different because the analyses use different fuel types and CHF correlations (see Figure 5.3.2). Comparison with the results of the steady state calculation of the limiting statepoint show MDNBR values that are essentially the same as the results obtained in the transient.

The transient analyses demonstrate that VIPRE-D is capable of performing stable transient calculations and the results obtained are adequate. Table 5.3-1 summarizes the results of the transient analysis.

Table 5.3-1: Summary of VIPRE-D Sample Transients

RWAP Sample Transient					
	POWER [MBtu/hr-ft^2]	FLOW [gpm]	TEMPERATURE [°F]	PRESSURE [psia]	DNBR
INITIAL CONDITION	0.20578	2.469	553.7	2250.0	2.847
LIMITING CONDITION [2.75 s]	0.22290	2.467	553.9	2286.5	2.597 [transient] 2.598 [steady state]
LOFA Sample Transient					
	POWER [MBtu/hr-ft^2]	FLOW [gpm]	TEMPERATURE [°F]	PRESSURE [psia]	DNBR
INITIAL CONDITION	0.20578	2.469	553.7	2250.0	2.847
LIMITING CONDITION [9.4 s]	0.19726	1.649	552.9	2360.7	1.820 [transient] 1.796 [steady state]

Figure 5.3-1: VIPRE-D RWAP Transient Sample Calculation Results

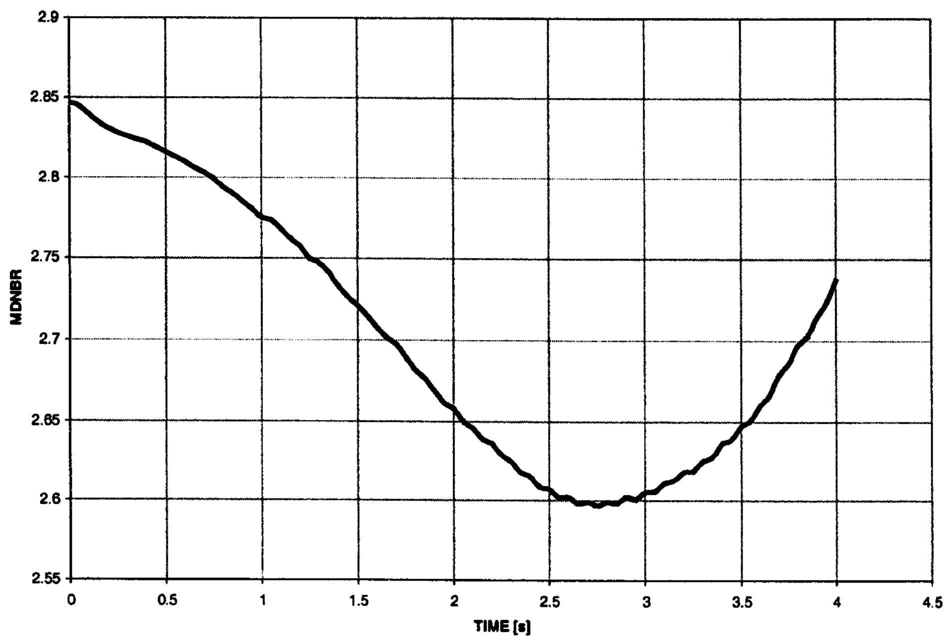
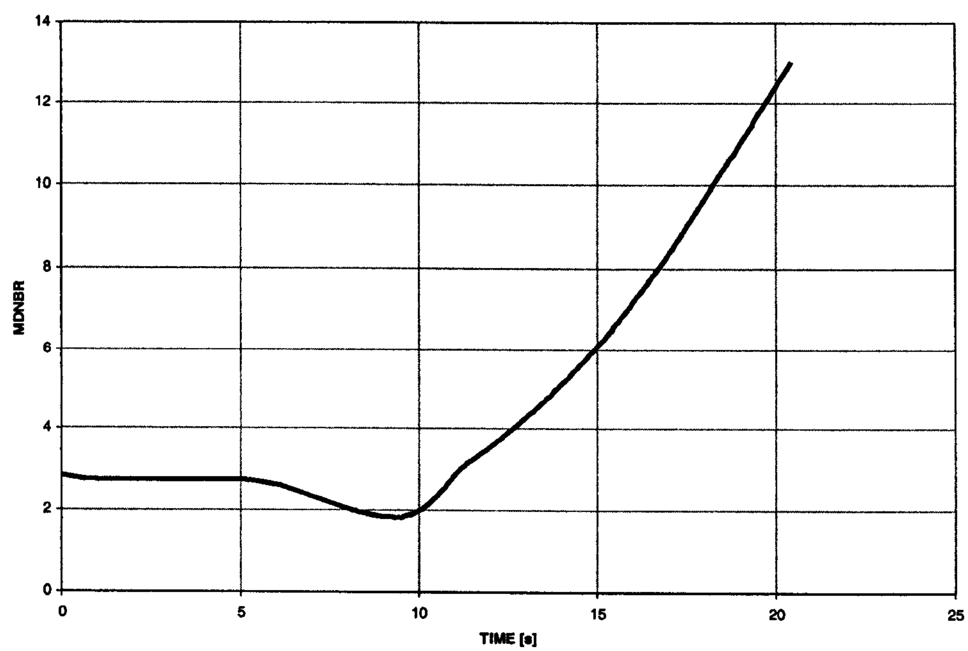


Figure 5.3-2: VIPRE-D LOFA Transient Sample Calculation Results



5.4 SENSITIVITY STUDIES

VIPRE-D has a number of empirical correlations available to simulate two-phase flow effects (Reference 1). These correlations can be grouped in three major categories: 1) two-phase friction multipliers; 2) subcooled void correlations; and 3) bulk boiling void correlations. In Reference 4 (Section 3.0), a sensitivity study was performed to assess the differences in the performance of the various correlations and, although significant differences were not found, the EPRI models were chosen as the default models for VIPRE-01. The USNRC staff reviewed these sensitivity studies and concluded in the SER for VIPRE-01 MOD-01 (Reference 6) that the EPRI void models and the EPRI correlations for two-phase friction are acceptable for licensing calculations.

Dominion performed another sensitivity study to determine the set of two-phase flow correlations most suitable for Dominion models. This sensitivity analysis provides justification for Dominion's modeling assumptions as discussed in Section 4.8, thus fulfilling condition (3) of the SER for VIPRE-01 MOD-01 (Reference 6). A detailed analysis of the available correlations was performed, including the modeling assumptions used in deriving the various correlations and four sets of correlations were chosen. The selected sets use together only those correlations that have consistent or complementary bases and take advantage of previous industry experience and vendor recommendations. The four cases studied were:

- Case 1 (EEE)
  Subcooled Void Model: EPRI
  Bulk Boiling Void Model: EPRI
  Two-Phase Friction Multiplier: EPRI

- Case 2 (LSE)
  Subcooled Void Model: LEVY
  Bulk Boiling Void Model: SMITH
  Two-Phase Friction Multiplier: EPRI

- Case 3 (LHH)
  Subcooled Void Model: LEVY
  Bulk Boiling Void Model: HOMOGENEOUS
  Two-Phase Friction Multiplier: HOMOGENEOUS

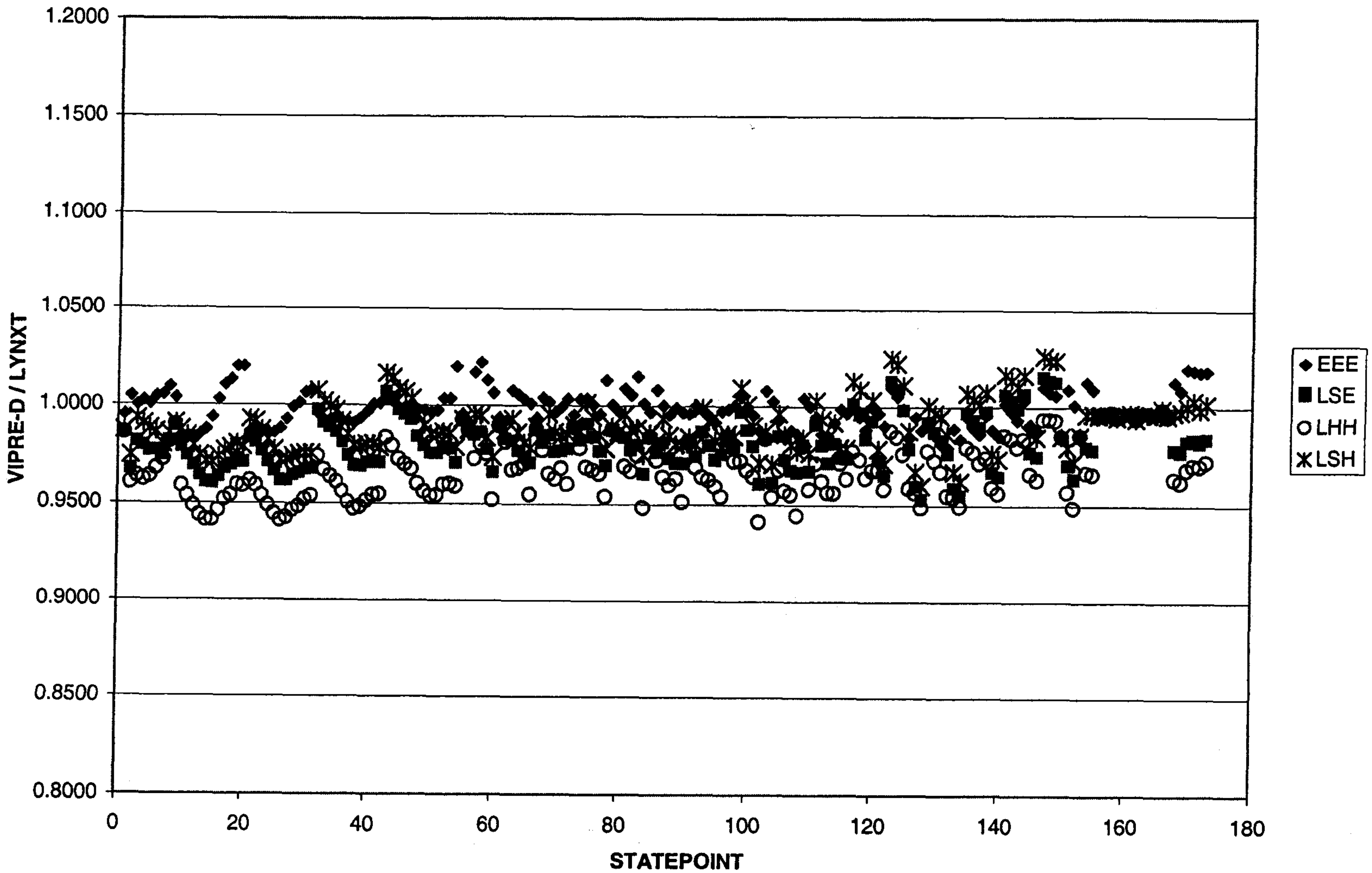
- Case 4 (LSH)
  Subcooled Void Model: LEVY
  Bulk Boiling Void Model: SMITH
  Two-Phase Friction Multiplier: HOMOGENEOUS

The 173 statepoints and the typical 12-channel model described in section 5.1 were executed by VIPRE-D using the four sets of two-phase models and correlations. The results were compared to the results of the USNRC approved code F-ANP LYNXT. Table 5.4-1 lists the average and maximum percent deviations in DNBR between the codes and Figure 5.4-1 shows the same results graphically. The set of EPRI correlations (option EEE), which is the default in the code, was then selected for VIPRE-D models as discussed in Section 4.8.

Table 5.4-1: Statistical Analysis of the MDNBR Results for the Four Sets of Two-Phase Models

	% DEVIATION IN DNBR LYNXT – VIPRE / LYNXT			
	EEE	LSE	LHH	LSH
AVERAGE	**0.14**	1.87	3.21	1.00
STANDARD DEVIATION	0.89	1.26	1.48	1.28

Figure 5.4-1: VIPRE-D vs. LYNXT for the Four Sets of Two-Phase Models



6.0 CONCLUSIONS

The VIPRE-01 code has been approved by the USNRC and is widely used throughout the nuclear industry for PWR safety analyses. VIPRE-D is the Dominion version of VIPRE-01. Other than the addition of vendor proprietary CHF correlations and minor input/output customizations, VIPRE-D is equivalent to VIPRE-01 as Dominion has preserved all the USNRC approved constitutive models and algorithms in the code. Dominion has shown VIPRE-D compliance with the requirements of the USNRC SERs regarding VIPRE-01 code applications. Dominion has validated VIPRE-D with extensive code benchmark calculations using the modeling methods outlined in this report, and the accuracy of the VIPRE-D models has been demonstrated through comparisons with other NRC-approved methodologies. VIPRE-D has been shown to meet or exceed the same standards for accuracy as other methodologies currently being used by Dominion and approved by the USNRC.

VIPRE-D includes several CHF correlations applicable to various F-ANP and Westinghouse fuel types, and the qualification of each one of them will be documented in the appendixes to this report. The critical heat flux correlation to be used for a particular fuel type will be documented and qualified in one of the appendixes and will have been approved by the USNRC for use with such fuel product prior to use by Dominion. The VIPRE-D CHF correlations will be used within the USNRC approved parameter ranges of the CHF correlations, including fuel assembly geometry and grid spacers. The DNBR design limits applied to each CHF correlation will be derived or verified using fluid conditions predicted by the VIPRE-D code.

With the modeling methods outlined in this report, and in conjunction with the appropriate CHF correlation and DNBR design limits qualified in the appendixes to this report, Dominion plans to use the VIPRE-D code for:

1) Analysis of 14x14, 15x15 and 17x17 fuel in PWR reactors.

2) Analysis of DNBR for statistical and deterministic transients in the Updated Final Safety Analysis Report (UFSAR), as identified in Table 2.1-1. Additional DNBR transients that are plant specific may be analyzed in a plant specific application that would be submitted to the USNRC for review and approval.

3) Steady state and transient DNB evaluations.

4) Development of reactor core safety limits (also known as core thermal limit lines, CTL).

5) Providing the basis for reactor protection setpoints.

6) Establishing or verifying 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 appendix to this document.

7.0 REFERENCES

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