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Dominion®

August 4, 2009

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	09-479
NL&OS/ETS:	R0
Docket Nos.	50-280/281
	50-338/339
	50-336/423
	50-305
License Nos.	DPR-32/37
	NPF-4/7
	DPR-65/NPF-49
	DPR-43

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
DOMINION NUCLEAR CONNECTICUT, INC. (DNC)
DOMINION ENERGY KEWAUNEE, INC. (DEK)
SURRY AND NORTH ANNA POWER STATIONS UNITS 1 AND 2
MILLSTONE POWER STATION UNITS 2 AND 3
KEWAUNEE POWER STATION
APPROVED TOPICAL REPORT DOM-NAF-2, Revision 0.1-A

In accordance with the NRC guidelines, Dominion, DNC and DEK are hereby submitting the published version of DOM-NAF-2, Rev. 0.1-A with appendices A, B, and C. The information requested in the guidelines has been incorporated into the published Topical Report DOM-NAF-2, Rev. 0.1-A. Under separate letter, three copies of DOM-NAF-2, Rev. 0.1-A with appendices A, B, and C have been provided to Dr. V Sreenivas, NRC Licensing Project Manager for North Anna Power Station.

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

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering
Virginia Electric and Power Company
Dominion Nuclear Connecticut, Inc.
Dominion Energy Kewaunee, Inc.

Attachment

Commitments made in this letter: None

T010
~~T007~~
NRR

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Attachment

Topical Report DOM-NAF-2, Rev. 0.1-A

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

Virginia Electric and Power Company (Dominion)
Dominion Nuclear Connecticut, Inc. (DNC)
Dominion Energy Kewaunee, Inc. (DEK)

Dominion®

Fleet Report DOM-NAF-2, Rev. 0.1-A

(with Appendixes A, B & C)

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

Nuclear Analysis and Fuel
Nuclear Engineering

July 2009

DOM-NAF-2, Rev. 0.1-A
Including Appendixes A, B and C

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

NUCLEAR ANALYSIS AND FUEL DEPARTMENT
DOMINION
RICHMOND, VIRGINIA
July 2009

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Approved:

K. L. Basehore
Director, Nuclear Analysis and Fuel

Fleet Report DOM-NAF-2, Rev. 0.1-A includes:

- Safety Evaluation by NRC, dated April 4, 2006 (DOM-NAF-2, including Appendixes A & B)
- Corrections to Safety Evaluation by NRC, dated June 23, 2006
- Safety Evaluation by NRC, dated April 22, 2009 (Appendix C)
- Classification / Disclaimer
- Abstract
- Acknowledgements
- Fleet Report DOM-NAF 2
- Appendix A to Fleet Report DOM-NAF-2
- Appendix B to Fleet Report DOM-NAF-2
- Appendix C to Fleet Report DOM-NAF-2

ATTACHMENTS:

1: Request For Additional Information, Set 1, Questions And Answers (31 pages).

2: Information Regarding a LYNXT Error Supporting the Request for Approval of Fleet Report DOM-NAF-2 (5 pages).

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

April 4, 2006

SERIAL # 06-319

REC'D APR 11 2006

NUCLEAR LICENSING

Mr. David A. Christian
Senior Vice President
and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
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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,

CI Grimes

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

Virginia Electric and Power Company

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

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:

- BWU-N, 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).

- 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 6).

- BWU-ZM, 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).

- 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 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	$\leq$0.30
Local Heat Flux [Mbtu/hr-ft^2]	$\leq$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.

5. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), "Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, 'VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores,' (TAC No. M79498)," dated October 30, 1993.

6. Topical Report, BAW-10199P-A, Addendum 1, "The BWU Critical Heat Flux Correlations," Framatome Cogema Fuels, August 1996, dated December 6, 2000.

7. Topical Report, BAW-10199P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," Framatome Cogema Fuels, dated September 5, 2002

8. Letter from L. N. Hartz (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes and Exemption Request, Use of Framatome ANP Advanced Mark-BW Fuel," dated March 28, 2002.

9. Technical Report, BAW-10156-A, Revision 1, "LYNXT, Core Transient Thermal-Hydraulic Program," Framatome ANP, August 1993.

10. Technical Report, AEC-TR-6630, "Handbook of Hydraulic Resistance, Coefficients of Local Resistance and of Friction," I. E. Idel'Chik, 1960.

11. Topical Report, VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," R. C. Anderson and N. P. Wolhope, July 1990.

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

13. 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, and Millstone Power Station Units 2 and 3 Request for Additional Information on Topical Report DOM-NAF-2: Reactor Core Thermal-Hydraulics using the VIPRE-D Computer Code Including Appendices A and B," dated June 30, 2005.

14. 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, Information Regarding a LYNXT Error Supporting the 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 8, 2005.

Principal Contributor: L. Ward

Date: April 4, 2006

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

June 23, 2006

SERIAL # 06-560

REC'D JUL 5 2006

NUCLEAR LICENSING

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, NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, AND SURRY POWER STATION, UNIT NOS. 1 AND 2 - CORRECTION TO DOMINION'S FLEET REPORT DOM-NAF-2, "REACTOR CORE THERMAL-HYDRAULICS USING THE VIPRE-D COMPUTER CODE"

Dear Mr. Christian:

On April 4, 2006, the Nuclear Regulatory Commission (NRC) issued Fleet Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code." Subsequently, in its letter dated June 1, 2006, you informed the NRC that the safety evaluation (SE) for Fleet Report DOM-NAF-2 contained several editorial errors, including language that could unnecessarily restrict the use of the fleet report to specific fuel vendors. The corrected pages for this SE are enclosed with this letter. The revisions to the SE are identified by lines in the margin.

Sincerely,

Stephen R. Monarque, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
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

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

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 NRC staff-approved pressurized water reactors (PWR) fuel types in the licensees' nuclear steam supply systems (NSSS).

Corrected by letter dated June 23, 2006

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 NRC staff-approved PWR 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 NRC staff-approved PWR 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 NRC staff-approved PWR 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.

3.3 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 NRC staff-approved PWR 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 3). 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).

Corrected by letter dated June 23, 2006

- 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

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	$\leq$0.30
Local Heat Flux [Mbtu/hr-ft^2]	$\leq$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.

Corrected by letter dated June 23, 2006

(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 1000 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 NRC staff-approved PWR fuel types. 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

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

SERIAL # 09-290

April 22, 2009

RECD APR 28 2009

NUCLEAR LICENSING

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

SUBJECT: KEWAUNEE POWER STATION, MILLSTONE POWER STATION, UNITS 2 AND 3, NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, AND SURRY POWER STATION, UNIT NOS. 1 AND 2 – APPENDIX C TO DOMINION FLEET REPORT DOM-NAF-2, "QUALIFICATION OF THE WESTINGHOUSE WRB-2M CHF CORRELATION IN THE DOMINION VIPRE-D COMPUTER CODE" (TAC NOS. MD8703, MD8704, MD8705, MD8706, MD8707, MD8708, MD8709)

Dear Mr. Christian:

By letter dated April 4, 2008, Dominion Energy Kewaunee, Inc., Dominion Nuclear Connecticut, Inc., and Virginia Electric and Power Company (Dominion), submitted an application to use Appendix C to Dominion Fleet Report DOM-NAF-2, "Qualification of the Westinghouse WRB-2M CHF [Critical Heat Flux] Correlation in the Dominion VIPRE-D Computer Code" to the U.S. Nuclear Regulatory Commission (NRC) for Kewaunee Power Station, Millstone Power Station, Units 2, and 3, North Anna Power Station, Unit Nos. 1 and 2, and Surry Power Station, Unit Nos. 1 and 2, respectively. The purpose of this report was to justify the use of the previously approved WRB-2M CHF Correlation in the previously approved Dominion VIPRE-D Computer Code. The proposed change would allow Dominion to use the WRB-2M CHF Correlation in VIPRE-D when performing thermal-hydraulic analysis on 17x17 Robust Fuel Assembly fuel.

On the basis of its review, the NRC staff finds the licensee's request acceptable. The enclosed safety evaluation documents the findings. Please contact me at (301) 415-1864, if you have any questions on this matter.

Sincerely,

Donna N. Wright, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-305, 50-336, 50-423,
50-338, 50-339, 50-280, and 50-281

Enclosure: *Safety Evaluation*

cc: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO APPENDIX C TO DOMINION FLEET REPORT DOM-NAF-2,

"QUALIFICATION OF THE WESTINGHOUSE WRB-2M CHF CORRELATION

IN THE DOMINION VIPPRE-D COMPUTER CODE"

DOMINION ENERGY KEWAUNEE, INC., DOMINION NUCLEAR CONNECTICUT, INC.,

VIRGINIA ELECTRIC AND POWER COMPANY

KEWAUNEE POWER STATION, MILLSTONE POWER STATION, UNITS 2 AND 3,

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-305, 50-336/423, 50-338/339, AND 50-280/281

1.0 INTRODUCTION

By letter dated April 4, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080980229), Dominion Energy Kewaunee, Inc., Dominion Nuclear Connecticut, Inc., and Virginia Electric and Power Company (Dominion), submitted Appendix C to Dominion Fleet Report DOM-NAF-2, Rev 0.0, and "Qualification of the Westinghouse WRB-2M CHF [Critical Heat Flux] Correlation in the Dominion VIPRE-D Computer Code" (Reference 1) to the Nuclear Regulatory Commission (NRC) for Kewaunee Power Station, Millstone Power Station, Units 2 and 3, North Anna Power Station, Unit Nos. 1 and 2, and Surry Power Station, Unit Nos. 1 and 2, respectively. The purpose of this report was to justify the use of the previously approved WRB-2M CHF Correlation (Reference 2) in the previously approved Dominion VIPRE-D Computer Code (Reference 3). The proposed change would allow Dominion to use the WRB-2M CHF Correlation in VIPRE-D when performing thermal-hydraulic analysis on 17x17 Robust Fuel Assembly (RFA) fuel.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.34, "Contents of construction permit and operating license applications; technical information," requires that Safety Analysis Reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload design process, licensees are responsible for reload safety evaluations to ensure that their safety analyses remain bounding

Enclosure

for the design cycle. To confirm that the analyses remain bounding, licensees confirm those key inputs to the safety analyses (such as the CHF) are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a re-analysis or a re-evaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

The NRC staff's review was based on the evaluation of the technical merit of the submittal and compliance with any applicable regulations associated with the review of topical reports.

3.0 TECHNICAL EVALUATION

3.1 Background Information

Boiling crisis occurs when the boiling water flowing past a fuel rod transitions from nucleate boiling to film boiling. This transition decreases the heat transfer rate at the fuel rod surface, forcing the fuel rod surface temperature to dramatically increase in order to maintain the same total heat transfer. This large increase in fuel rod surface temperature may lead to fuel damage. The heat flux which causes this transition from nucleate boiling to film boiling is known as CHF. To prevent possible fuel damage, boiling crisis is prevented via correlations used to predict the CHF. For normal reactor operations, thermal-hydraulic analysis is used to demonstrate that the peak heat flux in the core will remain below the CHF.

In pressurized-water reactors (PWR), CHF is primarily a local phenomenon caused by bubbles which crowd the surface of the fuel rod. If the bubbles prevent the cooling water from reaching the surface of the fuel rod, the flow can transition from nucleate boiling to film boiling. This form of CHF happens very quickly and is known as departure from nucleate boiling (DNB). Keeping with common practice, DNB and CHF are used interchangeably. Many parameters can impact DNB such as: flow pattern, bubble size and population, bubble layer thickness, wall superheat and flow memory, flow instability, local pressure, local enthalpy, mass velocity, inlet conditions, heated length, rod bundle shape, grid spacers, and others (Reference 4). Due to the complex nature of the phenomenon of DNB, the functional form of DNB correlations are generally empirical and are often based solely on experimental observations of the relationship between the measured DNB and the measured DNB parameters.

To prevent DNB, the departure from nucleate boiling ratio (DNBR) is used. DNBR is the ratio of the CHF at a location along the fuel rod divided by the current heat flux at that same location under the same flow conditions. To ensure an accurate prediction of DNBR, CHF experiments are performed in which the heat flux in prototypical fuel assemblies increases to the point that CHF is reached. The flow conditions of the experiment are measured and those same flow conditions are input into a specific thermal-hydraulic computer code with a specific CHF correlation. The measured CHF value from the test is compared with the predicated CHF value from the thermal-hydraulic computer code and an analysis is performed to determine if the computer code with the specific CHF correlation can accurately predict the CHF behavior of the fuel assembly.

The WRB-2M CHF correlation along with Dominion VIPRE-D thermal-hydraulic computer code were used to predict the CHF behavior of Modified 17x17 Vantage 5H fuel with or without modified intermediate flow mixer (MIFM) grids (with MIFM grids the fuel is referred to as 17x17 RFA fuel).

3.2 Critical Heat Flux Test Program

Data for development of WRB-2M was obtained at the Columbia University Heat Transfer Research Facility. This facility consisted of an instrumented high pressure loop that could supply water at pressures up to 2500 psia, flow rates up to 650 (gallons per minute), and inlet temperatures up to 650 °F. The power supply was capable of producing 12.5 megawatts of direct current.

Four types of test sections were analyzed to evaluate the combinations of different grid spacing and the effects of a central control rod guide thimble. The test sections were designed to have a chopped cosine axial power distribution and a non-uniform radial power distribution so that the highest power rods and peak heat flux locations were in the middle of the bundle and were prototypical of modified 17x17 Vantage 5H and 5H/IFM fuel.

CHF tests were performed by maintaining a constant test section outlet pressure, inlet temperature, and mass flux. Total power to the test section was then increased in small increments until a sudden temperature increase occurred in one or more of the thermocouples positioned on the heater rods. This temperature excursion indicated that DNB had occurred. When the temperature excursion occurred, power to the test section was reduced and preparation for the next test was begun.

3.3 Use of WRB-2M CHF Correlation with VIPRE-D

The WRB-2M CHF correlation is based on local conditions within the fuel bundles. The WRB-2M CHF correlation has been previously reviewed and approved by the NRC staff (Reference 2) and no further review is intended in this evaluation.

To evaluate local conditions within the Modified Vantage 5H test bundles, Dominion used the VIPRE-D code. VIPRE-D is a modified version of the Electric Power Research Institute's VIPRE-01 computer code. The VIPRE-D computer code has already been previously and approved by the NRC staff (Reference 3) and no further review is intended in this evaluation.

The NRC staff focused their review efforts on verifying that the WRB-2M correlation when used in the VIPRE-D computer code provided a conservative predicted CHF value with no bias or trends in the prediction of CHF.

The NRC staff reviewed the intended range of the correlation (Table 1) and finds that the behavior of the correlation is consistent over that range. The NRC staff reviewed the trend analysis and finds that there are no trends in the WRB-2M CHF correlation prediction of CHF as a function of any of the thermal-hydraulic variables (pressure, mass flow, and quality), as consistent with the WRB-2M approved topical report.

3.4 Statistical Evaluation of DNBR

Part of the correlation procedure involves the reduction of the calculated CHF to account for non-uniform axial flux shapes. This is accomplished by the use of the Tong F-factor (Reference 4). This factor takes into account that the CHF is affected by a bubble that later separates the main stream in a fluid channel from the superheated liquid near the heated surface. The bubble later is affected by axial distribution of the upstream heating so that for the same total power input, a peaked heat flux location will have a lower CHF than if the axial heating rate had been uniform. Tong derived the F-factor from theoretical considerations with empirical constants that were determined from test data. Use of the F-factor permits development of CHF correlations that are independent of the axial flux shape. The F-factor must also be applied when correlations are used to predict CHF for nuclear reactor safety analysis.

WRB-2M CHF correlation was developed from test data taken from two types of grid structures, with and without the MIFM grids. Of the 241 CHF tests performed, 143 tests contained MIFMs. In the other 98 tests, the MIFMs were omitted. Similar to the original WRB-2M Topical Report, Dominion performed statistical analyses which determined both data sets were random samples from the same population and therefore could be correlated together. Both data sets were first determined to be random samples of normal distributions in accordance with Regulatory Guide 5.22, "Assessment of the Assumption of Normality (Employing Individual Observed Values)," (Reference 5). The statistical variances of the two populations of data taken from similar test sections were then compared using the F-distribution test. The F-test demonstrated that both data sets were from the same total population and could therefore be correlated together. The tests with and without MIFM grids are of the same population since the mixing vanes in the structural support grids were modified to be of similar shape and size to those in the MIFMs. The only effective difference between the test assemblies is the grid spacing and the presence of thimble tubes. Both of which are accounted for in the data correlation.

During plant operation, the ratio of heat fluxes between the CHF and the actual heat flux, which is the DNBR, provides a method for describing the safety margin to fuel damage. One component in this margin is the minimum DNBR limit for acceptance of reactor core thermal/hydraulic calculations. A separate DNBR limit is calculated for each CHF correlation based on the scatter in predicted test results. The NRC staff has accepted DNBR limits that ensure a 95 percent probability that CHF will not occur with a confidence of 95 percent for the hottest pins of the reactor core (Reference 6). A DNBR limit of 1.14 was derived for the VIPRE-D computer code using the WRB-2M CHF correlation to meet this criterion.

The DNBR limit which meets the 95/95 acceptance criterion was determined using Owen's one-sided tolerance limit method (Reference 7). The general equation for Owen's method is as follows:

$$95/95 \text{ DNBR limit} = \frac{1}{\overline{\frac{M}{P}} - K_{95/95} \cdot \sigma}$$

Where,

$\overline{\frac{M}{P}}$ is the test population mean of measured to predicted CHF ratios.

σ is the effective standard deviation of all the M/P data.

$K_{95/95}$ is a tolerance multiplier which provides the 95/95 probability/confidence limit. The constant $K_{95/95}$ is a function of the effective degrees of freedom in the test series.

Consistent with previous CHF correlations, the standard deviation, σ, shall be calculated from combining the variance within the test series and the variance among the test series. The effective degrees of freedom shall also be calculated in a similar manner.

Considering the evaluation above, the NRC staff finds that with a DNBR limit of 1.14, the VIPRE-D computer code using the WRB-2M CHF correlation will conservatively predict the CHF behavior of the fuel designs described herein.

3.5 Conditions and Limitations

Based on the forgoing considerations, the NRC staff concludes that the use of the of VIPRE-D computer code with the WRB-2M CHF correlation with a DNBR limit of 1.14 is acceptable for plant safety analyses provided that the following conditions are met:

1. Because WRB-2M CHF correlation was developed from test assemblies designed to simulate Modified 17x17 Vantage 5H fuel with or without modified intermediate flow mixer grids, the correlation may only be used to perform evaluations for fuel of that type without further justification.

2. The WRB-2M CHF correlation shall not be applied outside its range of applicability defined by the original WRB-2M topical report and repeated in Table 1 of this evaluation.

3. The WRB-2M CHF correlation shall be used with a DNBR limit of 1.14 with the Dominion VIPRE-D computer code. WRB-2M is dependent on calculated local fluid properties that shall be only be calculated by a computer code approved by the NRC staff for that purpose, such as the VIPRE-D computer code.

4. The WRB-2M CHF correlation can be used for PWR plant analyses of steady state and reactor transients other than loss of coolant accidents. The WRB-2M CHF correlation shall not be used for loss of coolant accident analysis before additional justification is provided to the NRC staff which demonstrates that the applicable regulations are met and the computer code used to calculate local fuel element thermal/hydraulic properties has been approved for that purpose.

The NRC staff will require licensees referencing this topical report in licensing applications to document how these conditions are met.

4.0 CONCLUSION

When implemented as stated, the NRC staff has reasonable assurance that the use of the WRB-2M CHF correlation with Dominion's VIPRE-D computer code, as documented in Reference 1, is acceptable in calculating the CHF for the specified fuel types. The NRC staff has reviewed the qualification of the WRB-2M CHF correlation in the VIPRE-D computer code, and finds the method applicable only when implemented in accordance with the conditions and limitations described in Section 3.5 of this safety evaluation. The NRC staff does not intend to review the associated topical report when referenced in license applications.

If the NRC's criteria or regulations change so that its conclusions about the acceptability of the thermal-hydraulic methods or statistical analyses are invalidated, the licensee(s) referencing the report (Reference 1) will be expected to revise and resubmit its respective documentation, or submit justification for the continued effective applicability of the methodologies without revision of the respective documentation.

Table 1: WRB-2M Applicability Range

Parameter	WNG-1 Applicability Range
Pressure (psia)	1405 to 2425
Local mass velocity (Mlbm/hr-ft^2)	0.97 to 3.1
Local quality (fraction)	-0.1 to 0.29
Heat length (ft)	≤14
Grid spacing (inches)	10 to 20.6
Equivalent hydraulic diameter (inches)	0.37 to 0.46
Equivalent heated diameter (inches)	0.46 to 0.54

5.0 REFERENCES

1. Letter from Gerald T. Bischoff to U.S. NRC, Serial Number 08-0174 "Request for Approval of Appendix C of Fleet Report DOM-NAF-2 Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code" dated April 4, 2008 (ADAMS Accession No. ML080980229 (Publically Available)).

2. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids", April 1999 (ADAMS Accession No. ML081610106 (Non-Publically Available)).

3. Letter from Gerald T. Bischoff to U.S. NRC, Serial Number 06-773 "Approved Topical Report DOM-NAF-2, Rev 0.0-A Reactor Core Thermal-Hydraulics using the VIPRE-D Computer Code Including Appendixes A and B" dated September 14, 2006 (ADAMS Accession No. ML062650184 (Publically Available)).

See footnote in Appendix C, page C-22, regarding typo in Table 1

4. L.S. Tong, "Boiling Crisis and Critical Heat Flux," Temporary Instruction Inspection Documentation-25687, 1972.

5. U.S. Atomic Energy Commission Regulatory Guide 5.22, "Assessment of the Assumption of Normality (Employing Individual Observed Values)," April 1974 (ADAMS Accession No. ML003739999 (Publically Available)).

6. U.S. Nuclear Regulatory Commission Standard Review Plan, Section 4.4, "Thermal and Hydraulic Design, Rev 2" NUREG-800, March 2007.

7. D.B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans", SC-R-607, Sandia Report, March 1963.

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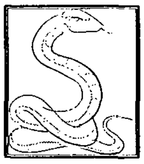
CLASSIFICATION/DISCLAIMER

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.

ACKNOWLEDGEMENTS

We would like to express our immense gratitude for the labors that Dr. Rosa M. Bilbao y León,



also recognized as Madame VIPRE, had in development and implementation of the VIPRE-D methodology at Dominion. She worked tirelessly to get every part of the *methodology to its current state today and lay the ground work for the future* development.

The original author [Dr. Rosa M. Bilbao y León] would like to express her huge appreciation to Mr. Kurt F. Flaig for his insight and considerable technical assistance in the development of the VIPRE-D methodology. In addition, she would like to thank Mr. Dana M. Knee, Ms. Jennifer L. Meszaros and Mr. Sean M. Blair for their significant contributions in the development of the methodology documented herein. The author would also like to acknowledge all the members of the Dominion Nuclear Safety Analysis group for their help and support through the long process involved in the production of this Fleet Report. Finally, the advice provided by Mr. Noval A. Smith, Dr. Ross C. Anderson, Mr. Kerry L. Basehore, Mr. David A. Farnsworth and Mr. John H. Jones (from AREVA), and Mr. Yixing Sung (from Westinghouse) was very appreciated.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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 $1/8^{th}$ 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	$\dfrac{\sum_{i=1}^{i=14} N_i - \sum_{i=1}^{i=13} F_{\max\Delta H} \cdot f_i \cdot N_i}{N_{14}}$	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).

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

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: $$A = 1.52 \cdot \left[\frac{Subchannel\,Pitch}{FuelRodOD} - 1\right]^{-0.5} \qquad [4.6.3]$$

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 $F\Delta H$.

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

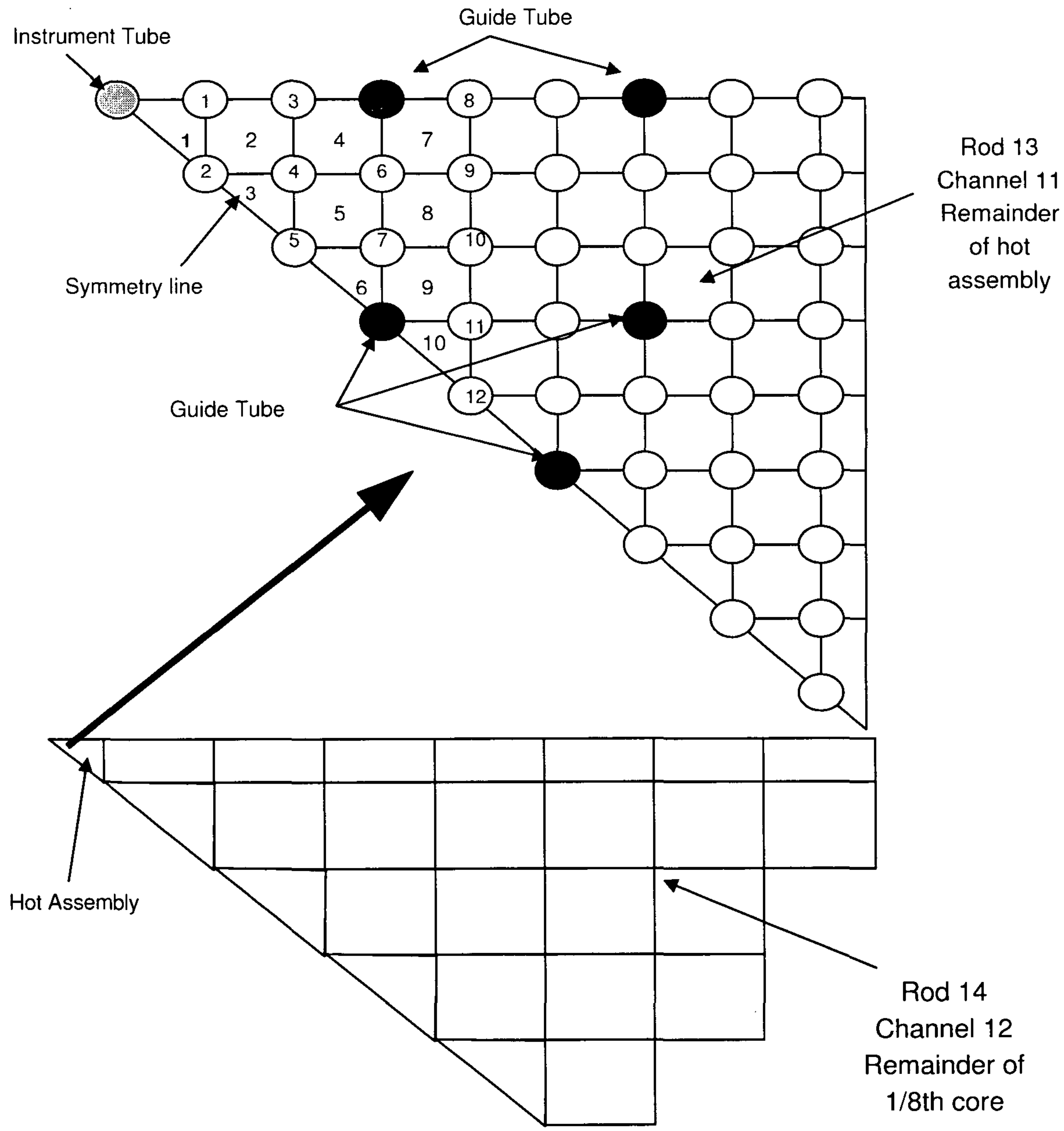
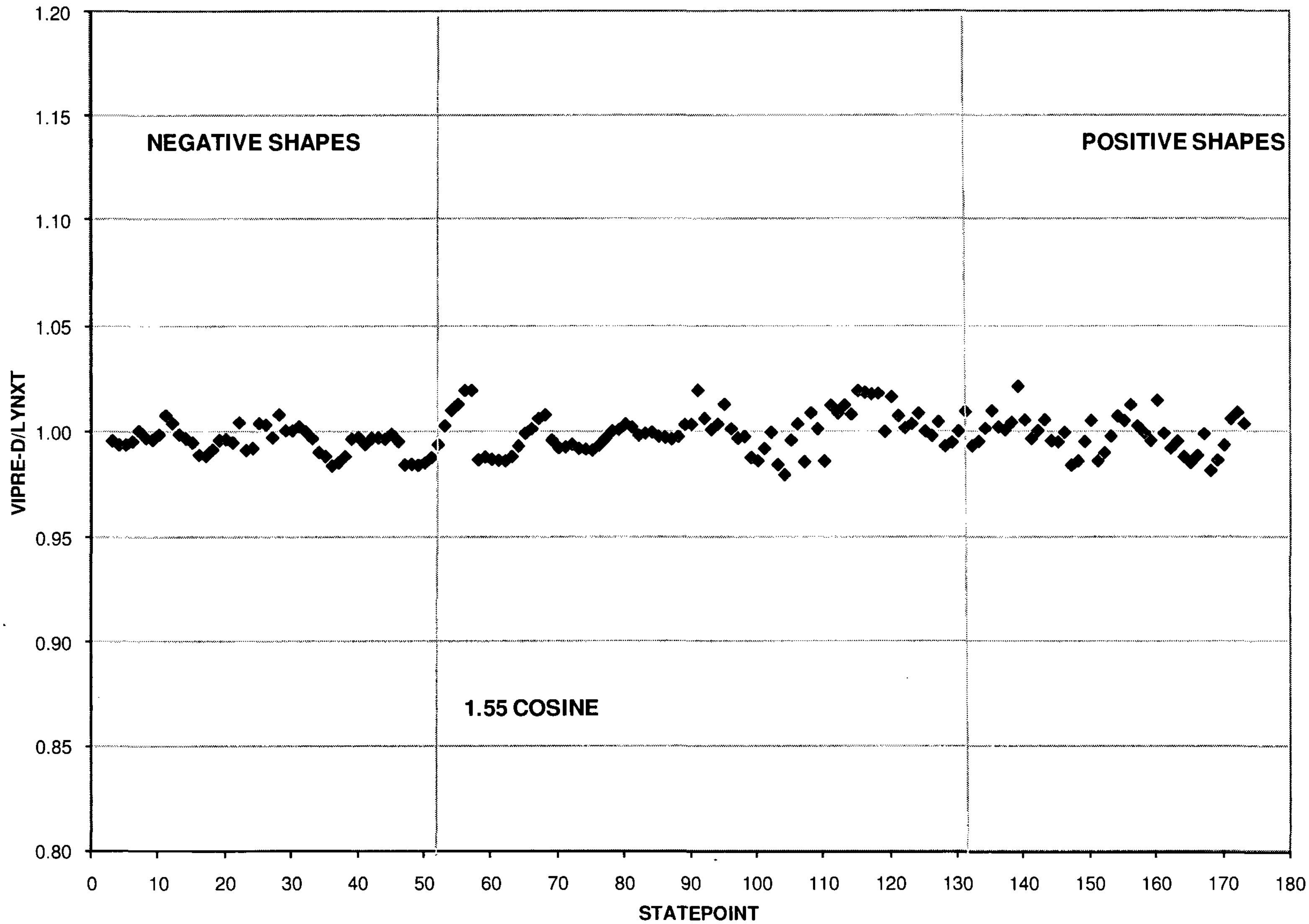


Figure 5.1-2: VIPRE-D vs. LYNXT for the 173 Statepoints



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.

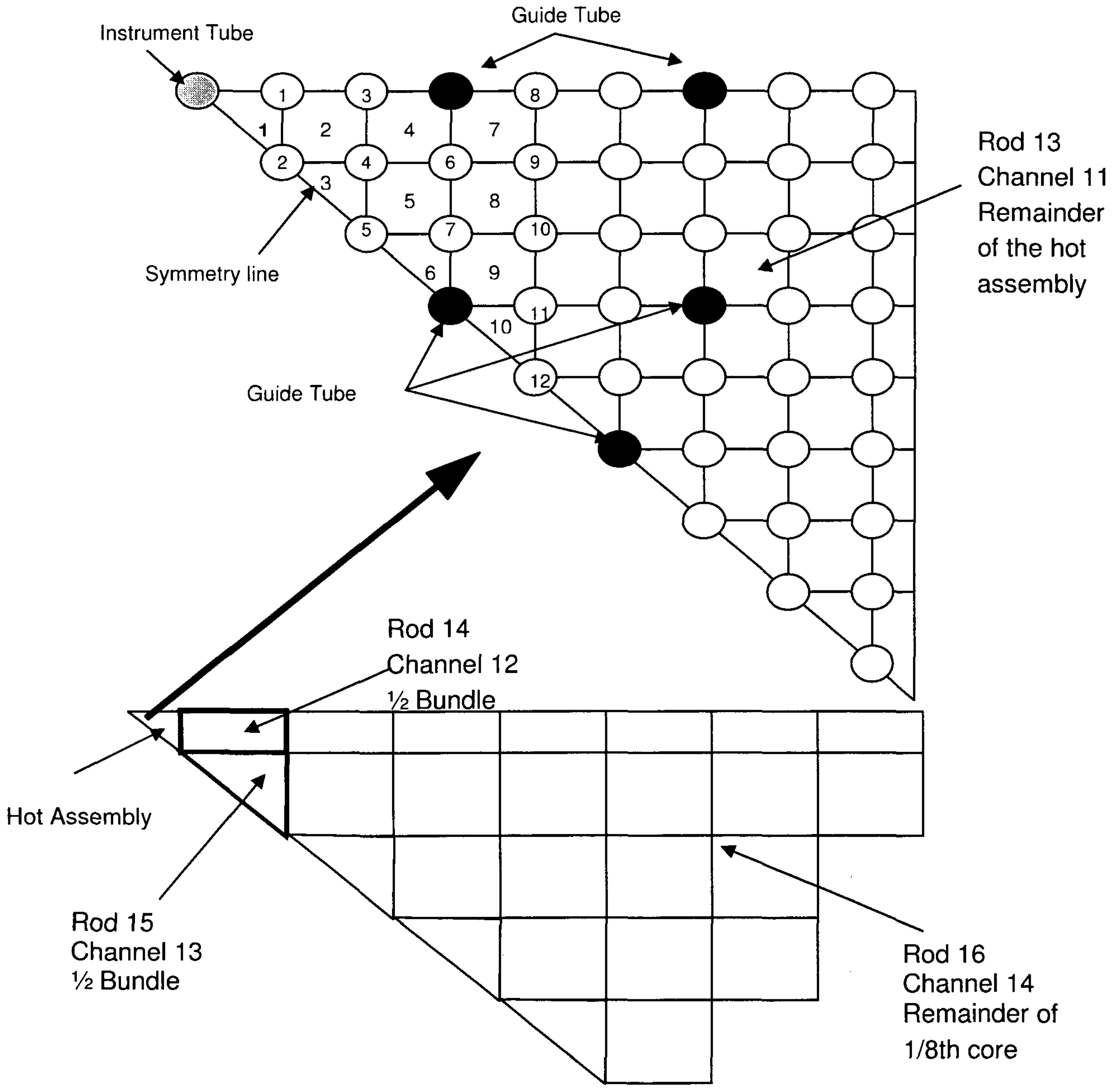
The VIPRE-D 14-channel model for a North Anna core containing F-ANP AMBW fuel assemblies consists of 14 channels (10 subchannels and 4 lumped channels) and 16 rods as shown in Figure 5.2-1. The two additional channels provide adequate solution detail of the flow field in the vicinity of the hot assembly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

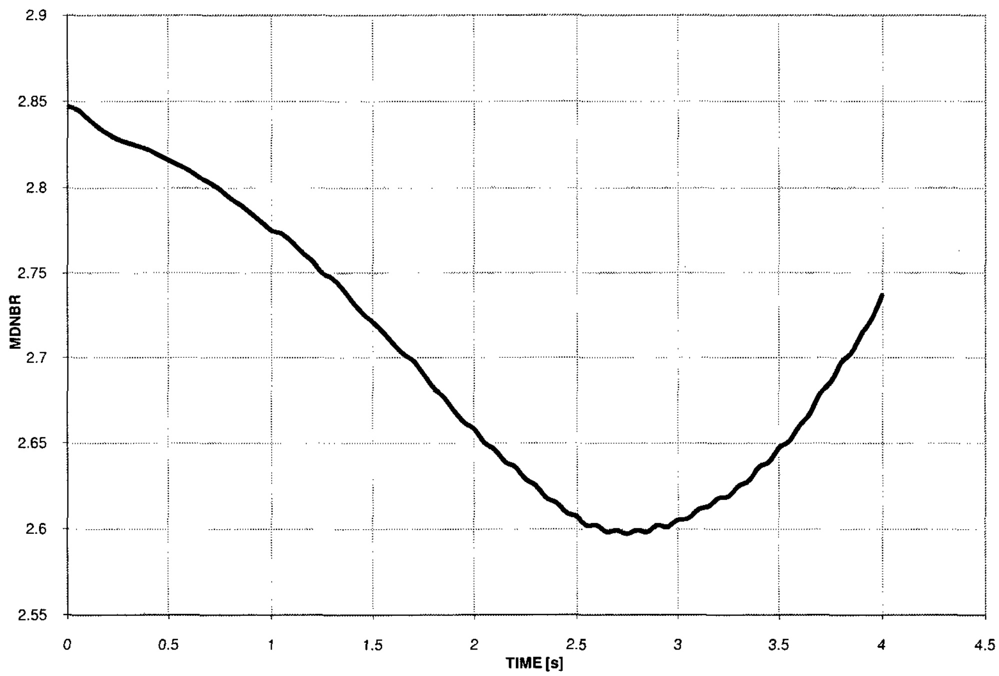
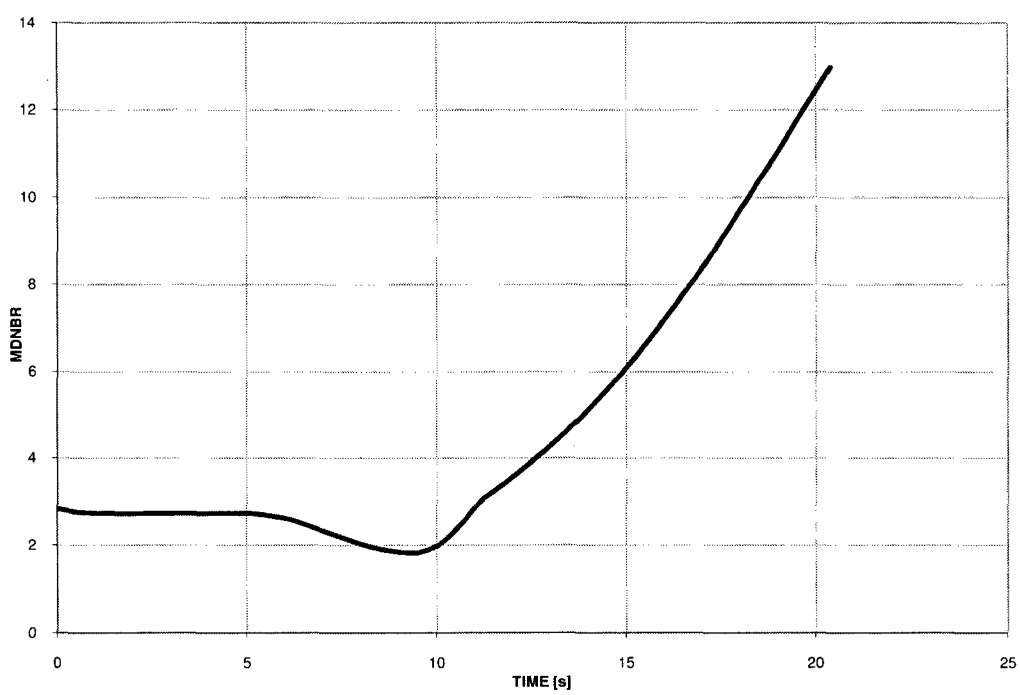


Figure 5.3-2: VIPRE-D LOFA Transient Sample Calculation Results



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

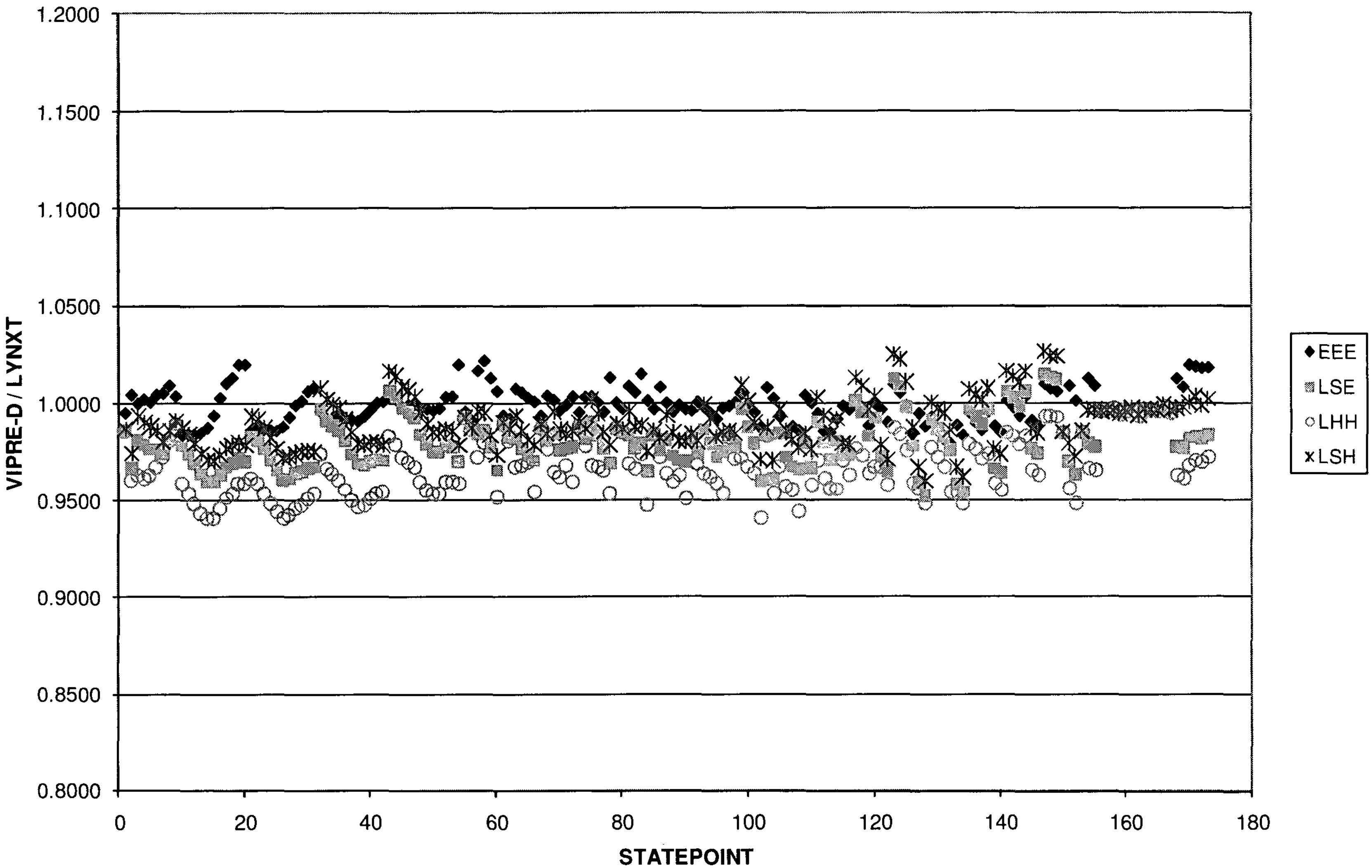
- Case 4 (LSH)
  Subcooled Void Model: LEVY
  Bulk Boiling Void Model: SMITH
  Two-Phase Friction Multiplier: HOMOGENEOUS

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 $\frac{\text{LYNXT} - \text{VIPRE}}{\text{LYNXT}}$			
	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.

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Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code

NUCLEAR ANALYSIS AND FUEL DEPARTMENT
DOMINION
RICHMOND, VIRGINIA

Approved by NRC Safety Evaluation dated April 4, 2006 and corrected June 23, 2006

CLASSIFICATION/DISCLAIMER

ABSTRACT

This appendix documents Dominion's qualification of the Framatome-ANP (F-ANP) BWU-N, BWU-Z and BWU-ZM correlations with the VIPRE-D code. This qualification was performed against the same CHF experimental database used by F-ANP to develop and license the correlations. This appendix summarizes the data evaluations that were performed to qualify the VIPRE-D/BWU code/correlation pair, and to develop the corresponding DNBR design limits for each correlation.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

ARC	Alliance Research Center
CHF	Critical Heat Flux
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
F-ANP	Framatome Advanced Nuclear Power
HTRF	Heat Transfer Research Facility at Columbia University
M/P	Ratio of Measured-to-Predicted CHF
MSMG	Mid-Span Mixing Grid
MVG	Mixing Vane Grid
NMVG	Non-Mixing Vane Grid
P/M	Ratio of Predicted-to-Measured CHF (equivalent to DNBR)
PWR	Pressurized Water Reactor
USNRC	US Nuclear Regulatory Commission

A.1 PURPOSE

Dominion has purchased fuel assemblies from Framatome ANP (F-ANP) for use at North Anna Power Station, Units 1 and 2. These new fuel assemblies are designated as Advanced Mark-BW fuel and are a one-for-one replacement for the resident fuel product, which is the North Anna Improved Fuel with ZIRLO components and PERFORMANCE+ debris resistant features (a Westinghouse fuel product). The thermal-hydraulic analysis of the F-ANP fuel product requires the use of the F-ANP BWU CHF correlations (References A1 and A2).

To be licensed for use, a critical heat flux (CHF) correlation must be tested against experimental data that span the anticipated range of conditions over which the correlation will be applied. Furthermore, the population statistics of the database must be used to establish a departure from nucleate boiling ratio (DNBR) design limit such that the probability of avoiding departure from nucleate boiling (DNB) will be at least 95% at a 95% confidence level.

This appendix documents Dominion's qualification of the BWU-N, BWU-Z and BWU-ZM correlations with the VIPRE-D code. This qualification was performed against the same CHF experimental database used by F-ANP to develop and license the correlations. This appendix summarizes the data evaluations that were performed to qualify the VIPRE-D/BWU code/correlation pair, and to develop the corresponding DNBR design limits for each correlation.

A.2 DESCRIPTION OF THE F-ANP CHF CORRELATIONS

In pressurized water reactor (PWR) cores, the energy generated inside the fuel pellets leaves the fuel rods at their surface in the form of heat flux, which is removed by the reactor coolant system flow. The normal heat transfer regime in this configuration is nucleate boiling, which is very efficient. However, as the capacity of the coolant to accept heat from the fuel rod surface degrades, a continuous layer of steam (a film) starts to blanket the tube. This heat transfer regime, termed film boiling, is less efficient than nucleate boiling and can result in significant increases of the fuel rod temperature for the same heat flux. Since the increase in temperature may lead to the failure of the fuel rod cladding, PWRs are designed to operate in the nucleate boiling regime and protection against operation in film boiling must be provided.

The heat flux at which the steam film starts to form is called CHF or the point of DNB. For design purposes, the DNBR is used as an indicator of the margin to DNB. The DNBR is the ratio of the predicted CHF to the actual local heat flux under a given set of conditions.

Thus, DNBR is a measure of the thermal margin to film boiling and its associated high temperatures. The greater the DNBR value (above 1.0), the greater the thermal margin.

The CHF cannot be predicted from first principles, so it is empirically correlated as a function of the local thermal-hydraulic conditions, the geometry, and the power distribution measured in the experiments. Since a CHF correlation is an analytical fit to experimental data, it has an associated uncertainty, which is quantified in a DNBR design limit. A calculated DNBR value greater than this design limit provides assurance that there is at least a 95% probability at the 95% confidence level that a departure from nucleate boiling will not occur.

F-ANP has developed and uses the B&W-2, the BWC and the BWCMV CHF correlations. The first two of these correlations apply to fuel assemblies with non-mixing vane spacer grids of inconel or zircaloy. The BWCMV correlation applies to fuel assemblies with mixing vane grids. These correlations are limited to applications in a high flow regime, but modern applications require the use of a correlation in the middle and low flow regimes. Using the response surface model and sequential optimization techniques, F-ANP developed a universal local conditions CHF correlation form. This correlation form, designated BWU, was modified and applied to three different fuel design types over the wider required ranges in Reference A1. This reference describes the CHF tests that provided the bases for the new correlations, analyzes the performance of the correlation for each fuel type, and provides limits and guidelines for its application.

The F-ANP BWU CHF correlations are defined in Reference A1 as:

$$Q_{CHF} = \frac{F_{MSM} \cdot FLS \cdot Q_{unif}}{F_{Tong}} \qquad [A.2.1]$$

where Q_{CHF} is the critical heat flux in Btu/hr-ft^2, F_{MSM} is a dimensionless performance factor dependent on the grid arrangement of the assembly and defined in References A1 and A2, FLS is a dimensionless length spacing factor, F_{Tong} is the dimensionless non-uniform flux shape factor (Tong factor) and Q_{unif} is the uniform heat flux in Btu/hr-ft^2. The specific formulations for each one of these components, as well as the corresponding constants are F-ANP proprietary and can be found in References A1 and A2.

References A1 and A2 discuss the application of the BWU correlation form to three different grid types:

- BWU-N, which is only applicable in the presence of non-mixing vane grids (NMVG).

- BWU-Z, which is the enhanced mixing vane correlation, is applicable to the DNB analysis of the fuel assembly in the mixing region.
- BWU-ZM, which is BWU-Z with a multiplicative enhancement factor, is applicable in the presence of mid-span mixing grids (MSMGs).

A.3 DESCRIPTION OF CHF EXPERIMENTAL TESTS

A.3.1 BWU-Z CORRELATION

F-ANP developed the BWU-Z correlation to be used for fuel designs with mixing spacer grids based on the experimental data obtained at the Heat Transfer Research Facility of Columbia University (HTRF) and with the Mark BW17 spacer grid designs. The HTRF is a ten-megawatt electric facility capable of testing full length (up to 14 ft heated length) rod arrays in up to a 6-by-6 matrix. HTRF testing conditions cover the full range of PWR operating conditions with pressures up to 2,500 psia, mass velocities up to 3.5 Mlbm/hr-ft^2 and inlet temperatures approaching saturation. Seven series of tests were used to develop the BWU-Z CHF correlation (References A1 and A4). These same tests were also used by Dominion to qualify the VIPRE-D/BWU-Z code correlation pair. Seven full assembly models were created for VIPRE-D to model these experimental test sections. Table A.3.1-1 summarizes the seven series of tests in the BWU-Z CHF experimental database.

Table A.3.1-1: BWU-Z CHF Experimental Database

TEST	TYPE	MATRIX	AXIAL HEAT FLUX SHAPE	PIN OD / GUIDE TUBE OD [inches]	HEATED LENGTH [inches]	GRID SPACING [inches]	NUMBER OF TESTS
BW 12.0	Unit Cell	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5	99
BW 13.1	Unit Cell	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5	94
BW 14.1	Guide Tube	5 x 5	1.55 Symmetric	0.374 / 0.482	143.4	20.5	76
BW 15.1	Cold Unit	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5	92
BW 16.0	Cold Row	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5	48
BW 19.0	Guide Tube	5 x 5	1.55 Symmetric	0.374 / 0.482	143.4	20.5	94
BW 20.0	Unit Cell	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5	48

A.3.2 BWU-ZM CORRELATION

F-ANP developed the BWU-ZM correlation to be used for fuel designs with MSMGs based on the experimental data obtained at the HTRF and with the Mark BW17 spacer grid designs. Three series of tests were used to validate the BWU-ZM CHF correlation (References A2 and A4). These same tests were also used by Dominion to qualify the VIPRE-D/BWU-ZM code correlation pair. Three full assembly models were created for VIPRE-D to model these experimental test sections. Table A.3.2-1 summarizes the three series of tests in the BWU-ZM CHF experimental database.

Table A.3.2-1: BWU-ZM CHF Experimental Database

TEST	TYPE	MATRIX	AXIAL HEAT FLUX SHAPE	PIN OD / GUIDE TUBE OD [inches]	HEATED LENGTH [inches]	GRID SPACING [inches]	NUMBER OF TESTS
BW 18.0	Unit Cell	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5 [mid-span grid]	18
BW 18.1	Unit Cell	5 x 5	1.55 Symmetric	0.374 / -	143.4	20.5 [mid-span grid]	58
BW 43.0	Guide Tube	5 x 5	1.55 Symmetric	0.374 / 0.482	143.4	20.5 [mid-span grid]	72

A.3.3 BWU-N CORRELATION

F-ANP developed the BWU-N correlation for fuel designs with NMVGs based on the experimental data obtained at the heat transfer facility at the Alliance Research Center (ARC) with the Mark C and Mark BZ non-mixing spacer grid designs. This experimental facility was similar in capacity to HTRF, but has since been decommissioned. Seven Mark C tests and 3 Mark BZ tests were used to develop the correlation (References A1 and A3). These same tests were also used by Dominion to qualify the VIPRE-D/BWU-N code correlation pair. Ten full assembly models were created for VIPRE-D to model these experimental test sections. Table A.3.3-1 summarizes the ten series of tests in the BWU-N CHF experimental database.

Table A.3.3-1: BWU-N CHF Experimental Database

TEST	TYPE	MATRIX	AXIAL HEAT FLUX SHAPE	PIN OD / GUIDE TUBE OD [inches]	HEATED LENGTH [inches]	GRID SPACING [inches]	NUMBER OF TESTS
C-3	Unit Cell	3 x 3[a]	1.0 Uniform	0.379 / -	72.0	21.0	107
C-6	Unit Cell	5 x 5	1.0 Uniform	0.3797 / -	144.0	21.0	130
C-7	Guide Tube	5 x 5	1.0 Uniform	0.379 / 0.465	144.0	21.0	122
C-8	Unit Cell	5 x 5	1.662 Cosine Symmetric	0.379 / -	144.0	b	155
C-9	Guide Tube	5 x 5	1.662 Cosine Symmetric	0.379 / 0.465	144.0	b	85
C-11	Unit Cell	5 x 5	1.595 Sine Symmetric	0.379 / -	144.0	b	34
C-12	Guide Tube	5 x 5	1.595 Sine Symmetric	0.379 / 0.465	144.0	b	133
B-15	Guide Tube	5 x 5	1.68 Cosine Symmetric	0.430 / 0.554	144.0	21.1	47
B-16	Unit Cell	5 x 5	1.68 Cosine Symmetric	0.430 / -	144.0	21.1	131
B-17	Intersection Cell	5 x 5	1.68 Cosine Symmetric	0.430 / -	144.0	21.1	157

[a] Bundle C-3 has a heated strip in each of the four walls (1.381" x 72.0").
[b] Grid centerline distances from the end of the heated length are 15.66", 37.66", 59.41", 80.91", 102.16", 123.16", 143.53".

A.4 VIPRE-D RESULTS AND COMPARISON TO LYNXT/LYNX2

References A3 and A4 describe the mathematical model for each separate test section by providing the bundle and cell geometry, the rod radial peaking values, the rod axial flux shapes, the types, axial locations and form losses associated to the spacer grids, as well as the thermocouple locations. References A1 and A2 provide the data for each CHF observation within a test, including power, flow, inlet temperature, pressure and CHF location (rod and axial location).

Each test section was modeled for analysis with the VIPRE-D thermal-hydraulic computer code as a full assembly model following the modeling methodology discussed in Section 4 in the main body of this report. For each set of bundle data, VIPRE-D produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc) at every axial node along the heated length of the test section. The ratio of measured-to-predicted CHF (M/P) is the variable that is normally used to evaluate the thermal-hydraulic performance of a code/correlation pair. The measured CHF is the local heat flux at a given location, while the predicted CHF is calculated by the code using the CHF correlation of interest (BWU-Z, BWU-ZM or BWU-N). The ratio of these two values provides the M/P ratio, which is the inverse of the DNB ratio. M/P ratios are frequently used to validate CHF correlations instead of DNB ratios, because their distribution is usually a normal distribution, which simplifies their manipulation and statistical analysis.

The axial location, the hot rod and the hot channel that are used to perform the M/P comparison are important. For each test, the M/P ratio must be evaluated at the axial location where burnout was observed experimentally, as listed in References A3 and A4. The axial nodalization for the various VIPRE-D models was developed taking into account the actual test location of the thermocouples, as well as the locations of the various spacer grids. The criteria used to select the hot channel and hot rod are supported by engineering judgment and use the information regarding burnout location provided by References A3 and A4. In general, when burnout was observed experimentally in a hot rod, a hot rod and a central (hot) channel were selected to perform the comparison. When the burnout was observed experimentally on a cold rod, a hot rod was still selected because it was considered unphysical to observe burnout in a cold rod earlier than in a hot rod (experimentally, even though a cold rod was reported to experience burnout first, the reality was that several rods saw burnout almost simultaneously and the limitations of the instrumentation and a desire to minimize damage to the test cell, caused the discrepancy). In this case, however, an external channel (cold) was selected to be the hot channel.

In addition to comparing to the experimental results, the results obtained by VIPRE-D when modeling the Mark BW, Mark C and Mark BZ experiments were benchmarked against the results obtained by F-ANP with the LYNXT/LYNX2 codes (References A1 and A2). This comparison was just a sanity check to verify that there are no suspect datapoints and that the statepoint conditions were correctly input to the code.

Some of the tests analyzed were discarded prior to their incorporation into the VIPRE-D/BWU database. Two criteria were used to justify data deletions.

1) If the M/P ratio obtained for a given data point was greater than 3.5 standard deviations from the average, the data point was eliminated. This criterion is consistent with the methodology used by F-ANP in Reference A1.
2) If any of the local conditions (pressure, mass velocity or thermodynamic quality) was outside the range of applicability of the correlation as given in References A1 and A2, the data point was eliminated. This criterion is also consistent with the methodology used by F-ANP in Reference A1.

Overall, 23 data points were excluded from the BWU-Z database (F-ANP discarded 21 data points in Reference A1), and 11 were excluded from the BWU-N database (F-ANP eliminated 8 data points in Reference A1). No data points were eliminated from the BWU-ZM database. The reason the VIPRE-D/BWU database is slightly smaller than the LYNXT/BWU database is that the local conditions predicted by VIPRE-D for a few test data were just barely outside the range of validity of the BWU correlations as given in Reference A1.

This section summarizes the VIPRE-D results and the associated significant statistics. In addition, this section shows a comparison to the results obtained by F-ANP with the LYNXT/LYNX2 codes as reported in References A1 and A2. This section also shows the variation of the M/P ratio with each independent variable to assess if there are any biases in the data. Finally, it provides the VIPRE-D overall statistics for the seven BWU-Z tests, the three BWU-ZM tests and the ten BWU-N tests, and generates the DNBR design limits for the various BWU CHF correlations with VIPRE-D.

A.4.1 VIPRE-D/BWU-Z RESULTS

The BWU-Z correlation was developed by F-ANP correlating the CHF experimental results obtained in tests BW 12.0, BW 13.1, BW 14.1, BW 15.1, BW 16.0, BW 19.0 and BW 20.0. Dominion used those same experimental data to develop the VIPRE-D/BWU-Z DNBR limit. Table A.4.1-1 summarizes the relevant statistics for each test, and calculates the aggregate statistics for the entire set of data.

One-sided tolerance theory (Reference A5) is used for the calculation of the VIPRE-D/BWU-Z DNBR design limit. This theory allows us to calculate a DNBR limit so that, for a DNBR equal to the design limit, DNB will be avoided with 95% probability at a 95% confidence level.

Table A.4.1-1: VIPRE-D/BWU-Z M/P Ratio Results

TEST	NUMBER OF TESTS	M/P RATIO AVERAGE	M/P RATIO STDEV	M/P RATIO MAX	M/P RATIO MIN
BW 12.0	99	1.0230	0.0848	1.1683	0.7812
BW 13.1	94	0.9907	0.0900	1.1609	0.7669
BW 14.1	76	0.9869	0.0951	1.1538	0.7261
BW 15.1	92	1.0086	0.0917	1.2974	0.7717
BW 16.0	48	0.9475	0.0716	1.0840	0.6980
BW 19.0	94	0.9833	0.0893	1.1693	0.7833
BW 20.0	25	1.0108	0.0971	1.1642	0.8342
BWU-Z	**528**	**0.9950**	**0.0907**	**1.2974**	**0.6980**

Because all the statistical techniques used below assume that the original data distribution is normal, it is necessary to verify that the overall distribution for the M/P ratios is a normal distribution. To evaluate if the distribution is normal, the D' normality test was applied (Reference A6). A value of D' equal to 3,430.23 was obtained for the VIPRE-D/BWU-Z database. This D' value is within the range of acceptability for 528 data points with a 95% confidence level (3,387.6 to 3,449.4)[c]. Thus, it is concluded that the M/P distribution for the VIPRE-D/BWU-Z database is indeed normal.

Based on the results listed in Table A.4.1-1, the deterministic DNBR design limit can be calculated as:

$$DNBR_L = \frac{1.0}{M/P - K_{N,C,P} \cdot \sigma_{M/P}} \qquad \text{[A.4.1.1]}$$

where

M/P = average measured-to-predicted CHF ratio

$\sigma_{M/P}$ = standard deviation of the measured-to-predicted CHF ratios of the database

$K_{N,C,P}$ = one-sided tolerance factor based on N degrees of freedom, C confidence level, and P portion of the population protected. This number is taken from Table 1.4.4 of Reference A5.

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[c] From Table 5 in Reference A6
D' Lower Limit (528) [P = 0.025] = 3,310 + (8 / 20) x (3,504 – 3,310) = 3,387.6
D' Upper Limit (528) [P = 0.975] = 3,371 + (8 / 20) x (3,567 – 3,371) = 3,449.4

Then, the DNBR design limit for the VIPRE-D and the BWU-Z correlation can be calculated as described in Table A.4.1-2:

Table A.4.1-2: VIPRE-D/BWU-Z DNBR Design Limit

			VIPRE-D/BWU-Z
Number of data	n		528
Degrees of freedom	N	= n - 1 - 14	513
Average M/P	M/P		0.9950
Standard Deviation	$\sigma_{M/P}$		0.0907
Corrected Standard Deviation	σ_N	$= \sigma_{M/P} \cdot [\,(n-1)/N\,]^{1/2}$	0.0919
Owen Factor	K(513,0.95,0.95)		1.7607
BWU-Z Design limit	$DNBR_L$	= 1 / (0.9950 – 1.7607 · 0.0919)	**1.2002**

Figures A.4.1-1 through A.4.1-4 display the performance of the M/P ratio, and its distributions as a function of the pressure, mass velocity and quality. The objective of these plots is to show that there are no biases in the M/P ratio distribution, and that the performance of the BWU-Z correlation is independent of the three variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes. These plots also show that all the tests in the BWU-Z database are within 3.5 standard deviations from the average. Figures A.4.1-5 through A.4.1-7 display the performance of the P/M ratio (i.e. the DNBR) against the major independent variables for the BWU-Z database. These plots also include a DNBR design limit line at 1.20. It can be seen that only 19 data points (3.6% of the database) are above the DNBR design limit, and that these data in excess of the limit are distributed over the variable ranges tested.

In Reference A1, the USNRC argued that the performance of the BWU-Z correlation might be deficient at the extremely low end of the pressure range. For that reason, F-ANP developed individual DNBR design limits for each low pressure group in the database. This approach allows users to use the BWU-Z correlation at low pressures but imposes a higher DNBR limit to ensure that the correlation is used conservatively. Table A.4.1-3 summarizes the VIPRE-D/BWU-Z DNBR limits calculated for the different pressure groups and compares them with the BWU-Z DNBR design limits obtained by F-ANP in Reference A1.

Table A.4.1-3: VIPRE-D/BWU-Z DNBR Limits for Pressure Groups

	400 psia	**700 psia**	**1000 psia**	**1500 – 2400 psia**
AVERAGE M/P	0.8504	1.0452	1.0623	0.9883
STDEV	0.0121	0.0879	0.0787	0.0883
# DATA	4	20	40	464
K(N,0.95,0.95)	6.882	2.396	2.125	1.768
VIPRE-D DNBR LIMIT	**1.304**	**1.198**	**1.117**	**1.202**
LYNXT DNBR LIMIT	*1.590*	*1.199*	*1.125*	*1.193*

Dominion will take the VIPRE-D/BWU-Z DNBR limit to be 1.20 for pressures greater than or equal to 700 psia, and 1.59 at pressures lower than 700 psia. Since the VIPRE-D/BWU-Z database at 400 psia only has four datapoints, Dominion has used the F-ANP more conservative DNBR limit of 1.59.

Figure A.4.1-1: Measured vs. Predicted CHF for BWU-Z

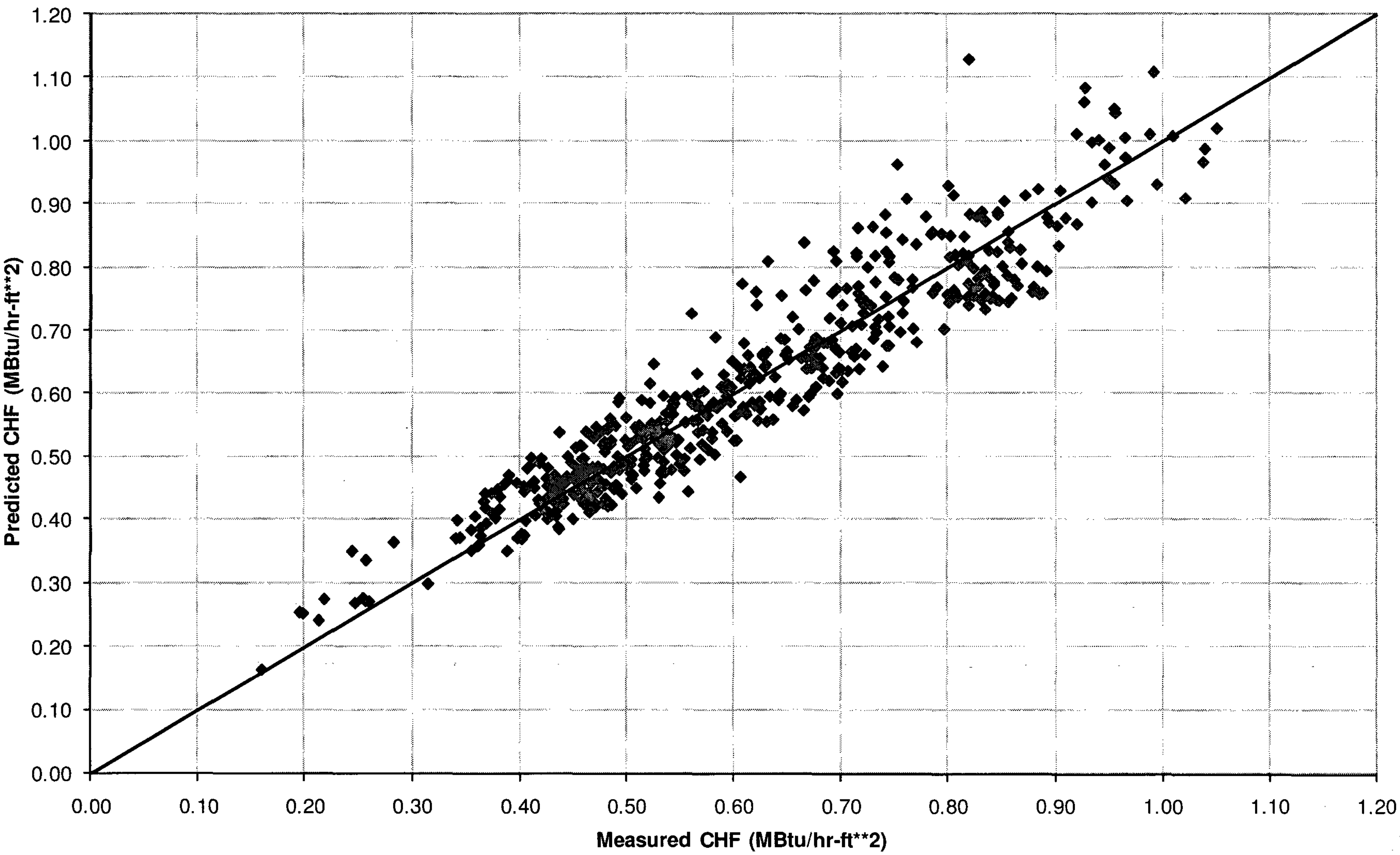


Figure A.4.1-2: M/P vs. Pressure for BWU-Z

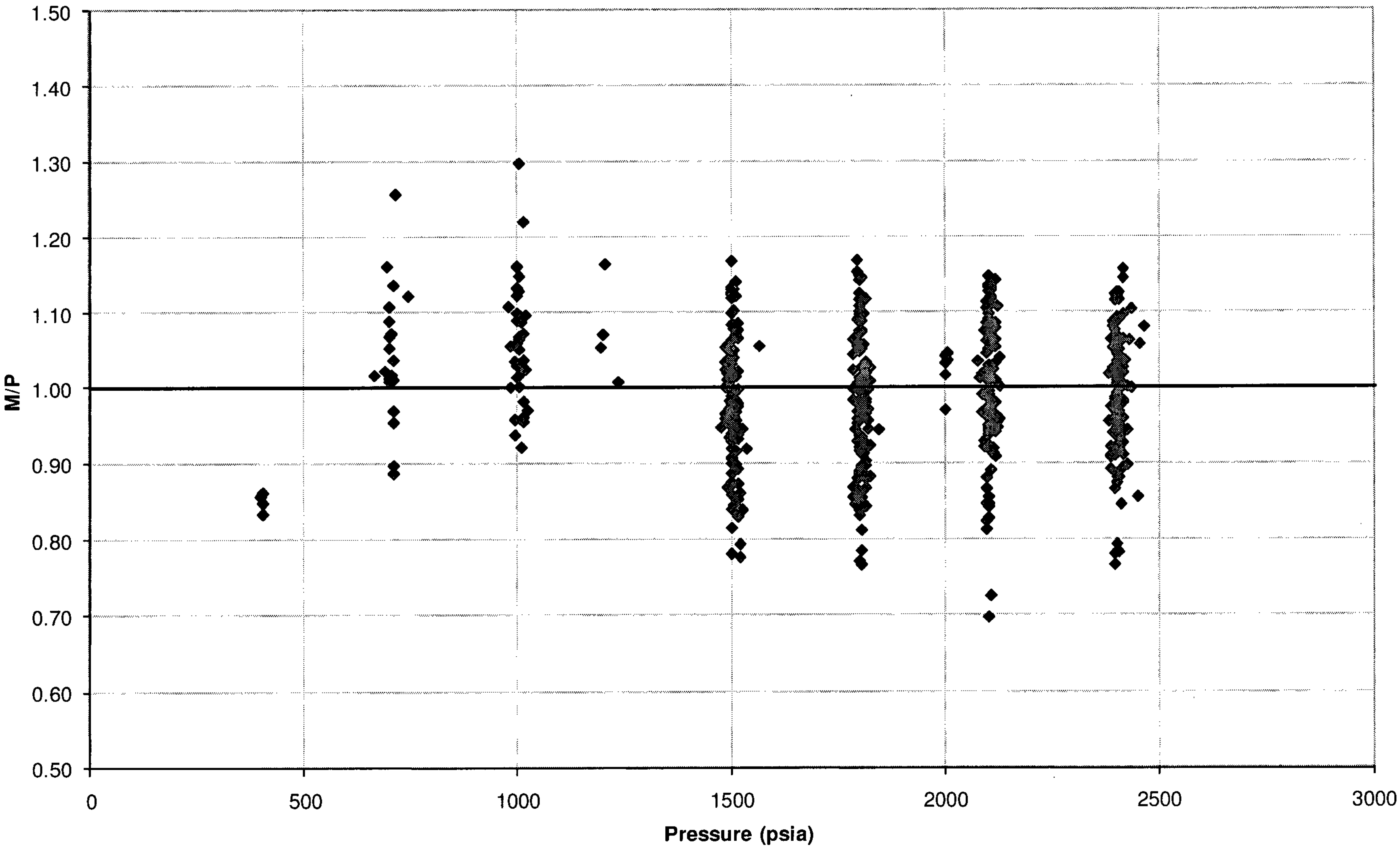


Figure A.4.1-3: M/P vs. Quality for BWU-Z

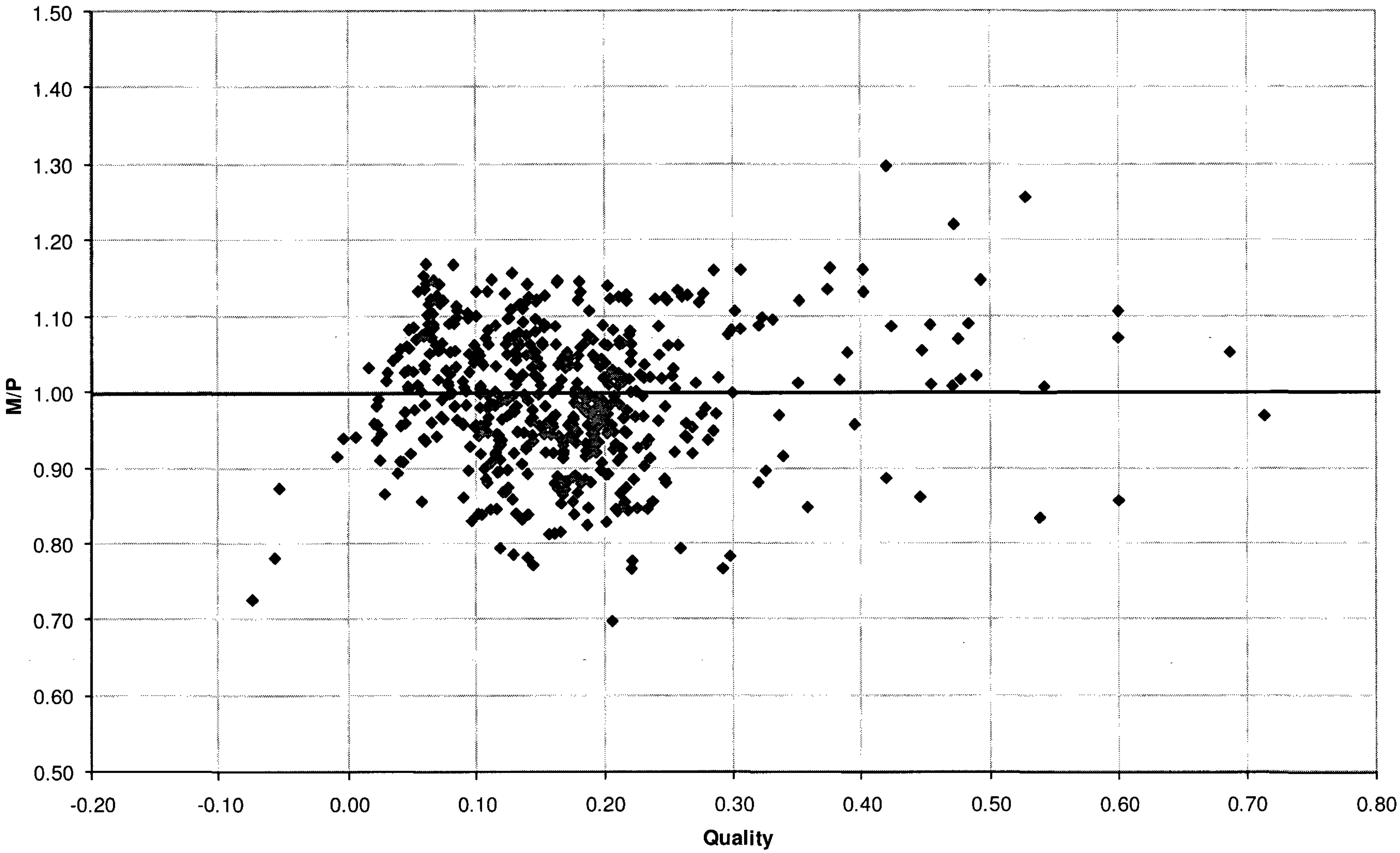


Figure A.4.1-4: M/P vs. Mass Velocity for BWU-Z

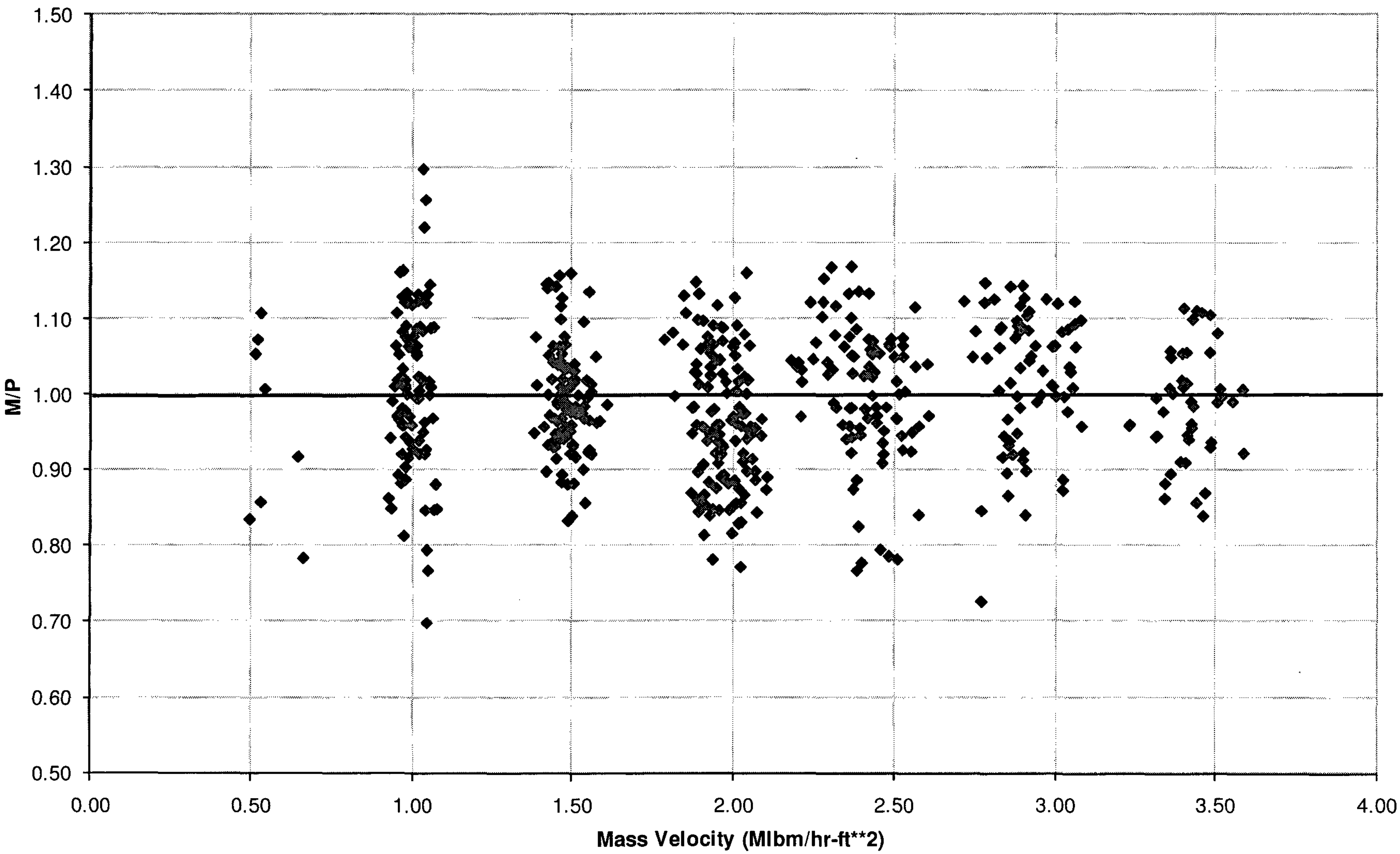


Figure A.4.1-5: DNBR vs. Pressure for BWU-Z

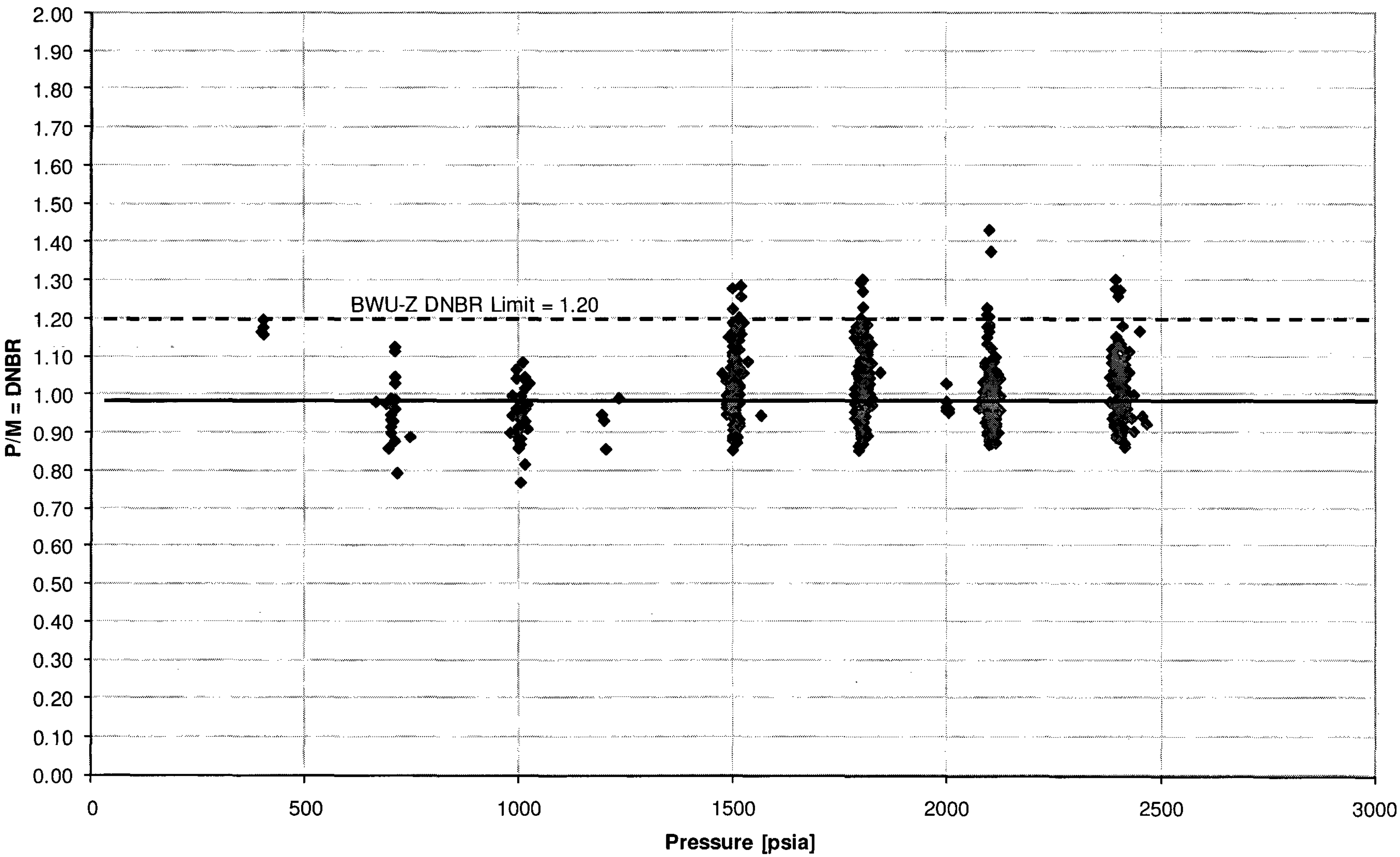


Figure A.4.1-6: DNBR vs. Quality for BWU-Z

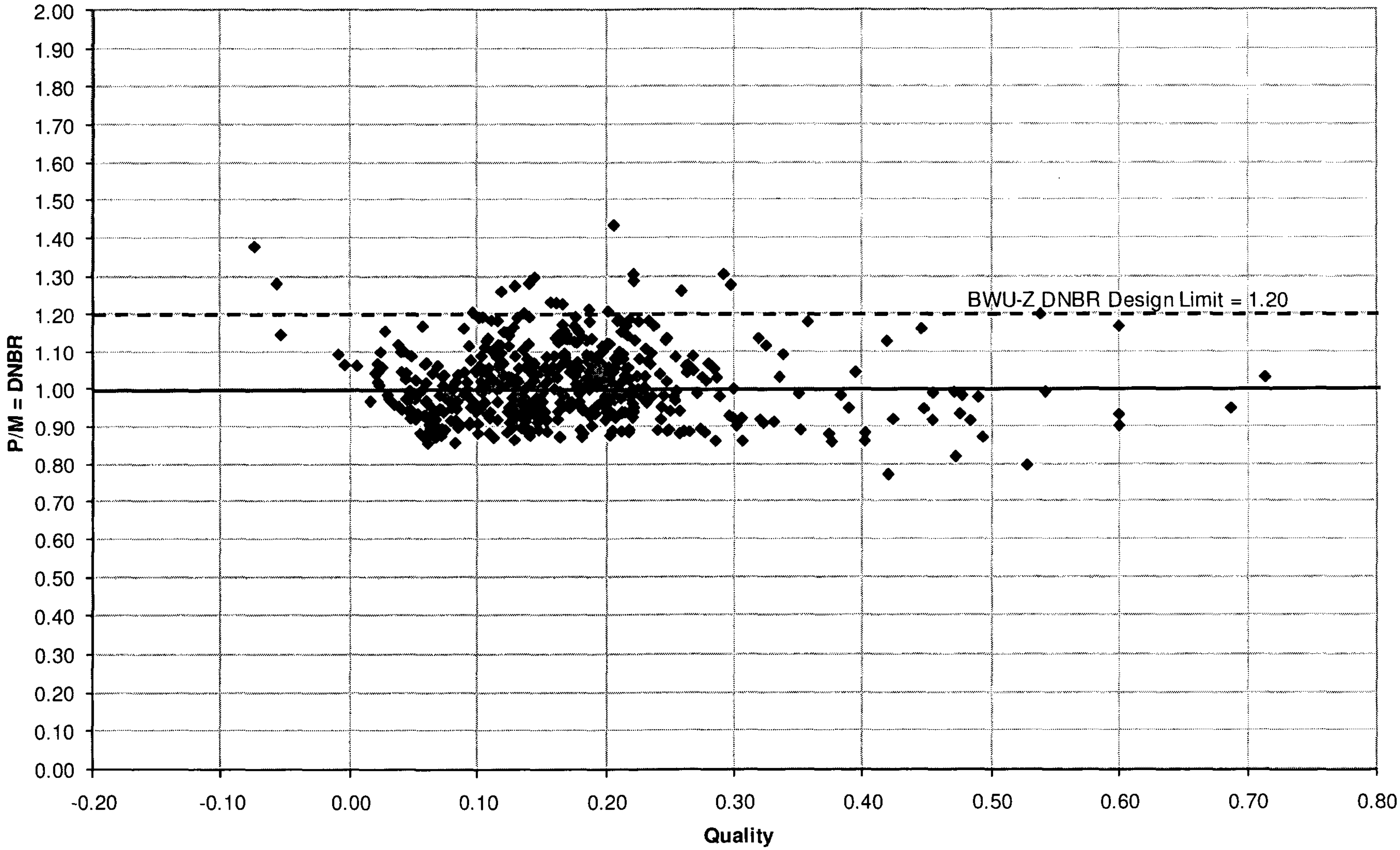
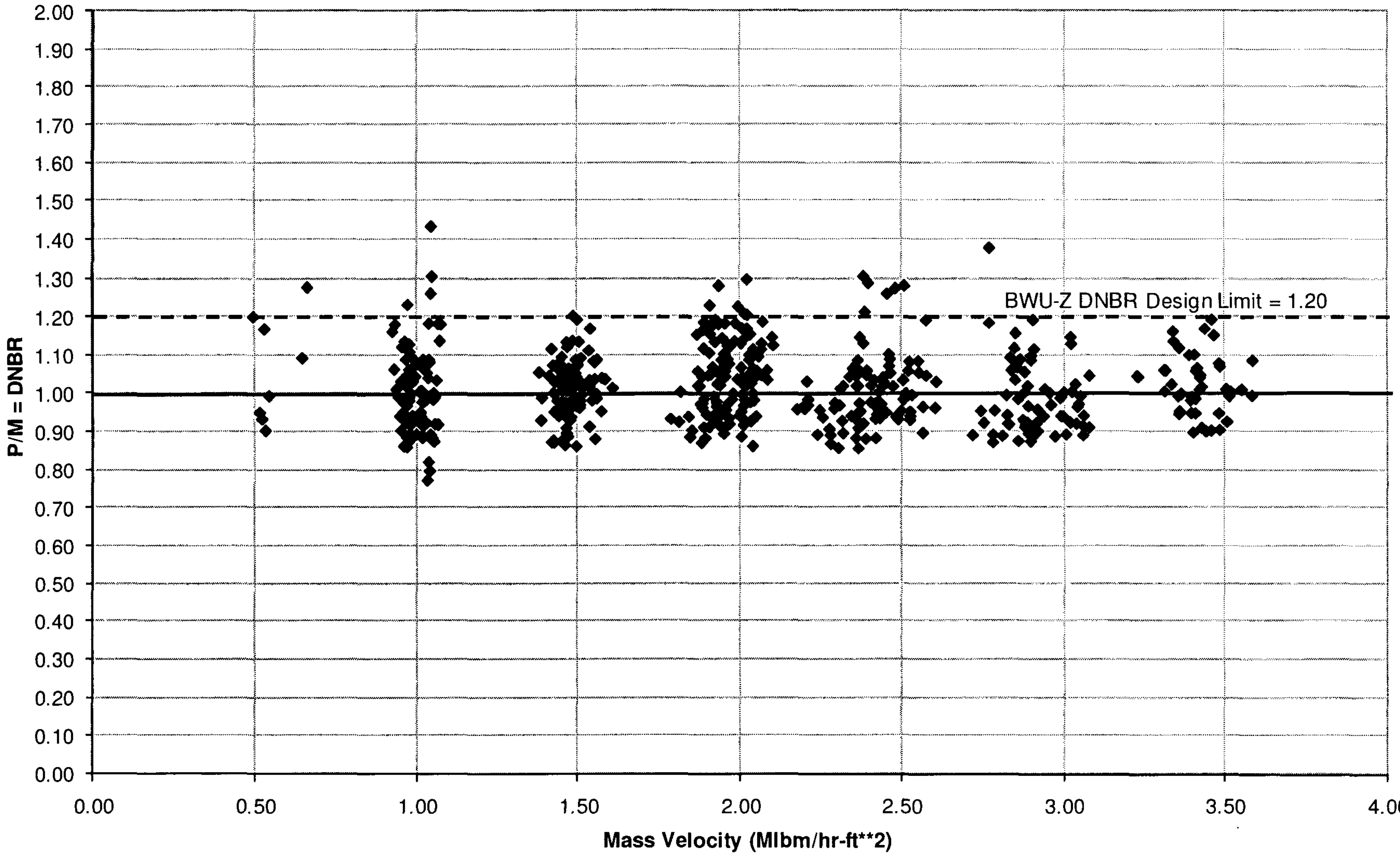


Figure A.4.1-7: DNBR vs. Mass Velocity for BWU-Z



A.4.2 VIPRE-D/BWU-ZM RESULTS

The BWU-ZM correlation was developed by F-ANP correlating the CHF experimental results obtained in tests BW 12.0, BW 13.1, BW 14.1, BW 15.1, BW 16.0, BW 19.0 and BW 20.0. F-ANP used the experimental data obtained in tests BW 18.0, BW 18.1 and BW 43.0 to determine F_{MSM} and to calculate the DNBR limit for the BWU-ZM correlation (Reference A2).

Dominion has used those same experimental data to determine the VIPRE-D/BWU-ZM DNBR limit. Table A.4.2-1 summarizes the relevant statistics for each test, and calculates the aggregate statistics for the entire set of data.

One-sided tolerance theory (Reference A5) is used for the calculation of the VIPRE-D/BWU-ZM DNBR design limit. This theory allows us to calculate a DNBR limit so that, for a DNBR equal to the design limit, DNB will be avoided with 95% probability at a 95% confidence level.

Table A.4.2-1: VIPRE-D/BWU-ZM M/P Ratio Results

TEST	NUMBER OF TESTS	M/P RATIO AVERAGE	M/P RATIO STDEV	M/P RATIO MAX	M/P RATIO MIN
BW 18.0	18	0.9931	0.1136	1.1467	0.8334
BW 18.1	58	1.0322	0.0945	1.2299	0.8142
BW 43.0	72	1.0041	0.0715	1.1747	0.7793
BWU-ZM	**148**	**1.0138**	**0.0875**	**1.2299**	**0.7793**

Because all the statistical techniques used below assume that the original data distribution is normal, it is necessary to verify that the overall distribution for the M/P ratios is a normal distribution. To evaluate if the distribution is normal, the D' normality test was applied (Reference A6). A value of D' equal to 510.55, was obtained for the VIPRE-D/BWU-ZM database. This D' value is within the range of acceptability for 148 data points with a 95% confidence level (497.82 to 515.04)[d]. Thus, it is concluded that the M/P distribution for BWU-ZM is indeed normal.

Based on the results listed in table A.4.2-1, the deterministic DNBR design limit can be calculated as:

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[d] From Table 5 in Reference A6

D' Lower Limit (148) [P = 0.025] = 456.9 + (8 / 20) x (559.2 – 456.9) = 497.82

D' Upper Limit (148) [P = 0.975] = 473.2 + (8 / 20) x (577.8 – 473.2) = 515.04

$$DNBR_L = \frac{1.0}{M/P - K_{N,C,P} \cdot \sigma_{M/P}} \quad \text{[A.4.2.1]}$$

where

M/P = average measured to predicted CHF ratio

$\sigma_{M/P}$ = standard deviation of the measured to predicted CHF ratios of the database

$K_{N,C,P}$ = one-sided tolerance factor based on N degrees of freedom, C confidence level, and P portion of the population protected. This number is taken from Table 1.4.3 of Reference A5.

Then, the DNBR design limit for the VIPRE-D and the BWU-ZM correlation can be calculated as described in Table A.4.2-2:

Table A.4.2-2: VIPRE-D/BWU-ZM DNBR Design Limit

			VIPRE-D/BWU-ZM
Number of data	n		148
Degrees of freedom	N	= n - 1	147
Average M/P	M/P		1.0138
Standard Deviation	$\sigma_{M/P}$		0.0875
Owen Factor	K(147,0.95,0.95)		1.872
BWU-ZM Design limit	$DNBR_L$	= 1 / (1.0138 – 1.872 · 0.0875)	**1.1765**

Figures A.4.2-1 through A.4.2-4 display the performance of the M/P ratio, and its distributions as a function of the pressure, mass velocity and quality. The objective of these plots is to show that there are no biases in the M/P ratio distribution, and that the performance of the BWU-ZM correlation is independent of the three variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes. Figures A.4.2-5 through A.4.2-7 display the performance of the P/M ratio (i.e. the DNBR) against the major independent variables for the BWU-ZM database. These plots also include a DNBR design limit line at 1.18. It can be seen that only 4 data points (2.7% of the database) are above the DNBR design limit, and that these data in excess of the limit are distributed over the variable ranges tested.

For the BWU-ZM database, no individual DNBR design limits were calculated for the low pressure data. However, in order to extend the validity of the BWU-ZM CHF correlation over the same range as the BWU-Z CHF correlation, the VIPRE-D/BWU-ZM DNBR design limit at pressures less than 594 psia was set to 1.59 (The same as for BWU-Z at low pressures). The DNBR design limit for VIPRE-D/BWU-ZM for pressures equal to or greater than 594 psia is 1.18.

Figure A.4.2-1: Measured vs. Predicted CHF for BWU-ZM

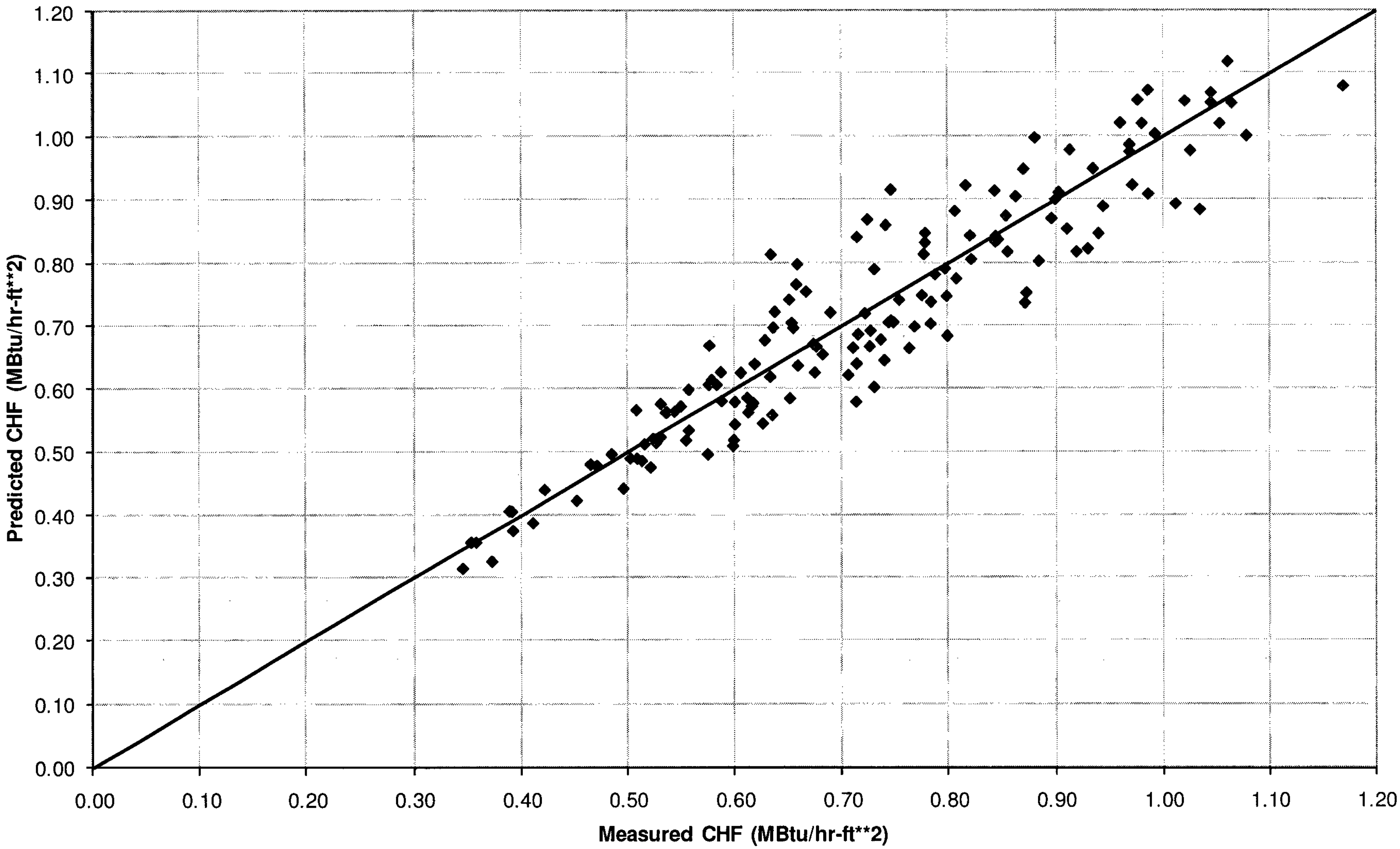


Figure A.4.2-2: M/P vs. Pressure for BWU-ZM

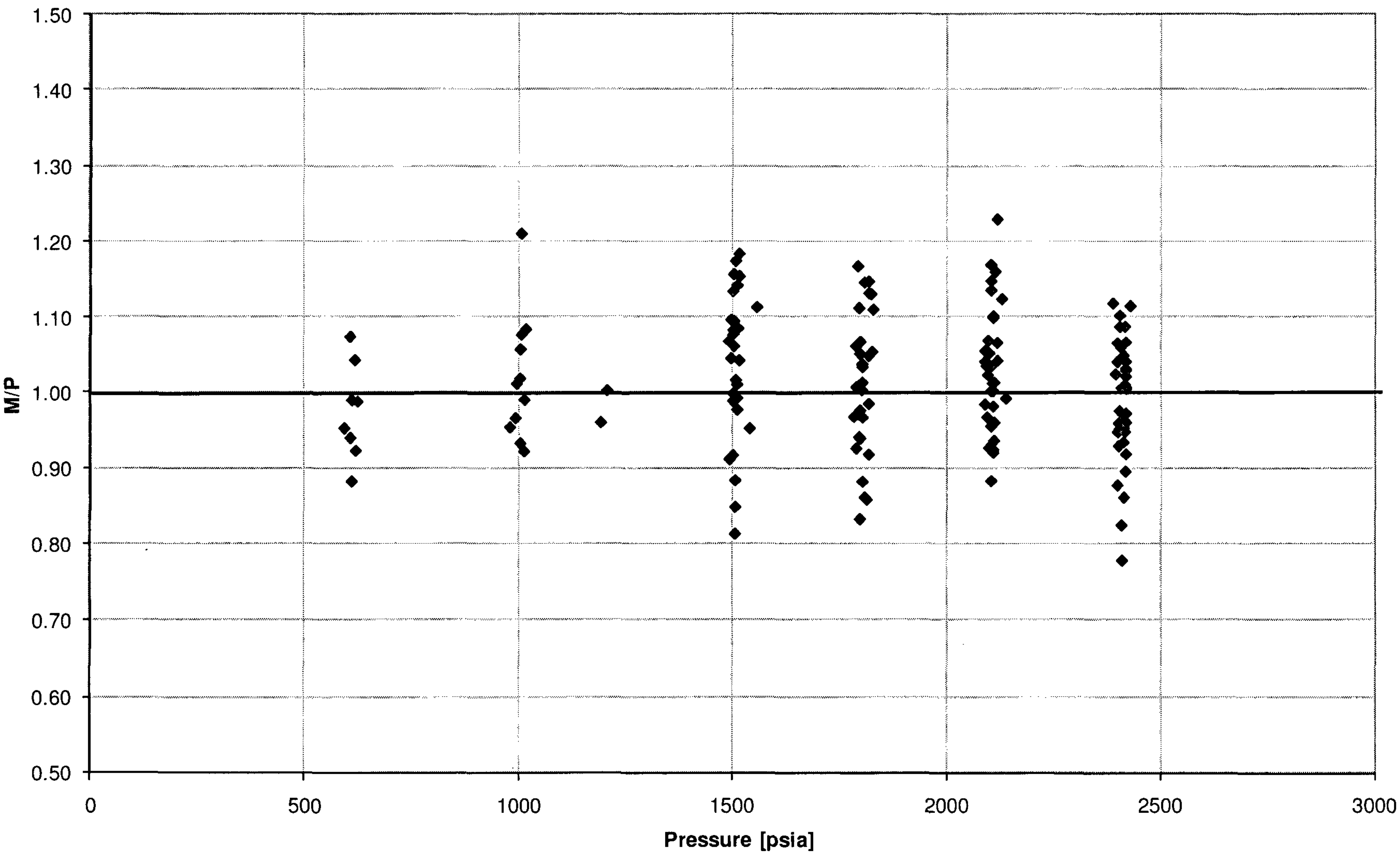


Figure A.4.2-3: M/P vs. Quality for BWU-ZM

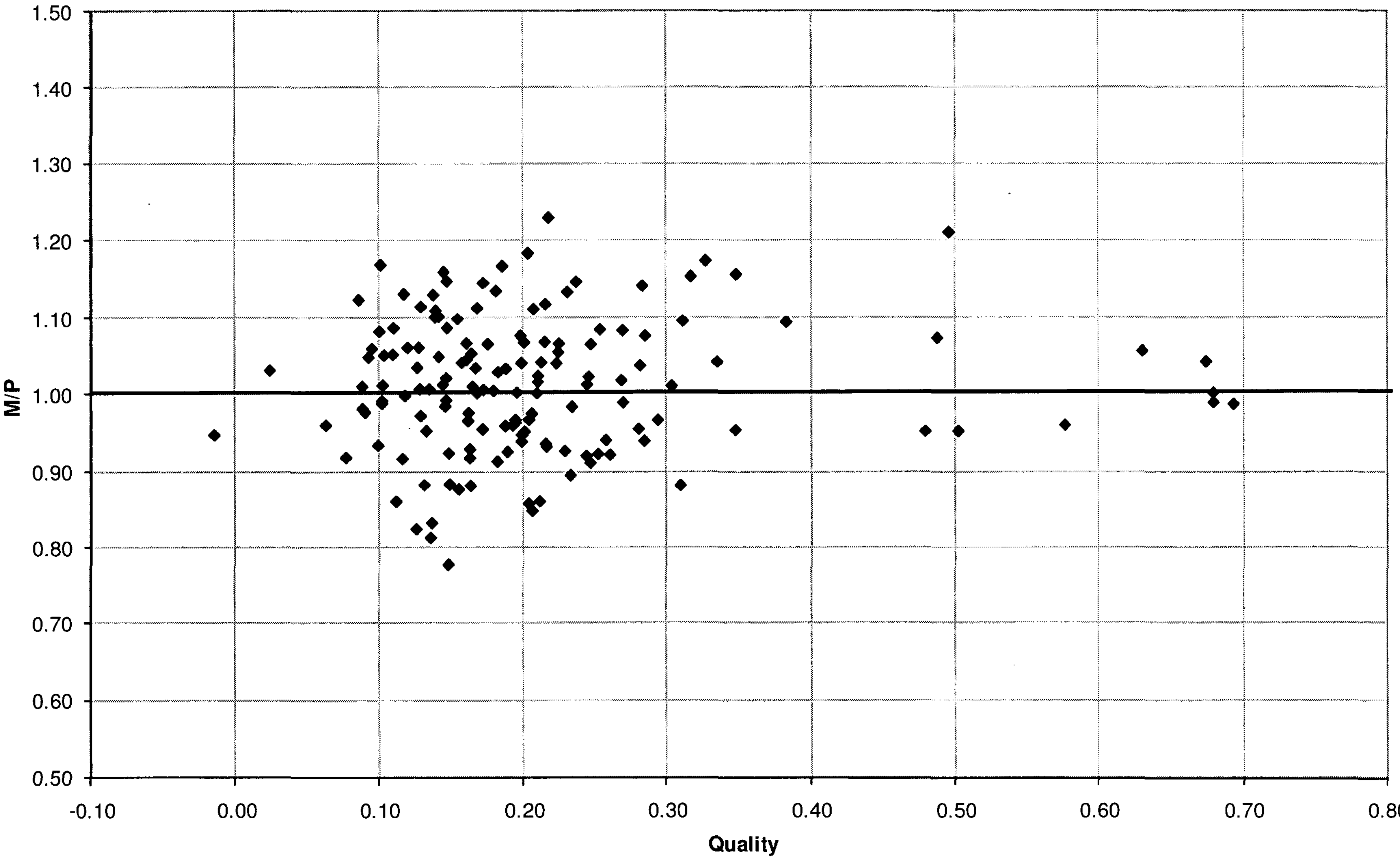


Figure A.4.2-4: M/P vs. Mass Velocity for BWU-ZM

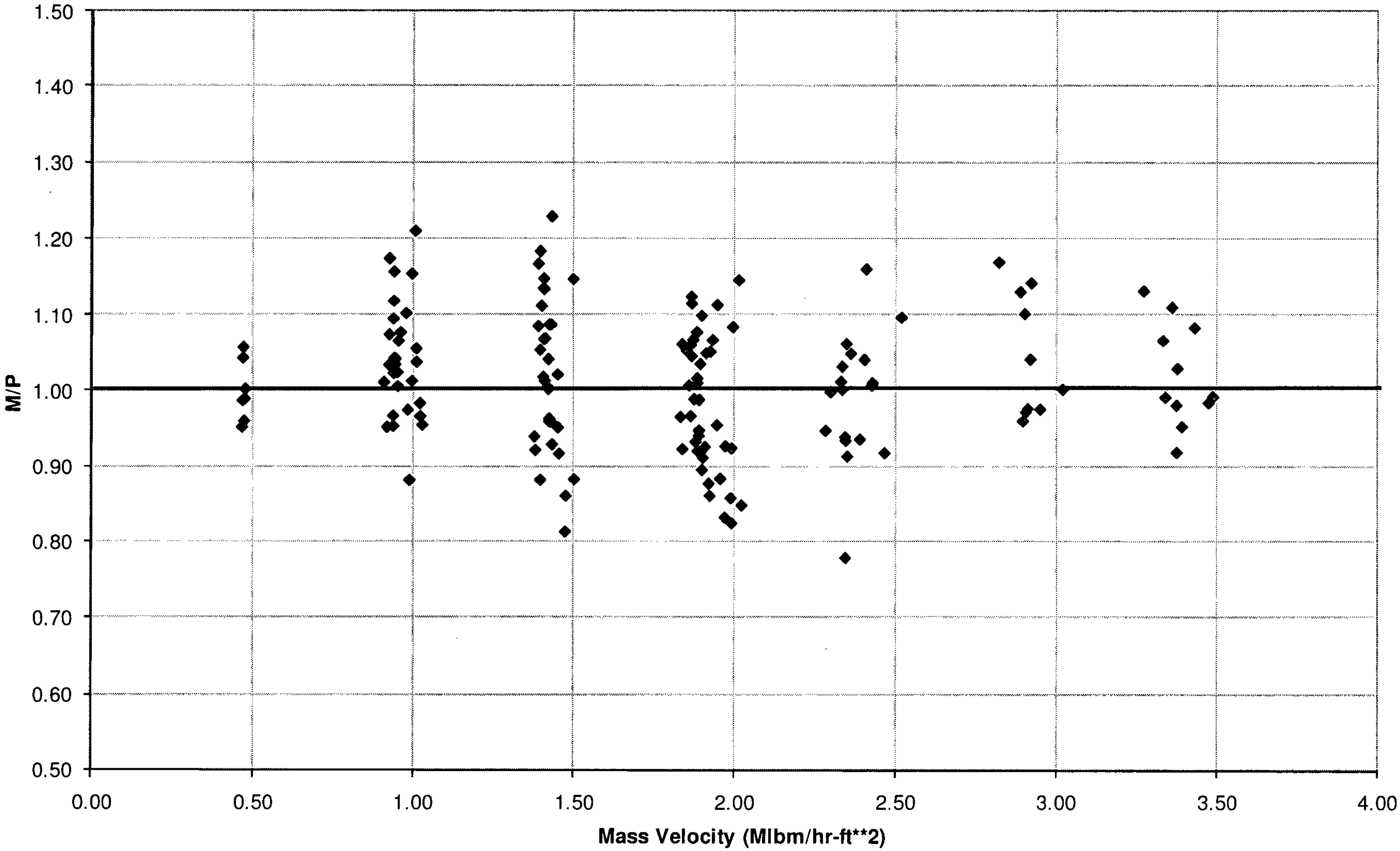


Figure A.4.2-5: DNBR vs. Pressure for BWU-ZM

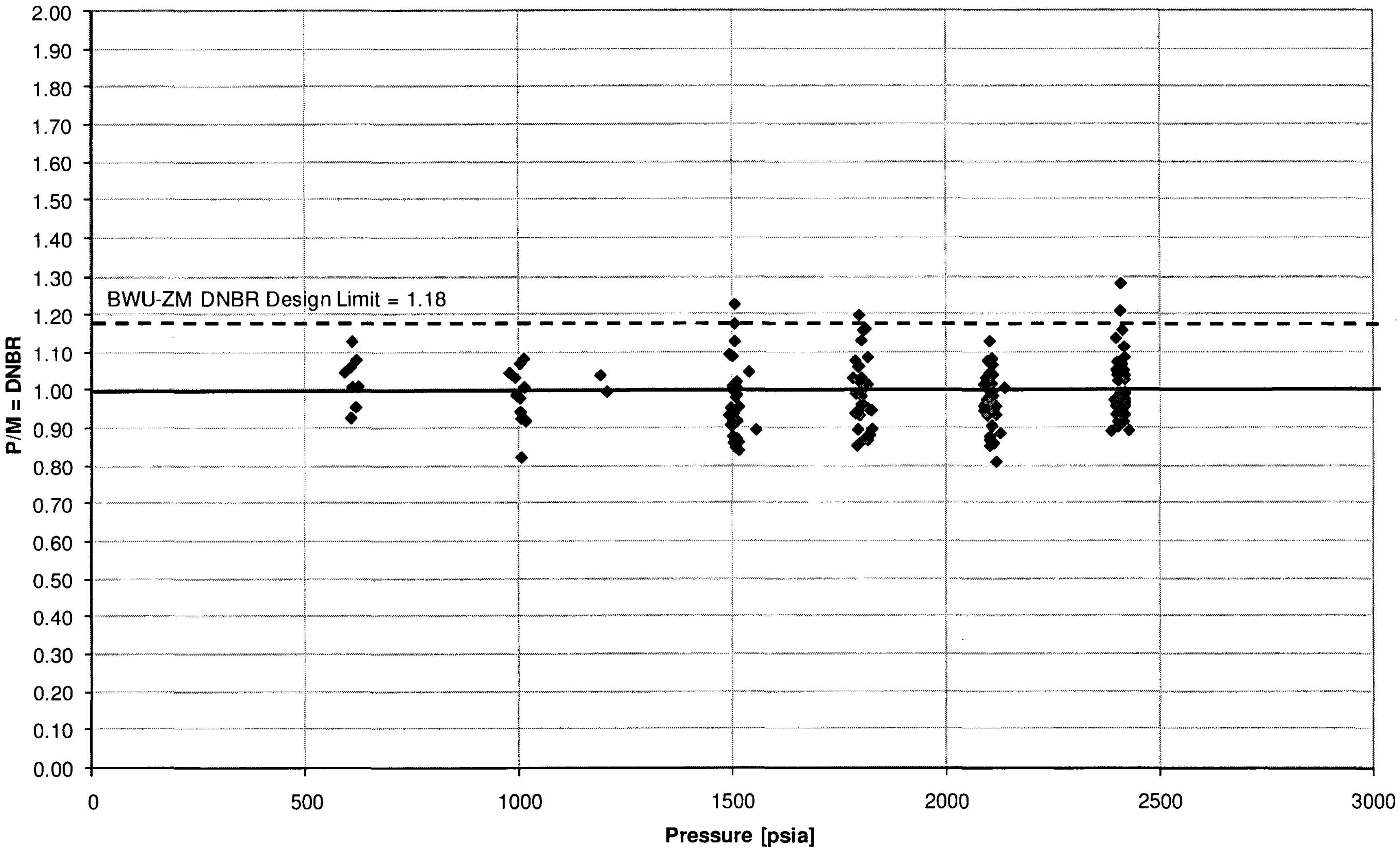


Figure A.4.2-6: DNBR vs. Quality for BWU-ZM

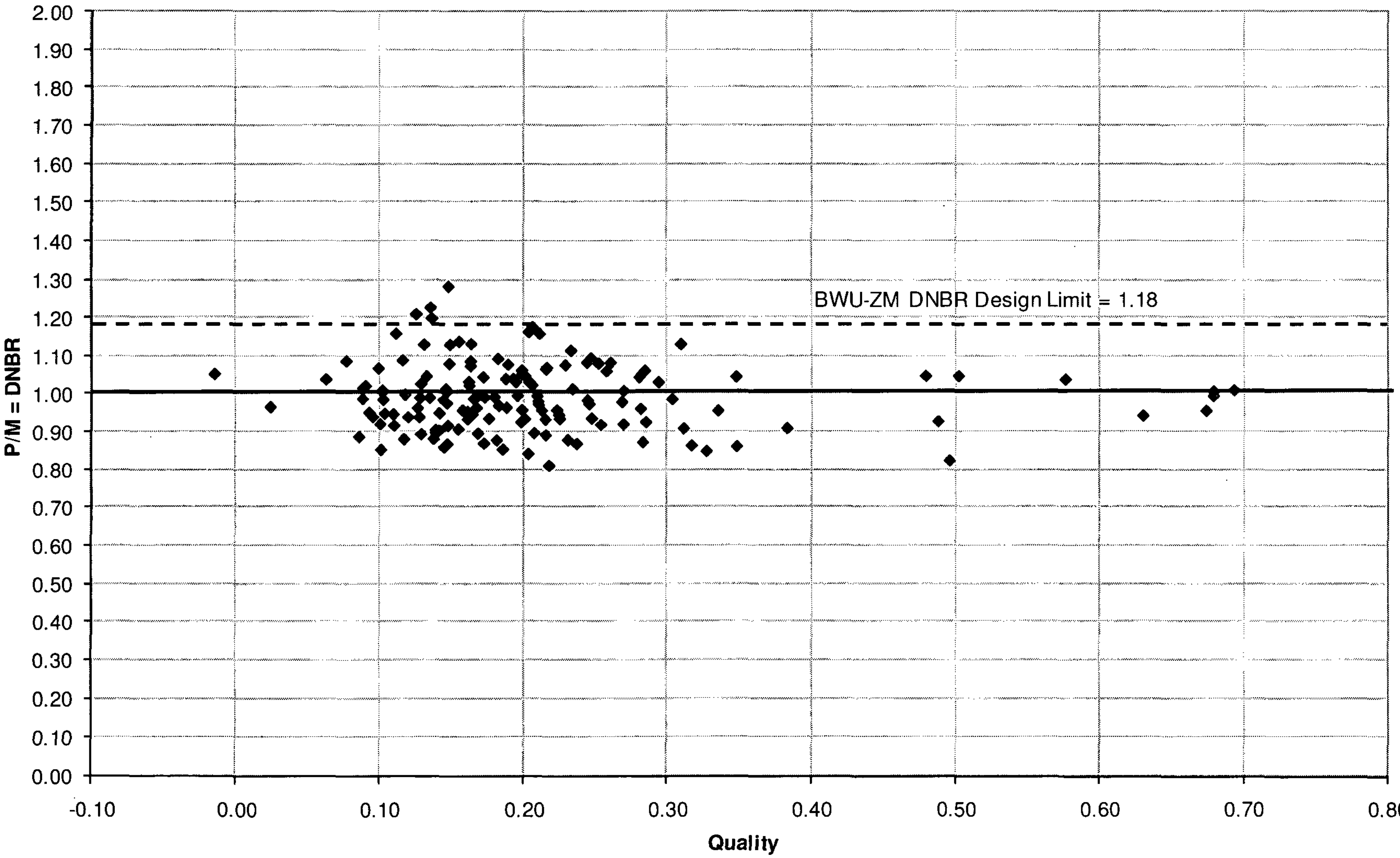
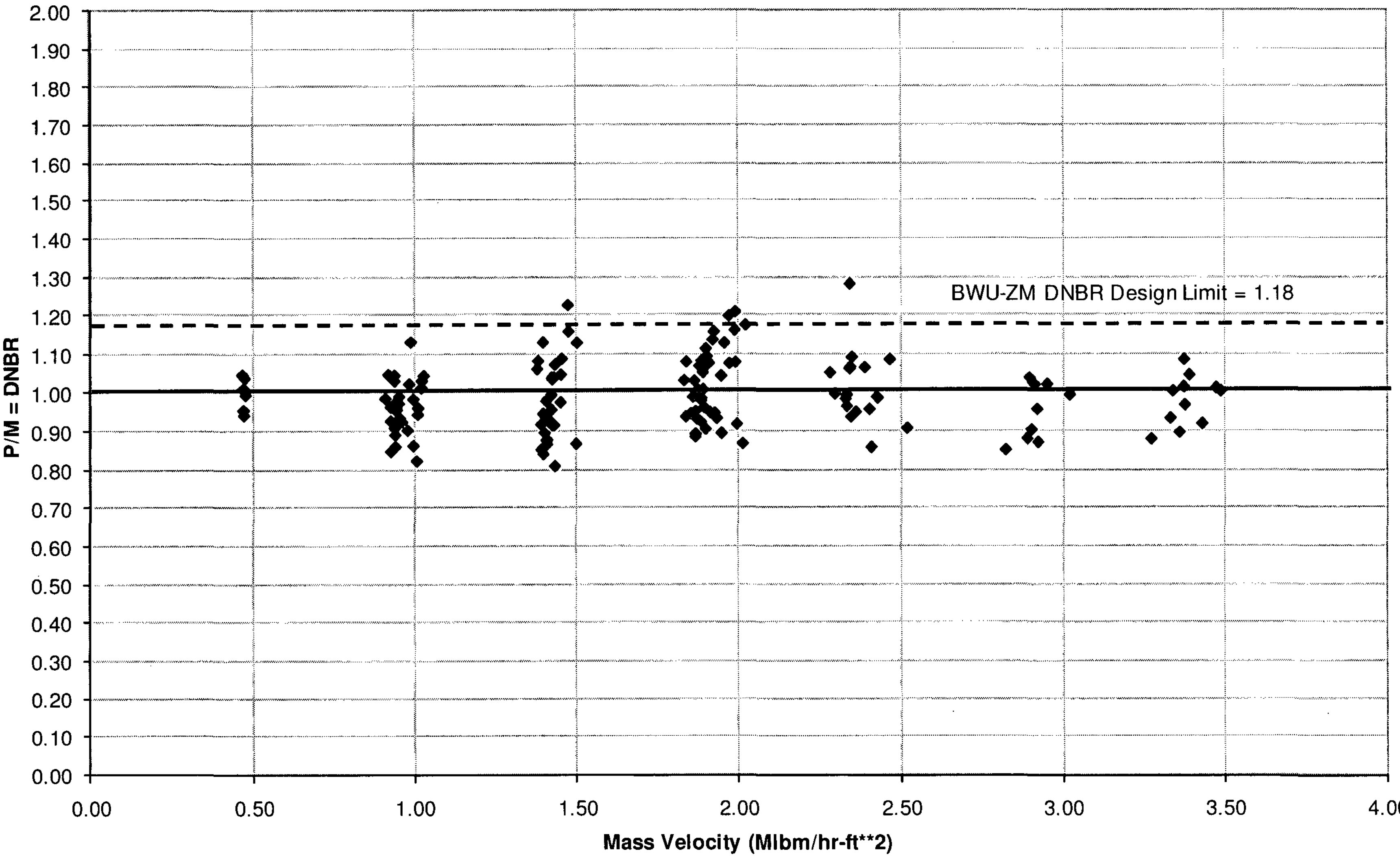


Figure A.4.2-7: DNBR vs. Mass Velocity for BWU-ZM



A.4.3 VIPRE-D/BWU-N RESULTS

The BWU-N correlation was developed by F-ANP correlating the CHF experimental results obtained in the ARC tests C-3, C-6, C-7, C-8, C-9, C-11, C-12, B-15, B-16 and B-17. Dominion has used those same experimental data to determine the VIPRE-D/BWU-N DNBR limit. Table A.4.3-1 summarizes the relevant statistics for each test, and calculates the aggregate statistics for the entire set of data.

Table A.4.3-1: VIPRE-D/BWU-N M/P Ratio Results

TEST	NUMBER OF TESTS	M/P RATIO AVERAGE	M/P RATIO STDEV	M/P RATIO MAX	M/P RATIO MIN
C-3	107	1.0655	0.1128	1.3251	0.7501
C-6	128	0.9445	0.1188	1.2966	0.6635
C-7	120	0.9757	0.0942	1.1553	0.6707
C-8	155	1.0076	0.0816	1.2127	0.7396
C-9	85	1.0373	0.0605	1.1681	0.8934
C-11	34	0.9986	0.0862	1.1389	0.8041
C-12	133	1.0083	0.0881	1.2003	0.7346
B-15	47	0.9806	0.0971	1.1263	0.7438
B-16	129	1.0052	0.1219	1.2627	0.6985
B-17	152	0.9988	0.1004	1.3507	0.8002
BWU-N	**1090**	**1.0018**	**0.1038**	**1.3507**	**0.6635**

One-sided tolerance theory (Reference A5) is used for the calculation of the VIPRE-D/BWU-N DNBR design limit. This theory allows us to calculate a DNBR limit so that, for a DNBR equal to the design limit, DNB will be avoided with 95% probability at a 95% confidence level.

Because all the statistical techniques used below assume that the original data distribution is normal, it is necessary to verify that the overall distribution for the M/P ratios is a normal distribution. To evaluate if the distribution is normal, the D' normality test was applied (Reference A6). A value of D' equal to 9,963.21 was obtained for the VIPRE-D/BWU-N database. This D' value is not within the range of acceptability for 1090 data points with a 95% confidence level (10,082.0 to 10,210.60)[e]. Since the value of D' is less than the lower critical value, the BWU-N distribution has greater kurtosis

[e] From Table 5 in Reference A6
D' Lower Limit (1090) [P = 0.025] = 9,530 + (40 / 50) x (10,220 – 9,530) = 10,082.0
D' Upper Limit (1090) [P = 0.975] = 9,653 + (40 / 50) x (10,350 – 9,653) = 10,210.6

than a normal distribution. Therefore, the one-sided theory is conservative for VIPRE-D/BWU-N. This behavior was also observed by F-ANP in Reference A1.

Based on the results listed in Table A.4.3-1, the DNBR limit can be calculated as:

$$DNBR_L = \frac{1.0}{M/P - K_{N,C,P} \cdot \sigma_{M/P}} \quad [A.4.3.1]$$

where

M/P = average measured to predicted ratio

$\sigma_{M/P}$ = standard deviation of the measured to predicted ratios of the database

$K_{N,C,P}$ = one-sided tolerance factor based on N degrees of freedom, C confidence level, and P portion of the population protected. This number is taken from Table 1.4.4 of Reference A5.

Then, the DNBR design limit for the VIPRE-D/BWU-N code/correlation pair can be calculated as described in Table A.4.3-2:

Table A.4.3-2: VIPRE-D/BWU-N DNBR Design Limit

			VIPRE-D/BWU-N
Number of data	n		1090
Degrees of freedom	N	= n - 1 - 14	1075
Average M/P	M/P		1.0018
Standard Deviation	$\sigma_{M/P}$		0.1038
Corrected Standard Deviation	σ_N	$= \sigma_{M/P} \cdot [\,(n-1)/N\,]^{1/2}$	0.1045
Owen Factor	K(1075,0.95,0.95)		1.7239
BWU-N Design limit	$DNBR_L$	= 1 / (1.0018 – 1.7239 · 0.1045)	**1.2170**

Figures A.4.3-1 through A.4.3-4 display the performance of the M/P ratio, and its distributions as a function of the pressure, mass velocity and quality. The objective of these plots is to show that there are no biases in the M/P ratio distribution, and that the performance of the BWU-N correlation is independent of the three variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes. Figures A.4.3-5 through A.4.3-7 display the performance of the P/M ratio (i.e. the DNBR) against the major independent variables for the BWU-N database. These plots also include a DNBR design limit line at 1.22. It can be seen that only 65 data points are above the DNBR design limit, and that these data in excess of the limit are distributed over the variable ranges tested.

In Reference A1, the USNRC argued that the performance of the BWU-N correlation might be deficient at the extremely low end of the pressure range. For that reason, F-ANP developed individual DNBR design limits for each low pressure group in the database. This approach allows users to use the BWU-N correlation at low pressures but imposes a higher DNBR limit to ensure that the correlation is used conservatively. Table A.4.3-3 summarizes the VIPRE-D/BWU-N DNBR limits calculated for the different pressure groups and compares them with the DNBR design limits obtained by F-ANP in Reference A1.

Table A.4.3-3: VIPRE-D/BWU-N DNBR Limits for Pressure Groups

	800 psia	**1200 psia**	**1500 – 2616 psia**
AVERAGE M/P	1.0019	1.0598	1.0007
STDEV	0.1186	0.0865	0.1036
N, # DATA	20	20	1050
K(N,0.95,0.95)	2.396	2.396	1.7249
VIPRE-D DNBR LIMIT	**1.393**	**1.173**	**1.217**
LYNX2 DNBR LIMIT	***1.387***	***1.290***	***1.207***

Dominion will take the VIPRE-D/BWU-N DNBR limit to be 1.22 for pressures equal to or greater than 1200 psia, and 1.39 at pressures less than 1200 psia.

Figure A.4.3-1: Measured vs. Predicted CHF for BWU-N

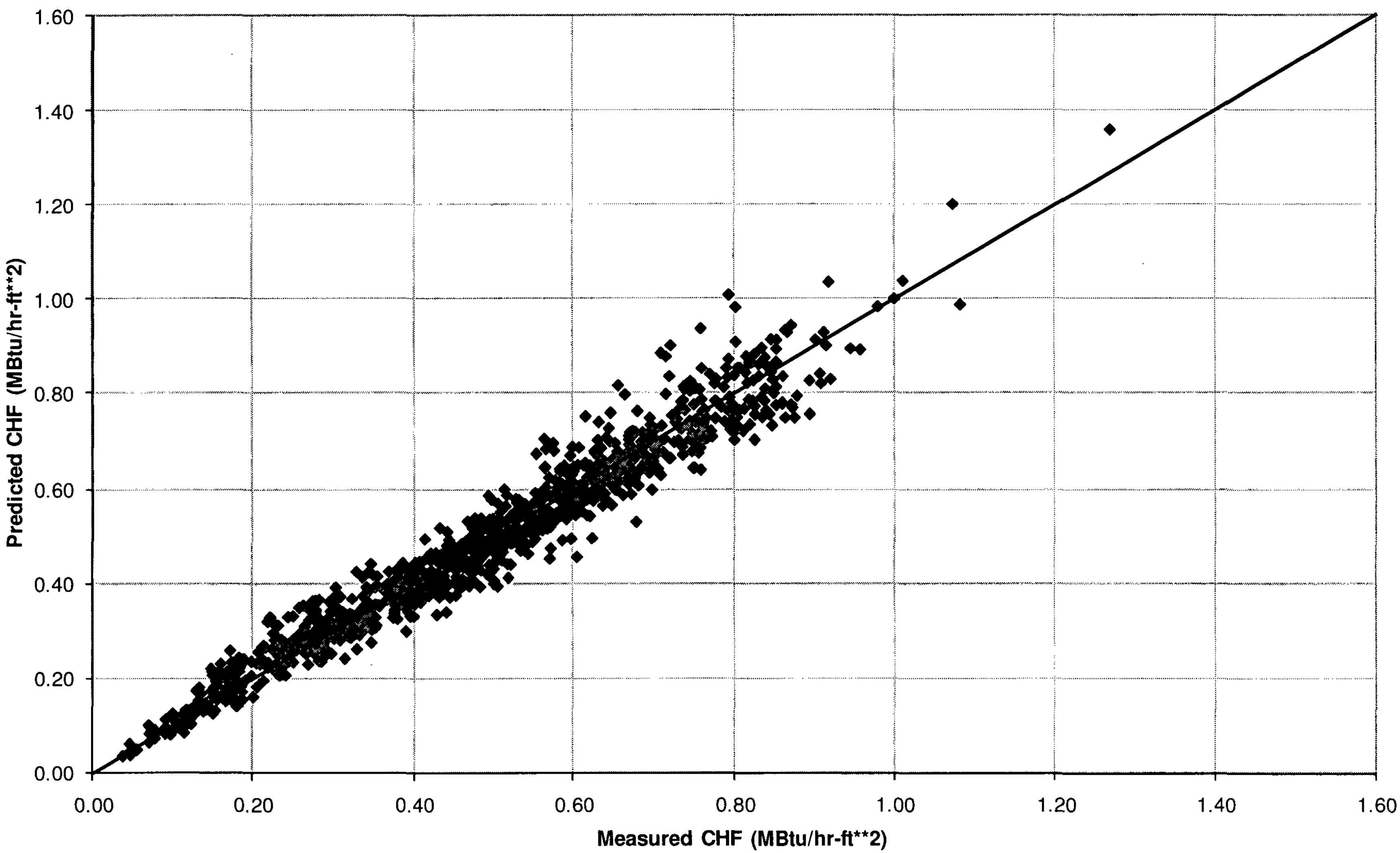


Figure A.4.3-2: M/P vs. Pressure for BWU-N

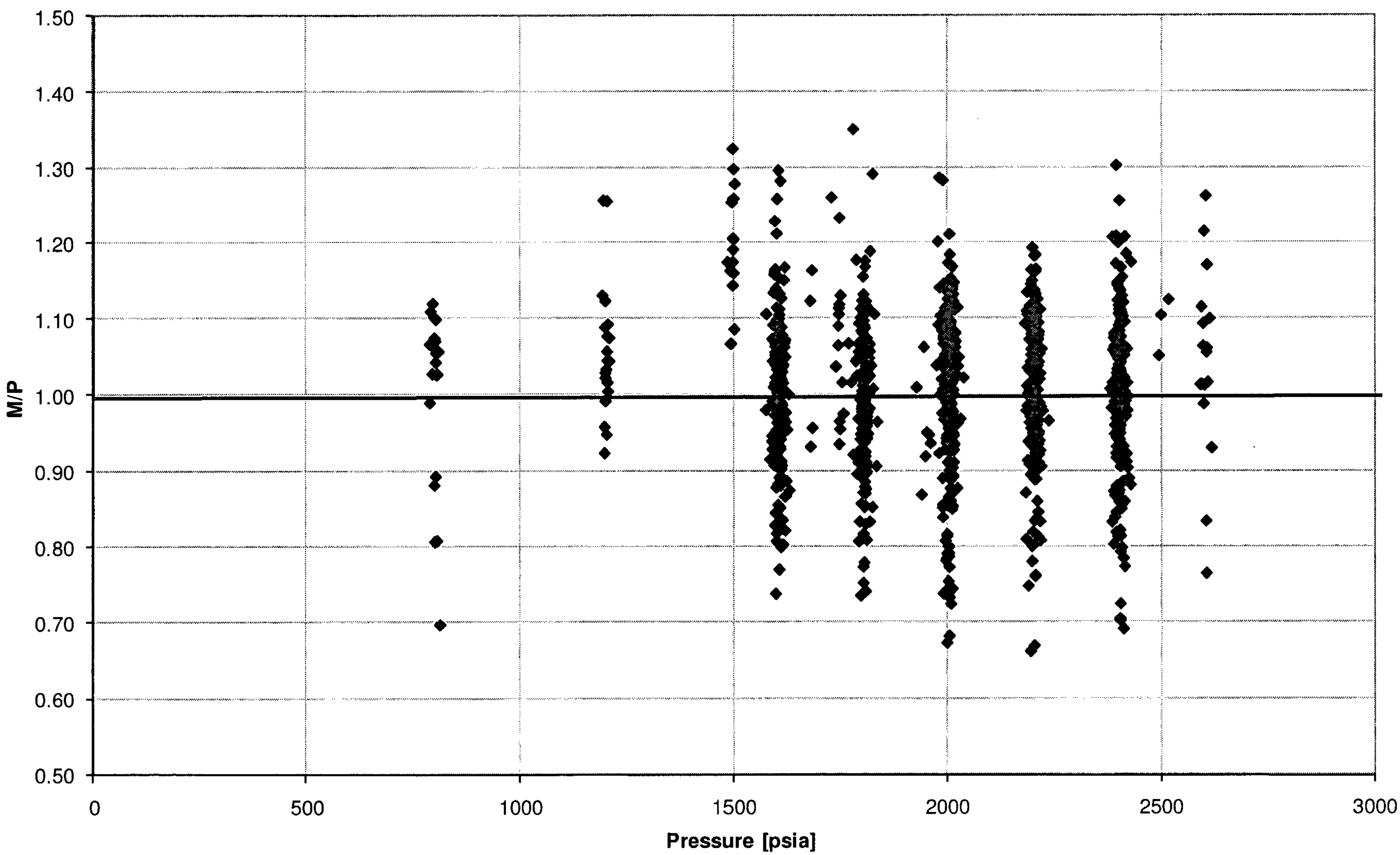


Figure A.4.3-3: M/P vs. Quality for BWU-N

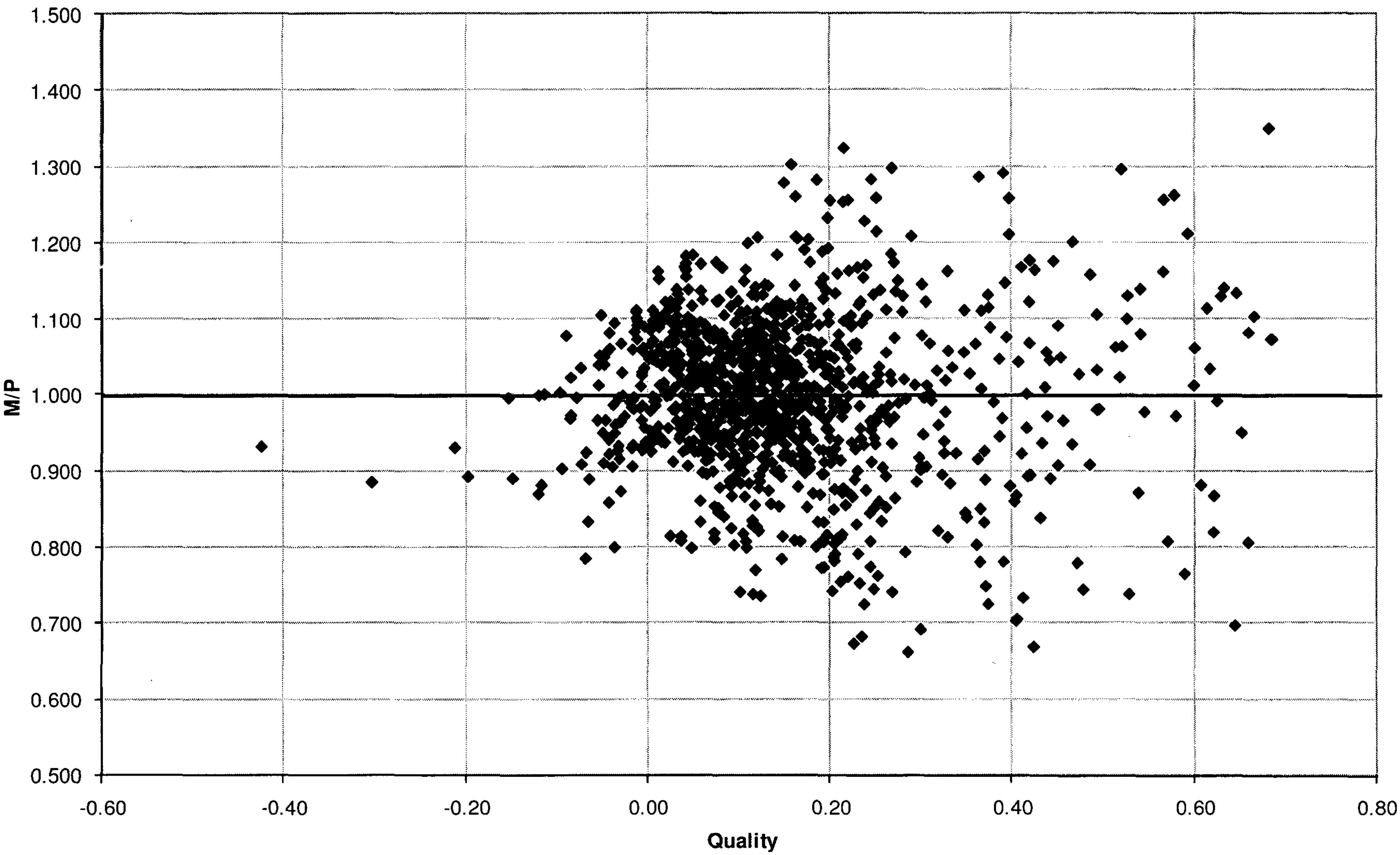


Figure A.4.3-4: M/P vs. Mass Velocity for BWU-N

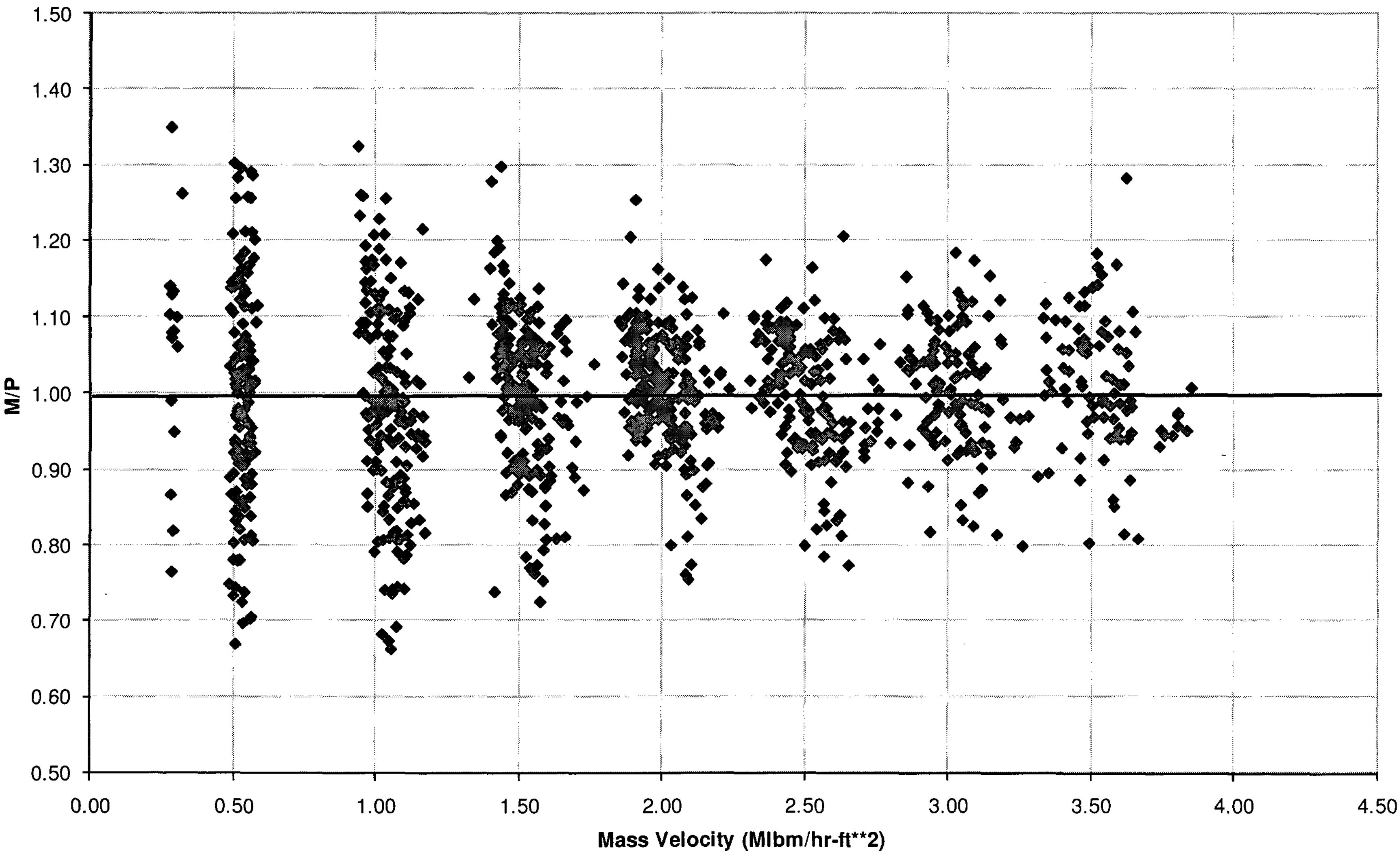


Figure A.4.3-5: DNBR vs. Pressure for BWU-N

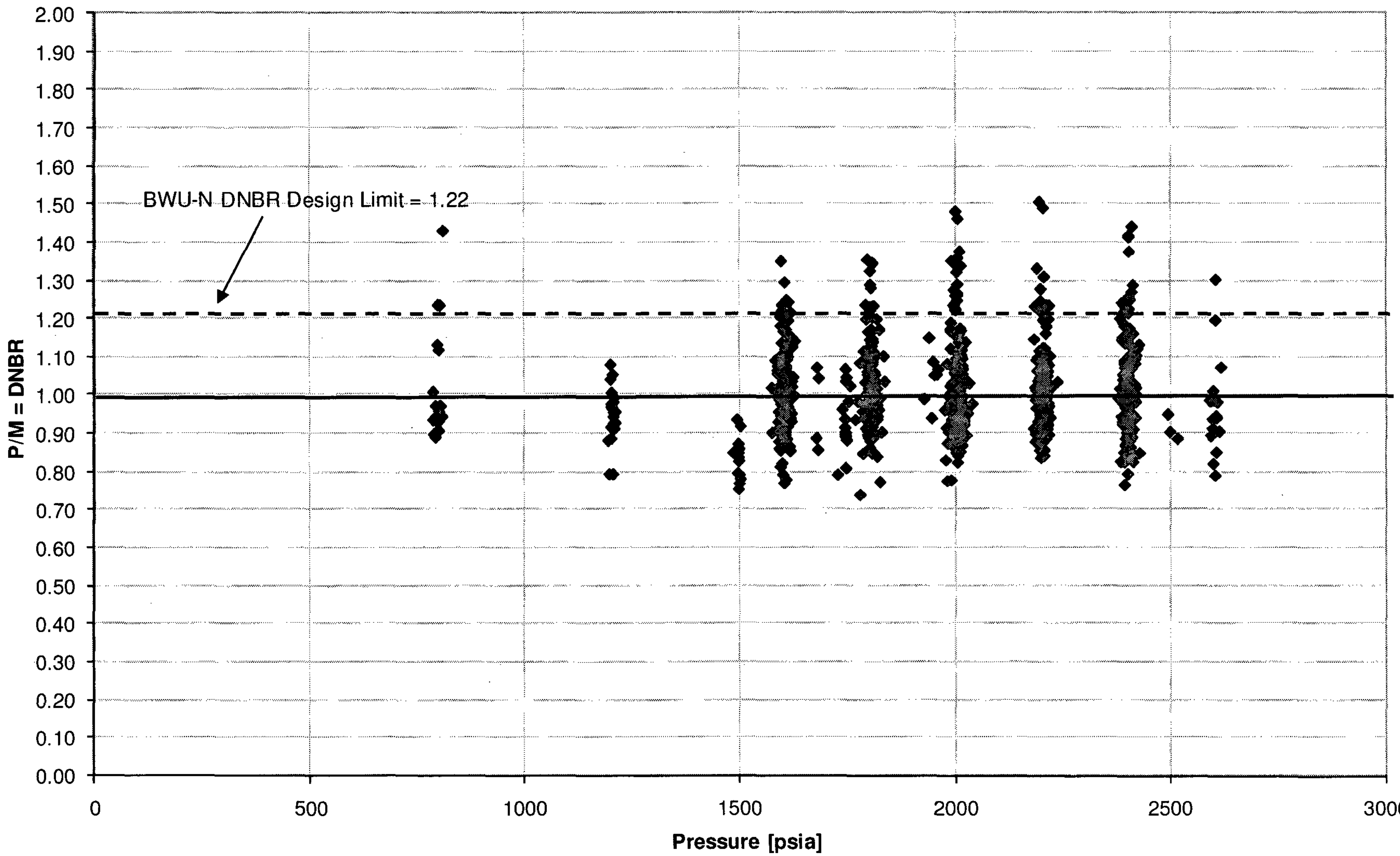


Figure A.4.3-6: DNBR vs. Quality for BWU-N

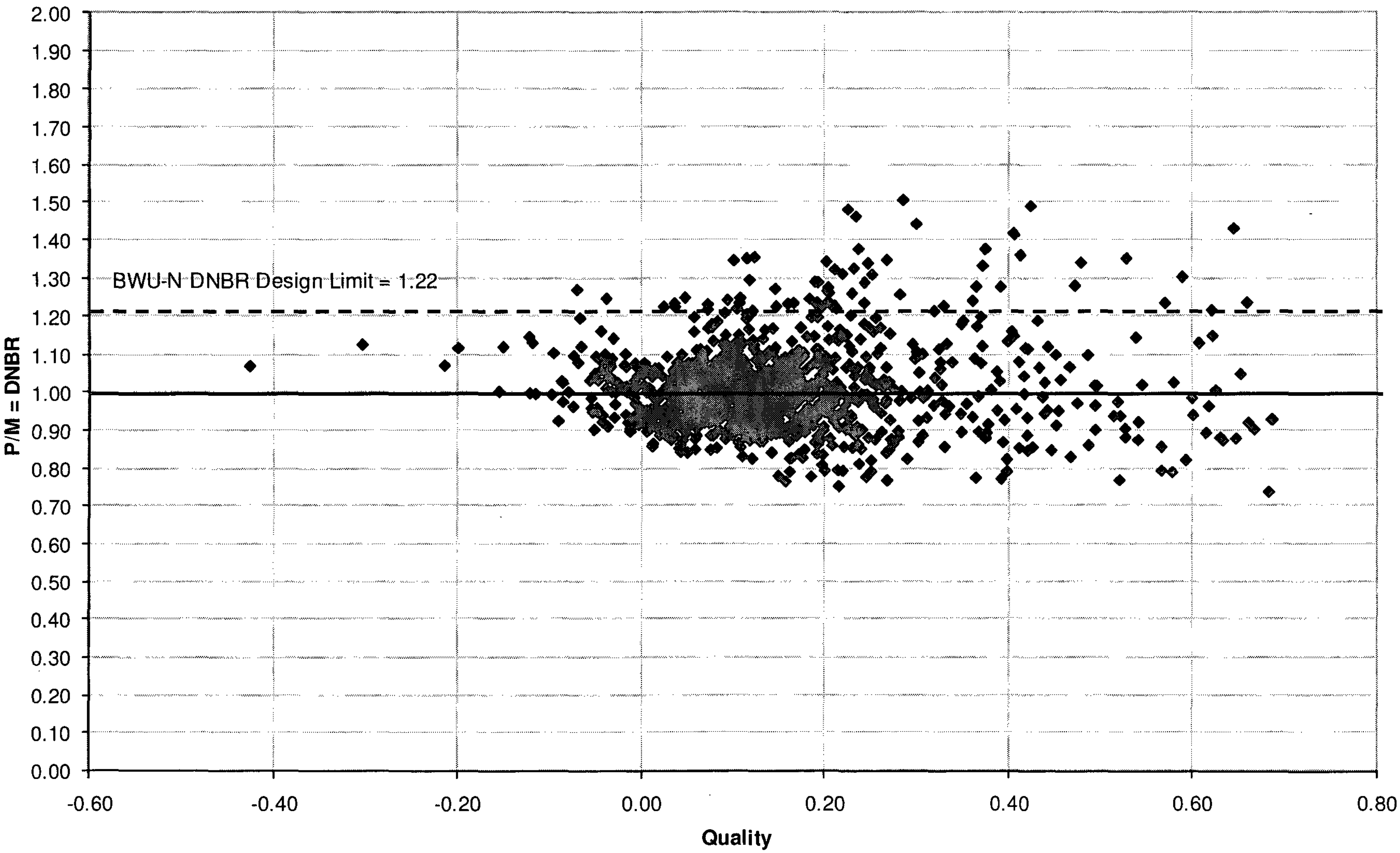
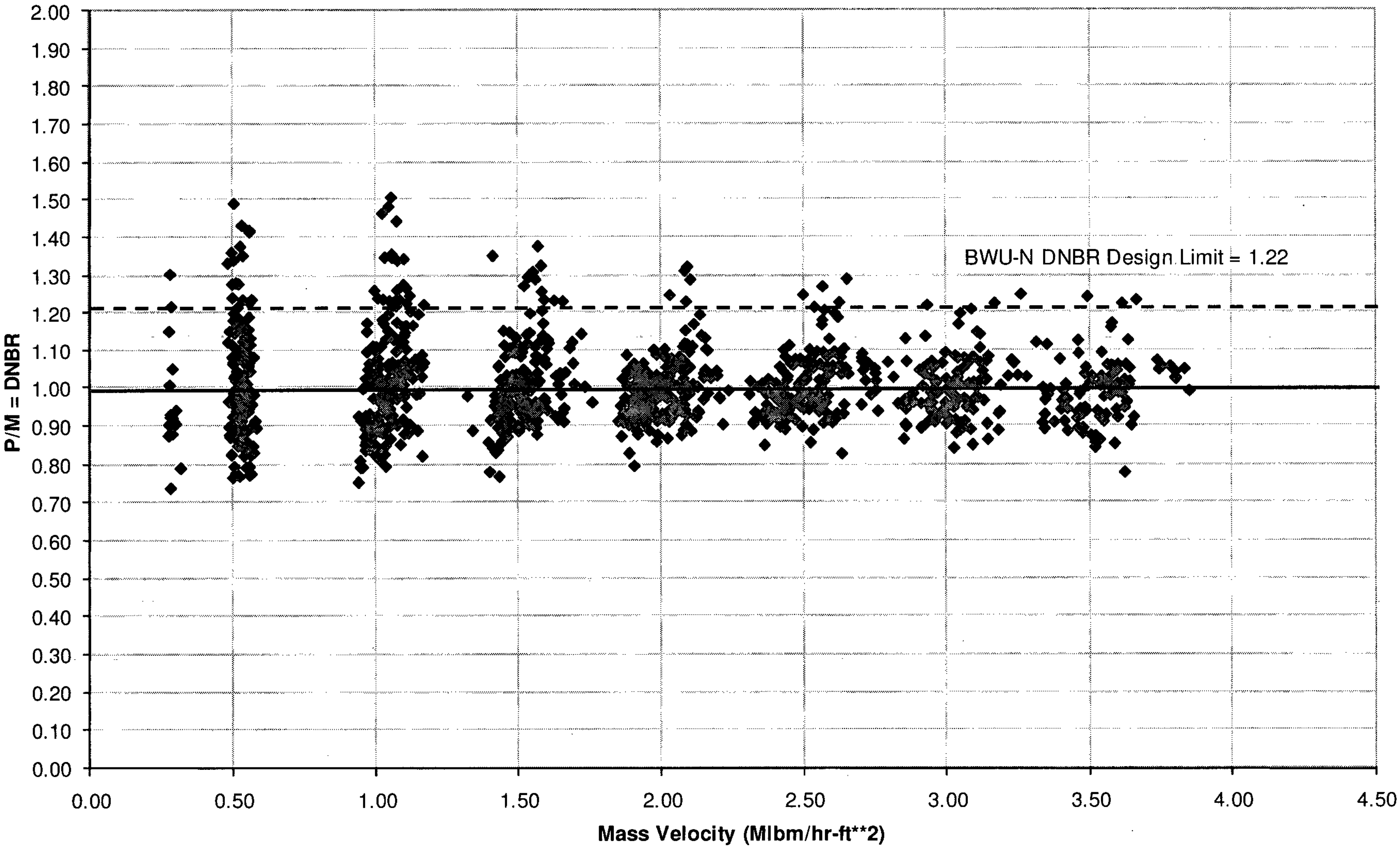


Figure A.4.3-7: DNBR vs. Mass Velocity for BWU-N



A.5 CONCLUSIONS

The BWU-Z, BWU-ZM and BWU-N correlations have been qualified with Dominion's VIPRE-D computer code. Table A.5-1 summarizes the DNBR design limits for VIPRE-D/BWU-Z, VIPRE-D/BWU-ZM and VIPRE-D/BWU-N that yield a 95% non-DNB probability at a 95% confidence level.

Table A.5-2 summarizes the applicability and the ranges of validity for all three CHF correlations, which are the same as those reported by F-ANP in References A1 and A2.

Table A.5-1: 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 at or above 594 psia	1.18
VIPRE-D/BWU-N	
DNBR limit below 1200 psia	1.39
DNBR limit at or above 1200 psia	1.22

Table A.5-2: 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

A.6 REFERENCES

A1. Technical Report, BAW-10199P-A, "The BWU Critical Heat Flux Correlations," Framatome Cogema Fuels, August 1996, including Addendum 1, December 2000.

A2. Technical Report, BAW-10199P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," Framatome Cogema Fuels, June 2002.

A3. Technical Report, BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, April 1985.

A4. Technical Report, BAW-10189P-A, "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," Framatome Technologies, January 1996.

A5. Technical Report, "Tables for Normal Tolerance Limits, Sampling Plans, and Screening," R. E. Odeh and D. B. Owen, 1980.

A6. Technical Report, "Assessment of the Assumption of Normality (employing individual observed values)," American National Standards Institute, ANSI N15.15.1974.

DOM-NAF-2, Rev. 0.1-A
APPENDIX B

Qualification of the Westinghouse WRB-1 CHF Correlations in the Dominion VIPRE-D Computer Code

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CLASSIFICATION/DISCLAIMER

ABSTRACT

This Appendix documents Dominion's qualification of the Westinghouse WRB-1 CHF correlation with the VIPRE-D code. This qualification was performed against the same CHF experimental database used by Dominion to qualify the COBRA/WRB-1 code/correlation pair. This Appendix summarizes the data evaluations that were performed to qualify the VIPRE-D/WRB-1 code/correlation pair and to develop the corresponding DNBR design limit for the correlation.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

AO's	Axial Offset Envelope
CHF	Critical Heat Flux
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
FLC	Form Loss Coefficient
FWMAL	Feedwater Malfunction Transient
HTRF	Heat Transfer Research Facility at Columbia University
LOCROT	Locked Rotor Accident
LOFA	Loss of Flow Accident
M/P	Ratio of Measured-to-Predicted CHF
MSLB	Main Steam Line Break
MVG	Mixing Vane Grid
NMVG	Non-Mixing Vane Grid
P/M	Ratio of Predicted-to-Measured CHF (equivalent to DNBR)
PWR	Pressurized Water Reactor
RWAP	Rod Withdrawal at Power
RWSC	Rod Withdrawal from Subcritical
SIF	Surry Improved Fuel
SPS	Surry Power Station
USNRC	US Nuclear Regulatory Commission

B.1 PURPOSE

Dominion currently uses Westinghouse 15x15 OFA fuel assemblies at Surry Power Station, Units 1 and 2. This fuel product as implemented at Surry is also known as Surry Improved Fuel (SIF). The thermal-hydraulic analysis of this Westinghouse fuel product requires the use of the Westinghouse WRB-1 CHF Correlation (References B1 and B3). In fact, Westinghouse WRB-1 CHF correlation has been approved by the USNRC for use with Westinghouse 15x15 and 17x17 "R" grid type fuel, and with Westinghouse 14x14, 15x15 and 17x17 OFA-type fuel products (Reference B1).

To be licensed for use, a critical heat flux (CHF) correlation must be tested against experimental data that span the anticipated range of conditions over which the correlation will be applied. Furthermore, the population statistics of the database must be used to establish a departure from nucleate boiling ratio (DNBR) design limit such that the probability of avoiding departure from nucleate boiling (DNB) will be at least 95% at a 95% confidence level.

This Appendix documents Dominion's qualification of the WRB-1 correlation with the VIPRE-D code. This qualification was performed against a subset of the data from the Columbia-EPRI CHF database for Westinghouse "R" grid 17x17 and 15x15 fuel (Reference B2). This is the same subset of the Columbia-EPRI CHF database used by Dominion in the qualification of the WRB-1 correlation with the COBRA code (Reference B3). This Appendix summarizes the data evaluations that were performed to qualify the VIPRE-D/WRB-1 code/correlation pair, and to develop the corresponding DNBR design limits for the correlation.

B.2 APPLICABILITY

Dominion intends to use the VIPRE-D/WRB-1 code/correlation pair for the analysis of Westinghouse 15x15 and 17x17 "R" grid type fuel, and Westinghouse 14x14, 15x15 and 17x17 OFA-type fuel products in PWR reactors. When evaluating these types of fuels outside of the range of validity of the WRB-1 CHF correlation, Dominion intends to use the VIPRE-D/W-3 code/correlation pair. W-3 is one of the CHF correlations contained in the USNRC approved generic version of VIPRE-01 (References B7 and B8).

The intended VIPRE-D/WRB-1 applications discussed in this Appendix are consistent with the generic intended applications listed in the main body of this report (Section 2.0). Also, more specifically, Dominion intends to use VIPRE-D/WRB-1 to analyze the transients delineated in Table 2.1-1 in Section 2.0 of the main body of this report.

The qualification of the WRB-1 CHF correlation with the VIPRE-D code has been performed following the modeling guidelines described in Section 4.0 of this report. In addition, extensive code benchmark calculations have confirmed that the VIPRE-D models specified in sections 4.1 through 4.12 in the main body of this report produce essentially the same results as equivalent Dominion COBRA models. Some of these benchmarks are described in section B.7 of this Appendix.

This Appendix is submitted to the USNRC for review and approval in order to meet the USNRC's requirement #2 listed in the VIPRE-01 SER, as outlined in Section 2.2 in the main body of this report.

B.3 DESCRIPTION OF THE WESTINGHOUSE WRB-1 CHF CORRELATION

In pressurized water reactor (PWR) cores, the energy generated inside the fuel pellets leaves the fuel rods at their surface in the form of heat flux, which is removed by the reactor coolant system flow. The normal heat transfer regime in this configuration is nucleate boiling, which is very efficient. However, as the capacity of the coolant to accept heat from the fuel rod surface degrades, a continuous layer of steam (a film) starts to blanket the tube. This heat transfer regime, termed film boiling, is less efficient than nucleate boiling and can result in significant increases of the fuel rod temperature for the same heat flux. Since the increase in temperature may lead to the failure of the fuel rod cladding, PWRs are designed to operate in the nucleate boiling regime and protection against operation in film boiling must be provided.

The heat flux at which the steam film starts to form is called CHF or the point of DNB. For design purposes, the DNBR is used as an indicator of the margin to DNB. The DNBR is the ratio of the predicted CHF to the actual local heat flux under a given set of conditions. Thus, DNBR is a measure of the thermal margin to film boiling and its associated high temperatures. The greater the DNBR value (above 1.0), the greater the thermal margin.

The CHF cannot be predicted from first principles, so it is empirically correlated as a function of the local thermal-hydraulic conditions, the geometry, and the power distribution measured in the experiments. Since a CHF correlation is an analytical fit to experimental data, it has an associated uncertainty, which is quantified in a DNBR design limit. A calculated DNBR value greater than this design limit provides assurance that there is at least a 95% probability at the 95% confidence level that a departure from nucleate boiling will not occur.

The Westinghouse WRB-1 CHF correlation is defined in Reference B1 as:

$$\frac{Q_{CHF}}{10^6} = PF + A_1 + B_3\frac{(G_{LOC})}{10^6} - B_4\frac{(G_{LOC})}{10^6}X_{LOC} \qquad \text{[B.3.1]}$$

where Q_{CHF} is the critical heat flux in Btu/hr-ft^2, PF is a dimensionless performance factor dependent on the outer diameter of the rods and defined in Reference B1, G_{LOC} is the local mass velocity in Mlbm/ft^2-hr, and X_{LOC} is the local quality. The specific formulations for each one of these components, as well as the corresponding constants, are Westinghouse proprietary and can be found in Reference B1. Reference B1 discusses the application of the WRB-1 correlation form to the "L" and "R" grid fuel assembly designs. Westinghouse WRB-1 CHF correlation has been approved by the USNRC for use with Westinghouse 15x15 and 17x17 "R" grid type fuel, and with Westinghouse 14x14, 15x15 and 17x17 OFA-type fuel products (Reference B1). Its intended range of application for operating conditions is as follows (Reference B3):

$1440 \le$ Pressure ≤ 2490 psia

$0.9 \le$ Mass Flux ≤ 3.7 Mlbm/hr-ft^2

Local Quality ≤ 0.30

In response to concerns raised by the NRC in Reference B3, Dominion will impose two additional restrictions on the intended range of application:

- VIPRE-D/WRB-1 will not be used when the local heat flux exceeds 1.0 Mbtu/hr-ft^2.
- VIPRE-D/WRB-1 will not be used for fuel with less than 13" mixing vane grid spacing.

The W-3 correlation is used when conditions are outside the range of the WRB-1 DNB correlation. Specifically, the W-3 correlation is applied to the lower portion of the fuel assemblies in the rod withdrawal from subcritical event because of the bottom peaked axial power distribution assumed, and in the steam line break event because of the low pressures involved. The W-3 correlation with a correlation limit of 1.30 is used below the fuel assembly first mixing vane grid for the rod withdrawal from subcritical event. For the steam line break event, the W-3 correlation is used with a correlation limit of 1.45 in the pressure range of 500 to 1000 psia and 1.30 for pressures above 1000 psia (Reference B11). The Westinghouse W-3 CHF correlation is described on page 10 in Reference B10.

B.4 DESCRIPTION OF VIPRE-D/WRB-1 DATABASE

The WRB-1 CHF correlation was developed from a large body of rod bundle CHF data obtained at the Columbia University Heat Transfer Research Facility (HTRF) using full-scale, electrically heated rod bundle test sections (Reference B2).

The Dominion qualification of WRB-1 in VIPRE-D was performed against a subset of the data from the Columbia-EPRI CHF database for Westinghouse "R" grid 17x17 and 15x15 fuel (Reference B2). Dominion analyzed 19 test series out of the 22 series used to develop the correlation; in particular, Dominion did not consider the three series of "L" grid tests and as a consequence no "L" grid data were included in the test population. The 19 tests represent the same subset of the Columbia-EPRI CHF database used by Dominion in the qualification of the WRB-1 correlation with the COBRA code (Reference B3).

Two criteria were used to justify data deletions:

1) The first was consistency with the practice of the test sponsor. Certain points were excluded from the COBRA/WRB-1 database because they had been excluded from the THINC/WRB-1 database in Reference B1. Most excluded data were deleted under this condition. These points were also excluded from the COBRA/WRB-1 database.

2) The second exclusion criterion was consistency of the input data in References B1 and B2. Although some differences were expected, data points that differed by more than ten standard deviations were excluded as being probable typographical errors in Reference B2.

With the exception of the "L" grid data, and the 25 data points that were thrown out under the second criterion, the VIPRE-D/WRB-1 database is the same as the one used in Reference B1 to qualify THINC/WRB-1 for "R" grid fuel. This is also the same database used by Westinghouse to qualify VIPRE-01/WRB-1 (Reference B6). The same 945 statepoints used in Reference B3 by Dominion in the qualification of the WRB-1 correlation with the COBRA code were used in this calculation. Since no "L" grid data were included in the test population, Dominion does not intend to apply the WRB-1 correlation to Westinghouse 15x15 standard fuel.

B.5 VIPRE-D/WRB-1 Test Assemblies

B.5.1 4x4 Geometry Tests

Twelve of the nineteen tests used by Dominion to qualify the VIPRE-D/WRB-1 code/correlation pair have a 4x4 geometry. These 4x4 test bundles have essentially a 15x15 subchannel geometry (Reference B3, page 13). Table B.5.1-1 provides a summary of the key information about each test.

Table B.5.1-1: 4x4 VIPRE-D/WRB-1 Experimental Database

TEST	MATRIX	AXIAL HEAT FLUX SHAPE	PIN OD / GUIDE TUBE OD [inches]	HEATED LENGTH [inches]	GRID SPACING [inches]	NUMBER OF TESTS IN VIPRE-D/WRB-1 DATABASE
124	4 x 4	Non-Uniform	0.422 / -	96	20	32
125	4 x 4	Non-Uniform	0.422 / -	96	20	33
127	4 x 4	Non-Uniform	0.422 / -	96	Non-Uniform	36
131	4 x 4	Non-Uniform	0.422 / -	168	26	32
132	4 x 4	Non-Uniform	0.422 / -	168	20	36
133	4 x 4	Non-Uniform	0.422 / -	168	13	35
134	4 x 4	Non-Uniform	0.422 / -	168	32	38
140	4 x 4	Non-Uniform	0.422 / -	96	32	30
148	4 x 4	Non-Uniform	0.422 / -	168	26	70
153	4 x 4	Uniform	0.422 / -	168	26	40
146	4 x 4	Non-Uniform	0.422 / 0.545	168	26	37
139	4 x 4	Non-Uniform	0.422 / 0.545	168	32	37

B.5.2 5x5 Geometry Tests

Seven of the nineteen tests used by Dominion to qualify the VIPRE-D/WRB-1 code/correlation pair have a 5x5 geometry. These 5x5 test bundles have the same subchannel geometry as the current Westinghouse 17x17 "R" grid fuel. Table B.5.2-1 provides a summary of key information about each test.

Table B.5.2-1: 5x5 VIPRE-D/WRB-1 Experimental Database

TEST	MATRIX	AXIAL HEAT FLUX SHAPE	PIN OD / GUIDE TUBE OD [inches]	HEATED LENGTH [inches]	GRID SPACING [inches]	NUMBER OF TESTS IN VIPRE-D/WRB-1 DATABASE
161	5 x 5	Uniform	0.374 / -	168	22	71
156	5 x 5	Uniform	0.374 / -	168	26	70
160	5 x 5	Uniform	0.374 / -	96	22	65
157	5 x 5	Uniform	0.374 / -	96	26	76
164	5 x 5	Non-Uniform	0.374 / -	168	22	74
162	5 x 5	Non-Uniform	0.374 / 0.485	168	22	70
158	5 x 5	Uniform	0.374 / 0.482	96	26	63

B.6 VIPRE-D RESULTS AND COMPARISON TO COBRA

Reference B2 describes the mathematical model for each separate test section by providing the bundle and cell geometry, the rod radial peaking values, the rod axial flux shapes, the types, axial locations and form losses associated to the spacer grids, as well as the thermocouple locations. Reference B2 provides the data for each CHF observation within a test, including power, flow, inlet temperature, pressure and CHF axial location.

Each test section was modeled for analysis with the VIPRE-D thermal-hydraulic computer code as a full assembly model following the modeling methodology discussed in Section 4 in the main body of this report. For each set of bundle data, VIPRE-D produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc.) at every axial node along the heated length of the test section. The ratio of measured-to-predicted CHF (M/P) is the variable that is normally used to evaluate the thermal-hydraulic performance of a code/correlation pair. The measured CHF is the local heat flux at a given location, while the predicted CHF is calculated by the code using the WRB-1 CHF correlation. The ratio of these two values provides the M/P ratio, which is the inverse of the DNB ratio. M/P ratios are frequently used to validate CHF correlations instead of DNB ratios, because their distribution is usually a normal distribution, which simplifies their manipulation and statistical analysis.

In addition to comparing to the experimental results, the results obtained by VIPRE-D when modeling the experiments were benchmarked against the results obtained with the COBRA code in the USNRC approved COBRA topical (Reference B3). This comparison was just a sanity check to verify that there are no suspect datapoints and that the statepoint conditions were correctly input to the code.

This section summarizes the VIPRE-D results and the associated significant statistics. In addition, this section shows a comparison to the results obtained with the COBRA code as reported in Reference B3. This section also shows the variation of the M/P ratio with each independent variable to demonstrate that there are no biases in the data. Finally, it provides the VIPRE-D overall statistics for the nineteen WRB-1 tests and generates the DNBR design limit for the WRB-1 CHF correlation with VIPRE-D.

The WRB-1 correlation was developed by Westinghouse by correlating the CHF experimental results obtained in the tests as described in Reference B1. Westinghouse also used these test data to calculate a DNBR design limit of 1.17 for the WRB-1 correlation (References B1 and B6). Dominion used a subset of this experimental data, as described in section B.4, to develop the VIPRE-D/WRB-1 DNBR limit. Table B.6-1 summarizes the relevant statistics for each test, and calculates the aggregate statistics for the entire set of data.

One-sided tolerance theory (Reference B4) is used for the calculation of the VIPRE-D/WRB-1 DNBR design limit. This theory allows us to calculate a DNBR limit so that, for a DNBR equal to the design limit, DNB will be avoided with 95% probability at a 95% confidence level.

Table B.6-1: VIPRE-D/WRB-1 M/P Ratio Results

TEST	NUMBER OF TESTS	M/P RATIO AVERAGE	M/P RATIO STDEV	M/P RATIO MAX	M/P RATIO MIN
TS124	32	0.9984	0.0510	1.083	0.851
TS125	33	0.9352	0.0533	1.041	0.802
TS127	36	1.0209	0.0877	1.307	0.897
TS131	32	1.0383	0.0827	1.188	0.799
TS132	36	1.0382	0.1006	1.185	0.804
TS133	35	0.9473	0.0713	1.093	0.786
TS134	38	1.0321	0.0891	1.236	0.864
TS140	30	1.0206	0.0714	1.154	0.803
TS148	70	1.0138	0.0815	1.170	0.766
TS153	40	0.9220	0.0592	1.013	0.760
TS146	37	0.9942	0.0516	1.086	0.899
TS139	37	0.9597	0.0834	1.122	0.787
4x4	**456**	**0.9941**	**0.0847**	**1.307**	**0.760**
TS161	71	1.0012	0.0624	1.170	0.833
TS156	70	1.0132	0.0780	1.163	0.812
TS160	65	1.0238	0.0812	1.171	0.763
TS157	76	1.0222	0.0769	1.223	0.800
TS164	74	1.0468	0.0869	1.271	0.841
TS162	70	0.9825	0.0712	1.156	0.845
TS158	63	1.0168	0.0848	1.223	0.823
5x5	**489**	**1.0154**	**0.0794**	**1.271**	**0.763**
VIPRE-D/WRB-1	**945**	**1.0051**	**0.0827**	**1.307**	**0.760**
COBRA/WRB-1 (Reference B3)	**945**	**1.0010**	**0.0838**	**1.287**	**0.745**

Because all the statistical techniques used below assume that the original data distribution is normal, it is necessary to verify that the overall distribution for the M/P ratios is a normal distribution. To evaluate if the distribution is normal, the D' normality test was applied (Reference B5). A value of D' equal to 8,160.9 was obtained for the VIPRE-D/WRB-1 database. This D' value is within the range of acceptability for 945 data points with a 95% confidence level (8,134.0 to 8,245.4)[a]. Thus, it is concluded that the M/P distribution for the VIPRE-D/WRB-1 database is indeed normal.

[a] From Table 5 in Reference B5

D' Lower Limit (945) [P = 0.025] = 7,558 + (45 / 50) x (8,198 – 7,558) = 8,134.0

D' Upper Limit (945) [P = 0.975] = 7,664 + (45 / 50) x (8,310 – 7,664) = 8,245.4

Based on the results listed in Table B.6-1, the deterministic DNBR design limit can be calculated as:

$$DNBR_L = \frac{1.0}{M/P - K_{N,C,P} \cdot \sigma_{M/P}} \qquad [B.6.1]$$

where

- M/P = average measured-to-predicted CHF ratio
- $\sigma_{M/P}$ = standard deviation of the measured-to-predicted CHF ratios of the database
- $K_{N,C,P}$ = one-sided tolerance factor based on N degrees of freedom, C confidence level, and P portion of the population protected. This number is taken from Table 1.4.4 in Reference B4.

Normally, the number of degrees of freedom would be the total number of data minus one. However, because Westinghouse used these experimental data to correlate the 12 constants that appear in the WRB-1 correlation, the total number of degrees of freedom must be corrected to account for this. In addition, the standard deviation of the database needs to be corrected accordingly to account for this reduced number of degrees of freedom:

$$N = n - 1 - 12$$

$$\sigma_N = \sigma_{M/P} \cdot [(n-1)/N]^{1/2} \qquad [B.6.2]$$

Then, the DNBR design limit for the VIPRE-D and the WRB-1 correlation can be calculated as described in Table B.6-2:

Table B.6-2: VIPRE-D/WRB-1 DNBR Design Limit

			VIPRE-D/WRB-1
Number of data	n		945
Degrees of freedom	N	$= n - 1 - 12$	932
Average M/P	M/P		1.005
Standard Deviation	$\sigma_{M/P}$		0.083
Corrected Standard Deviation	σ_N	$= \sigma_{M/P} \cdot [(n-1)/N]^{1/2}$	0.084
Owens Factor	K(N,0.95,0.95)		1.730
WRB-1 Design limit	$DNBR_L$	$= 1 / (1.005 - 1.730 \cdot 0.084)$	1.163

With a large database such as this, with 945 statepoints, correcting for the number of constants in the WRB-1 correlation has no significant effect, though technically it is more conservative to make the correction. Either way, the calculated DNBR limit results in a value of **1.17**.

Figures B.6-1 through B.6-4 display the performance of the M/P ratio and its distributions as a function of the pressure, mass velocity and quality. These plots show that there are no biases in the M/P ratio distribution, and that the performance of the WRB-1 CHF correlation is independent of the three variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes. These plots also show that all the tests in the WRB-1 database are within 3.6 standard deviations from the average. Figures B.6-5 through B.6-7 display the performance of the P/M ratio (i.e. the DNBR) against the major independent variables for the WRB-1 database. These plots also include a DNBR design limit line at 1.17. It can be seen that only 35 data points (3.70% of the database) are above the DNBR design limit, and that these data in excess of the limit are distributed over the entire range of the relevant variables.

Figure B.6-1: Measured vs. Predicted CHF for VIPRE-D/WRB-1 Database

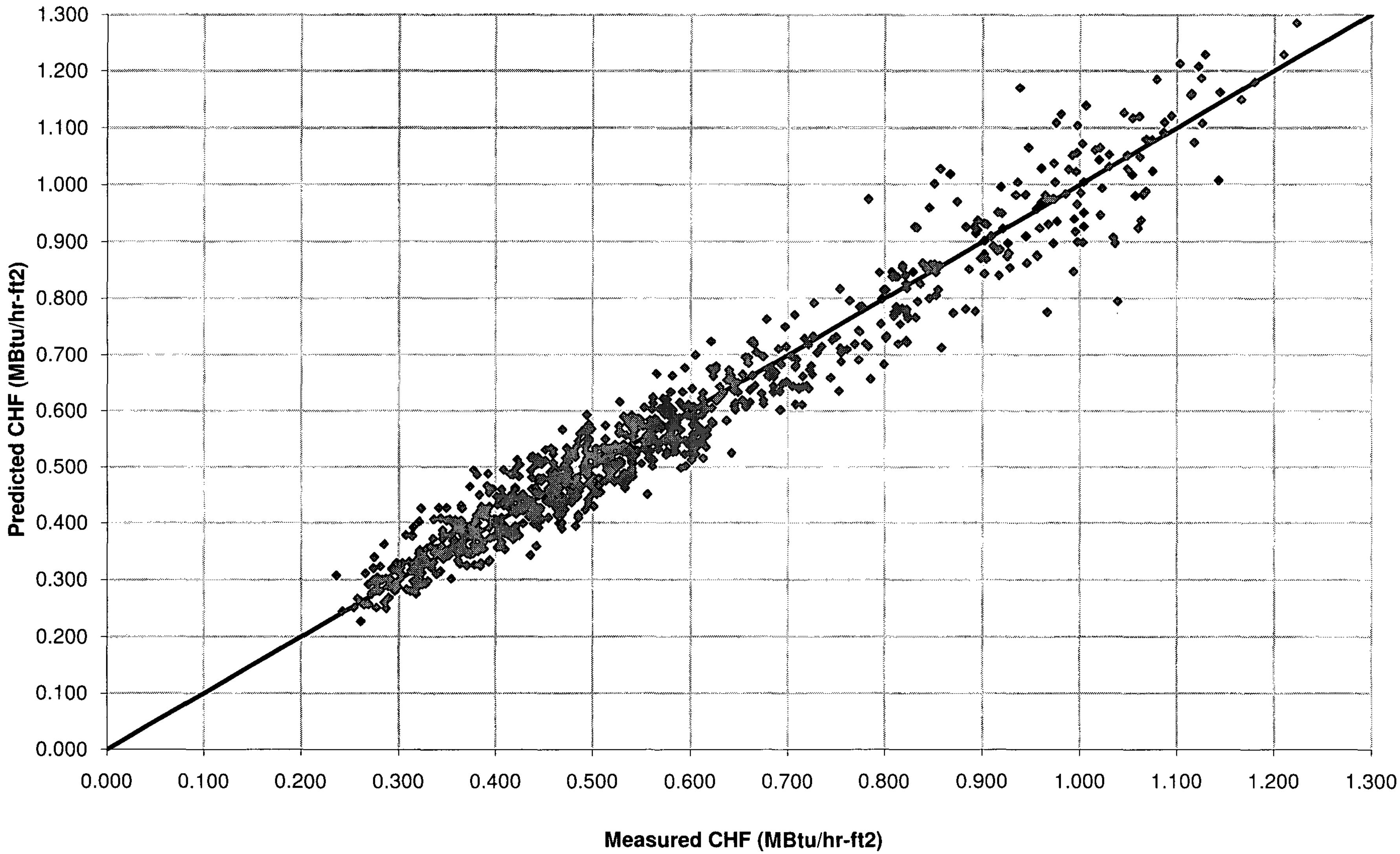


Figure B.6-2: M/P vs. Pressure for VIPRE-D/WRB-1 Database

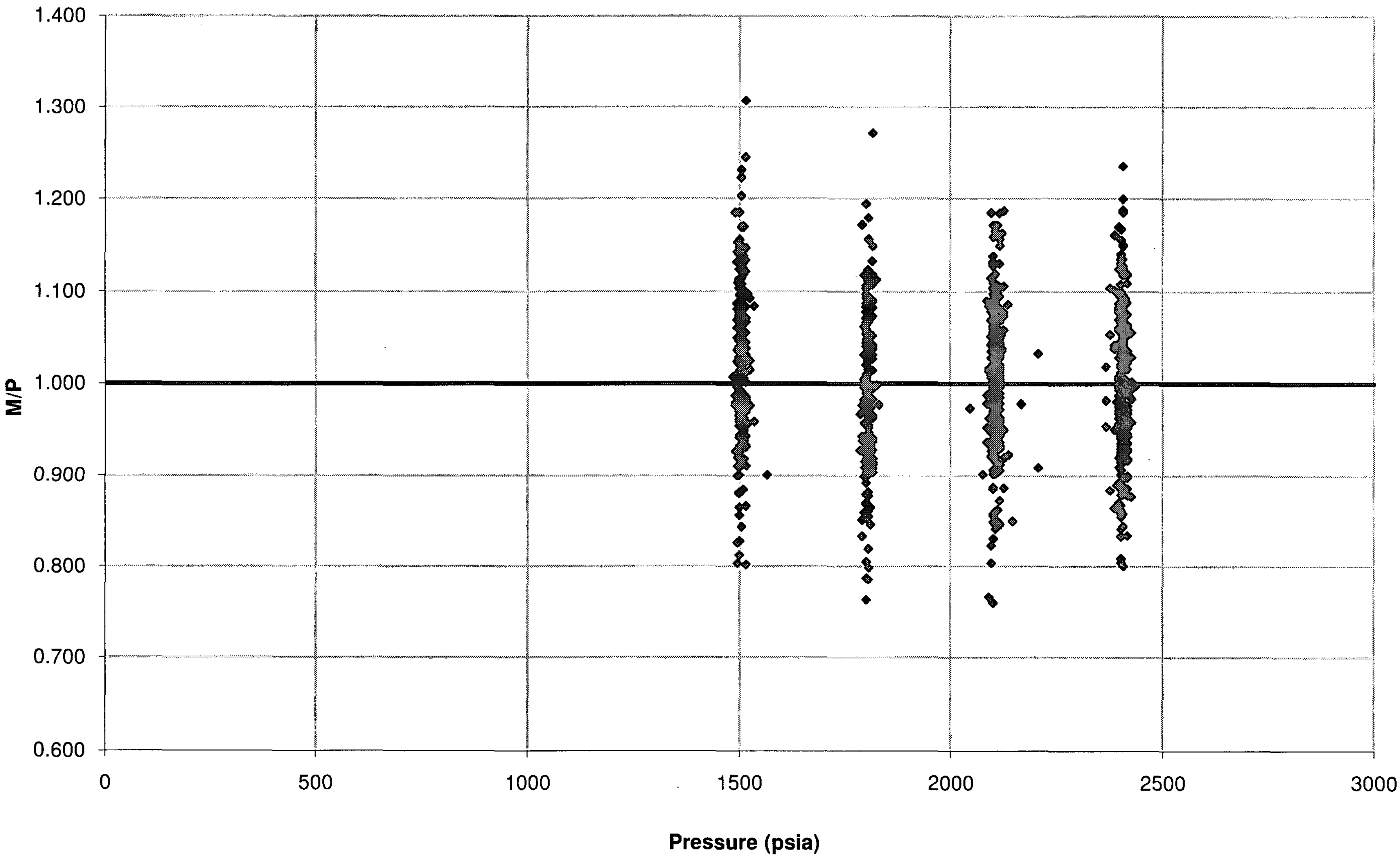


Figure B.6-3: M/P vs. Mass Velocity for VIPRE-D/WRB-1 Database

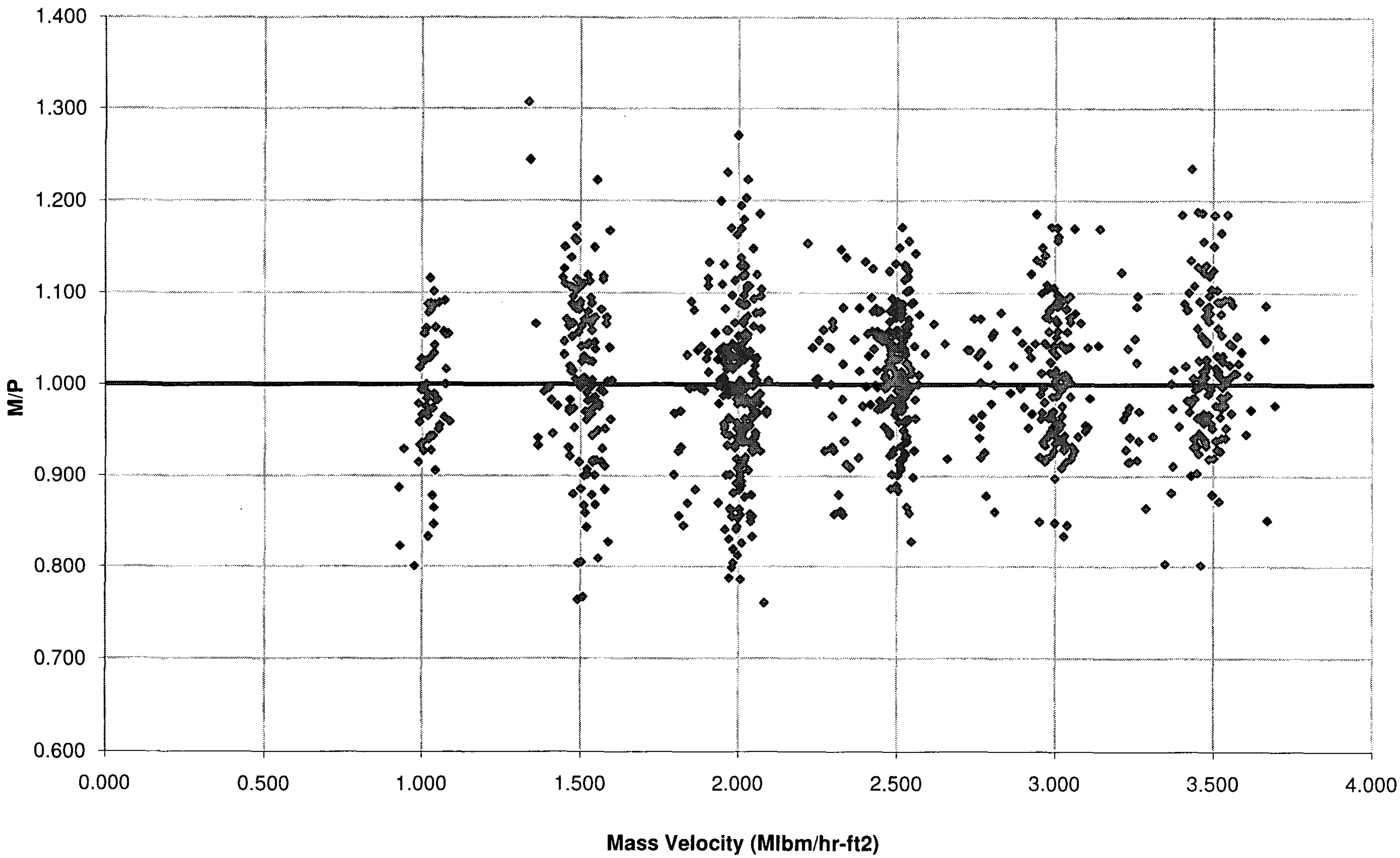


Figure B.6-4: M/P vs. Quality for VIPRE-D/WRB-1 Database

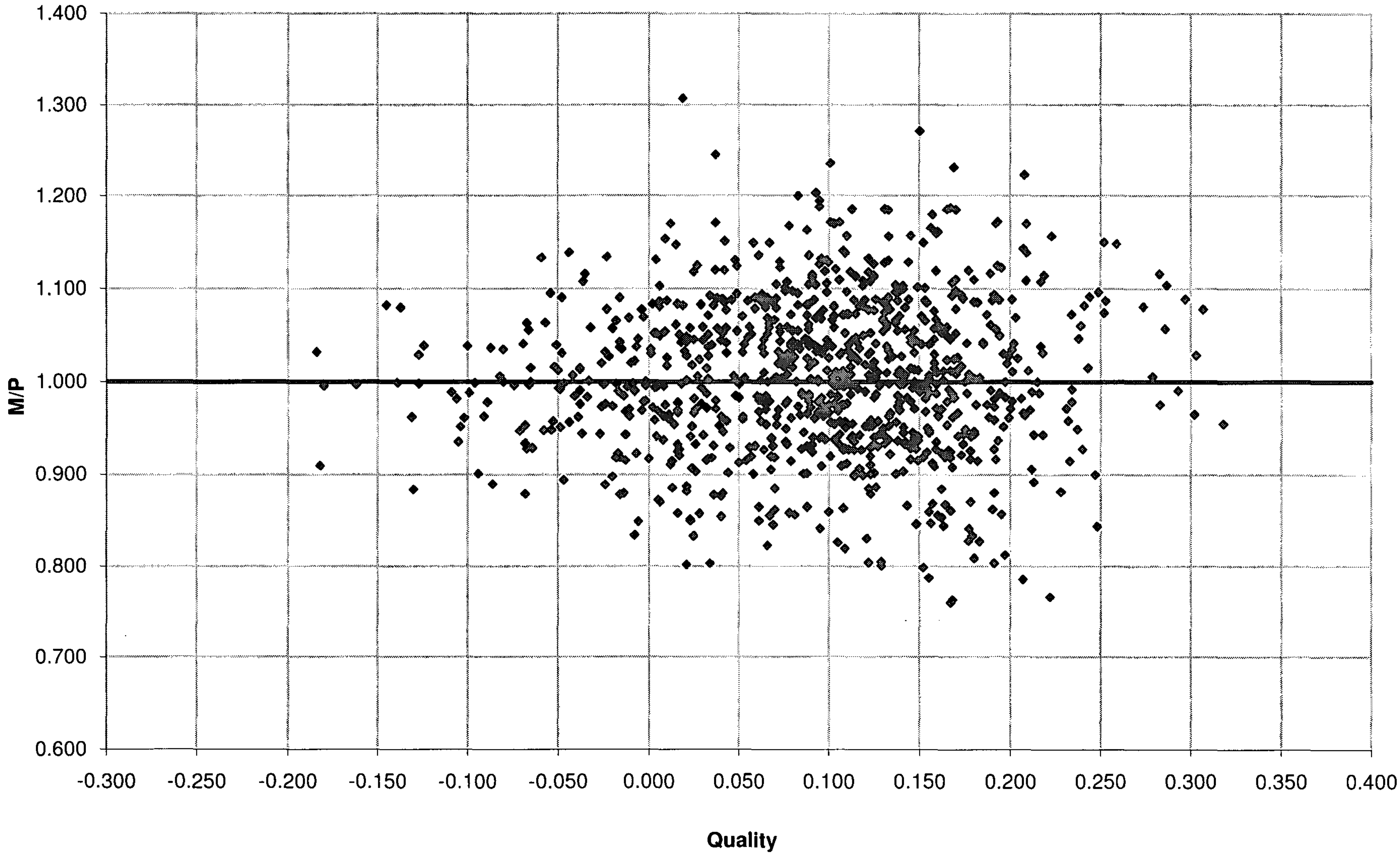


Figure B.6-5: DNBR vs. Pressure for VIPRE-D/WRB-1 Database

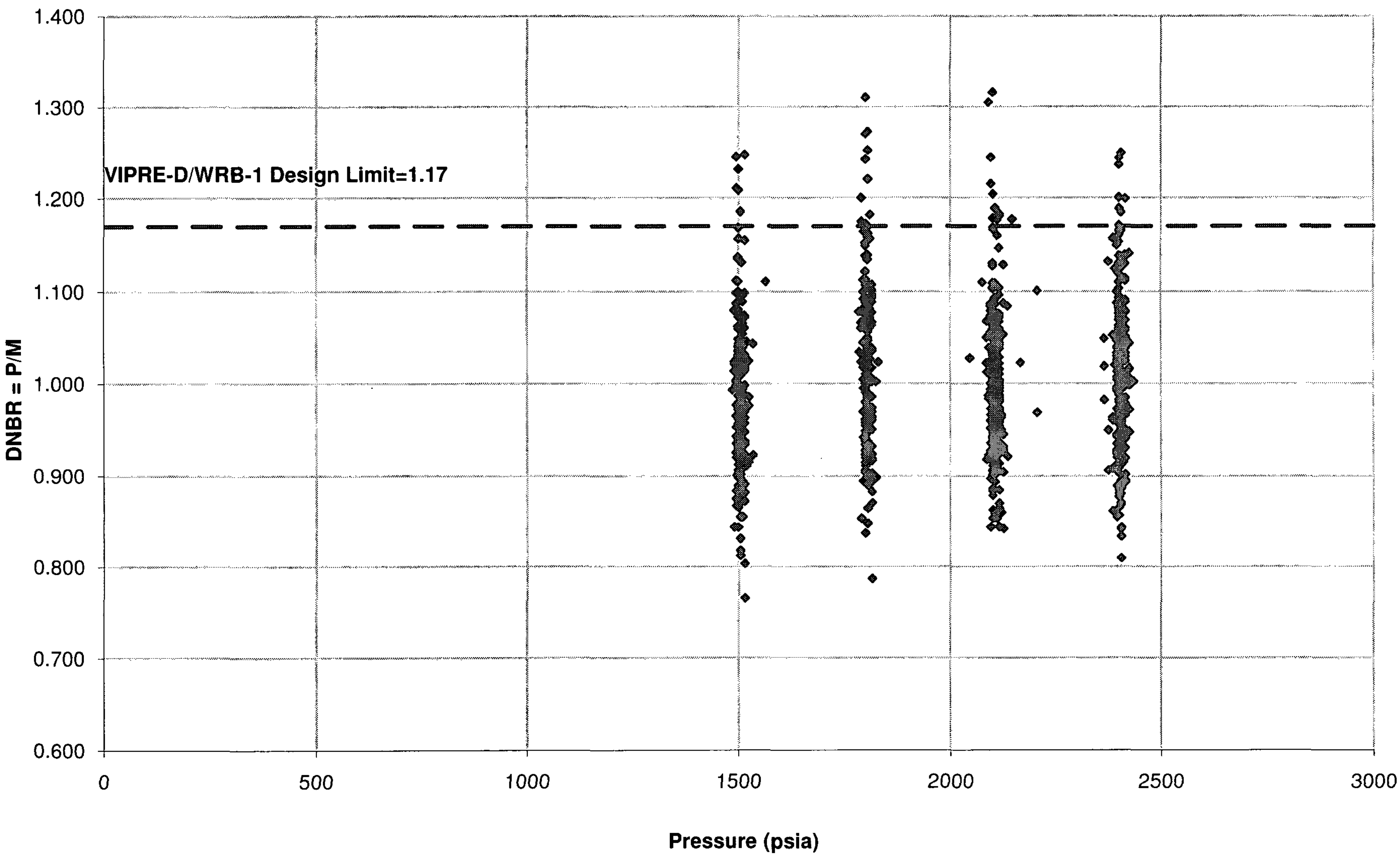


Figure B.6-6: DNBR vs. Mass Velocity for VIPRE-D/WRB-1 Database

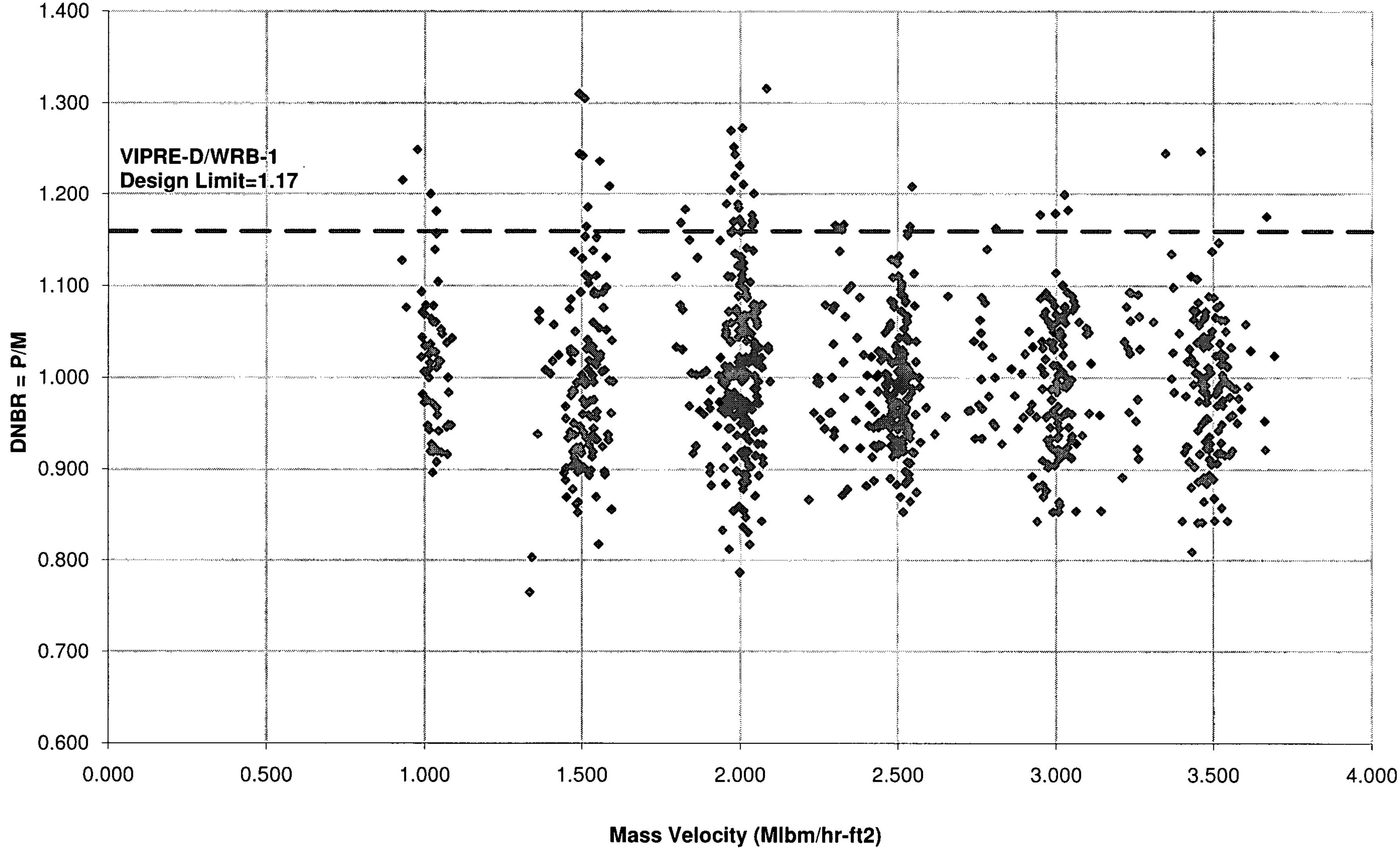
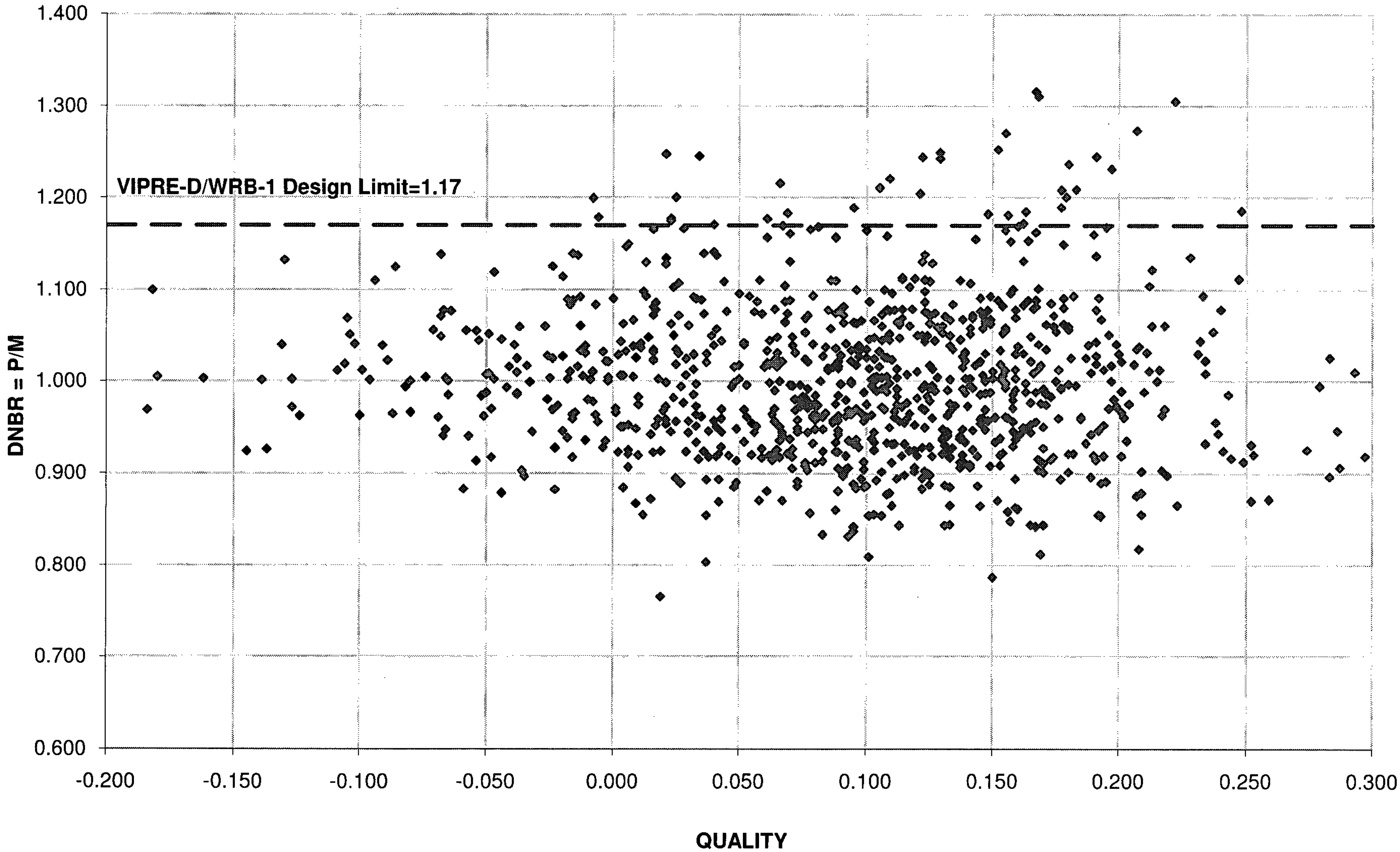


Figure B.6-7: DNBR vs. Quality for VIPRE-D/WRB-1 Database



B.7 BENCHMARK OF THE VIPRE-D/WRB-1 SUBCHANNEL MODEL

In Section 5 of the main body of this report, the Dominion VIPRE-D models created using the selections and modeling guidelines described in Section 4 in the main body of this report provided close comparison to the Framatome ANP LYNXT code, which is a USNRC approved subchannel code. This section in Appendix B demonstrates that the Dominion VIPRE-D models created using the selections and modeling guidelines described in Section 4 in the main body of the report provide close comparison to Dominion's COBRA code, which is also a USNRC approved subchannel code. This benchmark is provided as an example to demonstrate in sufficient detail the validity of the methodology discussed in the body of this report, and it is not meant to be linked to a specific plant or fuel product.

B.7.1 STEADY STATE APPLICATION

Dominion created a 19-channel model for Westinghouse 15x15 SIF fuel at SPS in accordance with the methodology described in Section 4 of this report. This VIPRE-D model of the 1/8th Surry core consists of 19 channels (15 subchannels and 4 lumped channels) and 20 rods, as shown in Figure B.7.1-1. The axial nodalization used in this model has been customized for Westinghouse 15x15 SIF fuel assemblies and contains 73 non-uniform axial nodes with typical node lengths of 2 inches and a maximum node length of less than 6 inches. The reference axial power profile (1.55 chopped cosine) was defined by the default function provided by the VIPRE-D code.

The Westinghouse SIF fuel assembly consists of 204 fuel rods with an outside diameter of 0.422 inches arranged in a 15x15 matrix with a pin pitch of 0.563 inches. The Westinghouse SIF fuel contains several advanced design features, such as mixing vane grids (MVG). The local FLCs used in this VIPRE-D 19-channel model were provided by Westinghouse from full-scale hydraulic tests.

VIPRE-D benchmark calculations were performed against the Dominion COBRA code and the COBRA 19-channel model created by Dominion to model SPS cores containing Westinghouse 15x15 SIF fuel assemblies. This benchmark uses 164 state points obtained from the UFSAR Chapter 14 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 COBRA. 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 164 statepoints cover the full range of conditions and axial offsets in the Surry UFSAR Chapter 14 evaluations (except for MSLB that is discussed in Section B.7.2), and were specifically selected to challenge both the WRB-1 and W-3 CHF correlations (Table B.7.1-1).

This benchmark study showed an average deviation between VIPRE-D and COBRA of less than 0.6% in DNBR, with a maximum deviation of 3.75%. These results are well within the uncertainty typically associated with thermal-hydraulic codes, which has been quantified to be 5%

(Reference B9), and justify the model selections in Section 4. Figure B.7.1-2 shows graphically the performance of VIPRE-D versus COBRA for the 164 statepoints. The close comparison of VIPRE-D to COBRA 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 in the main body of this report (MSLB will be discussed in Section B.7.2).

Table B.7.1-1: Range of VIPRE-D / COBRA 164 Benchmark Statepoints

VARIABLE	RANGE
Pressure [psia]	1800 to 2483.2
Power [% of 2546 MWt]	53.4 to 144.5
Inlet Temperature [°F]	505.1 to 631.7
Flow [% of Minimum Measured Flow]	66.8 to 100
$F_{\Delta H}$	1.56 to 2.106
Axial Offset [%]	-76.6 to 32.2

Figure B.7.1-1. Typical Surry VIPRE-D 19-Channel Model
for Westinghouse 15x15 SIF Fuel Assemblies

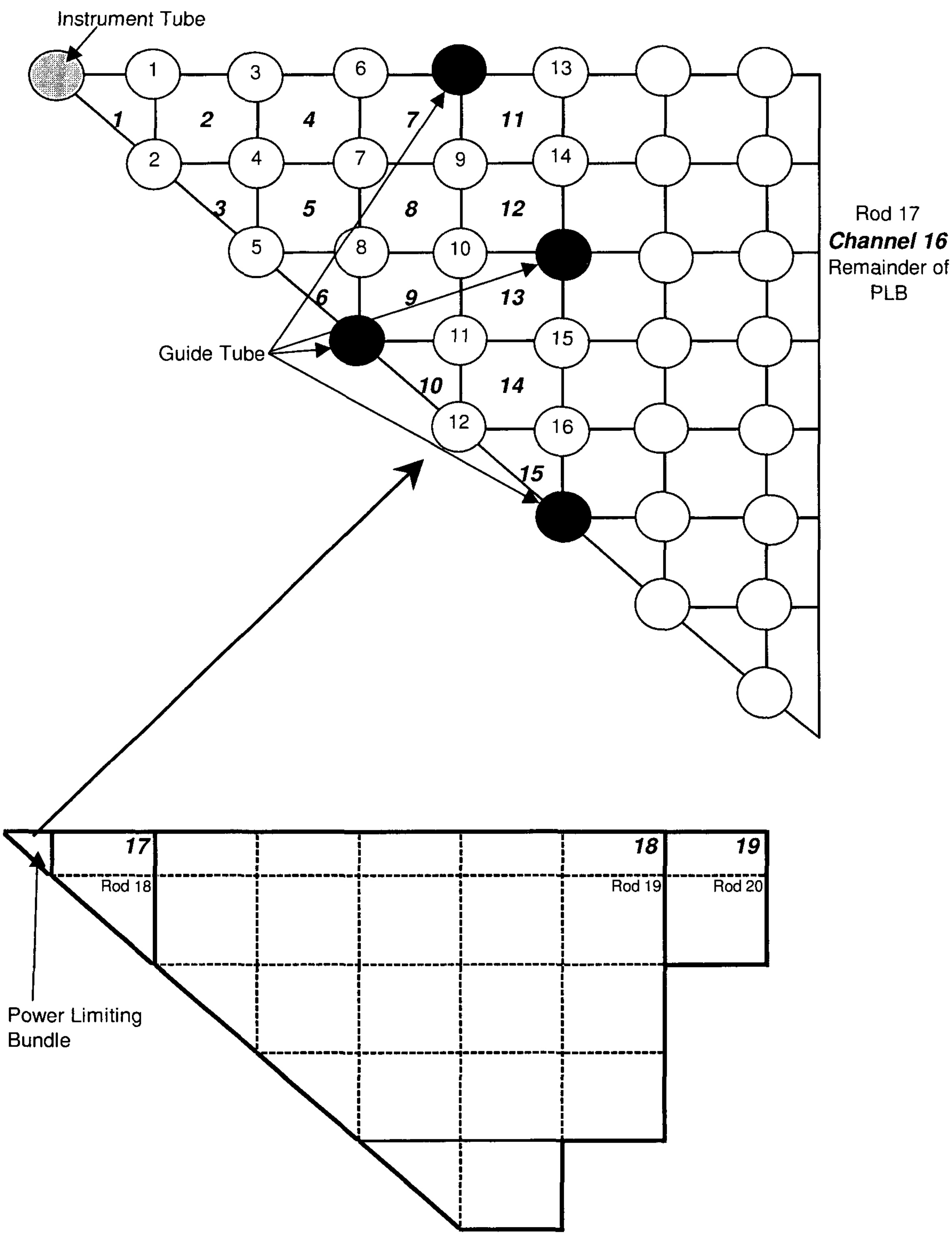
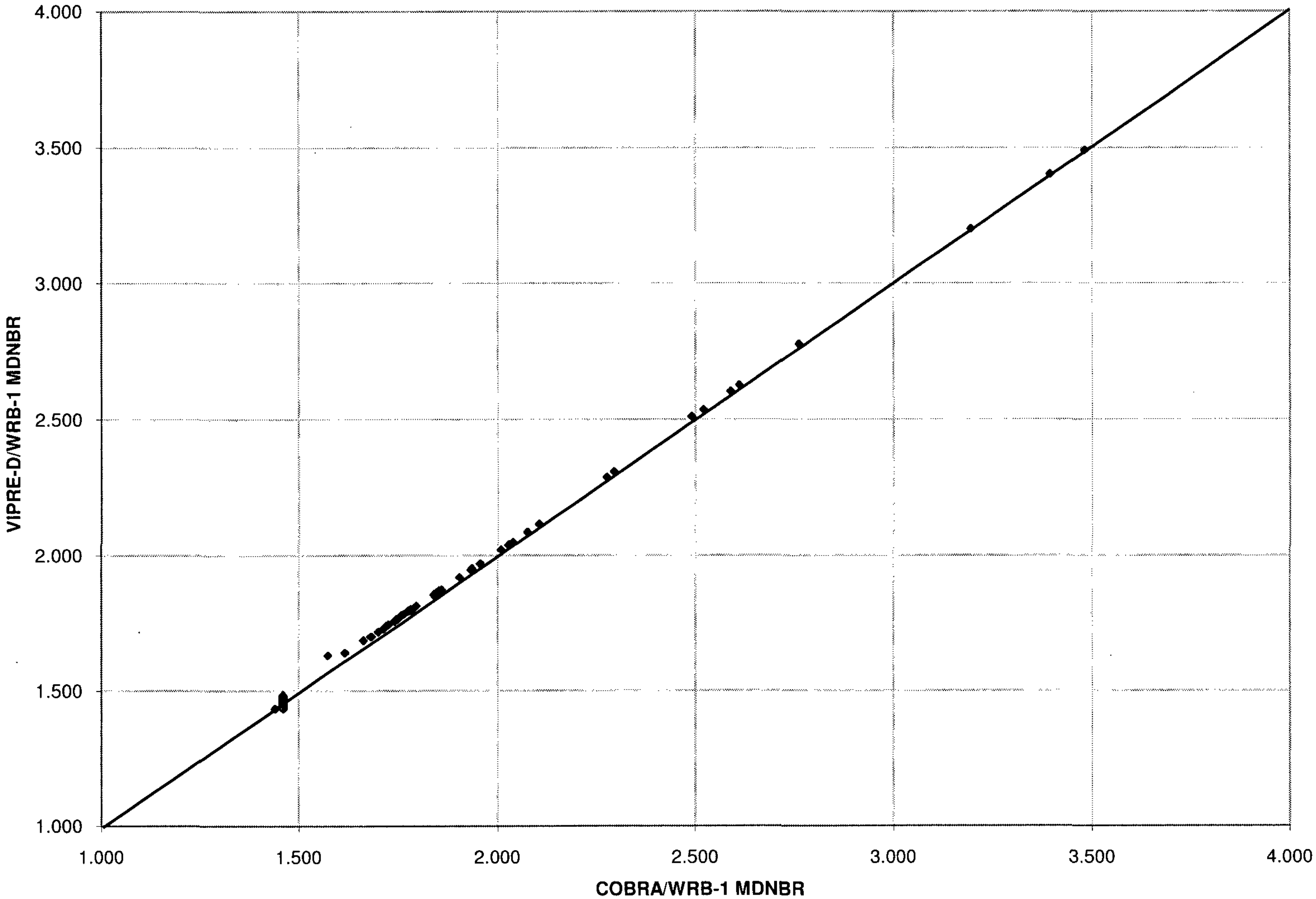


Figure B.7.1-2:VIPRE-D/WRB-1 vs. COBRA/WRB-1 for the 164 Analyzed Statepoints



B.7.2 MAIN STEAM LINE BREAK APPLICATION

The VIPRE-D 19-channel model discussed in section B.7.1 was also used to simulate the behavior of the core during a MSLB event, as it allows the modeling of the peaking and inlet boundary conditions in the fuel assemblies adjacent to the hot assembly. The two most limiting cases from a recent reload were evaluated with the VIPRE-D code, and their results compared to the COBRA results. The results obtained show a maximum deviation of 1.5% 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 19-channel model described here. It is important to note that both MSLB statepoints evaluated occurred at pressures below 1000 psia, and therefore the MDNBR was evaluated with the W-3 CHF correlation, and the appropriate correlation limit was 1.45 (Reference B6).

B.7.3 TRANSIENT APPLICATION

As demonstrated in Section 5.3 in the main body if this report, 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/WRB-1 transient capability was tested by performing two sample transient calculations. 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 the main body of this report. In both cases, the behavior of the VIPRE-D results was successfully compared to the behavior of the COBRA analysis of record in the UFSAR.

The first sample transient selected to perform this verification was the Feedwater Malfunction Transient (FWMAL). Forcing functions for the FWMAL transient were obtained from the SPS UFSAR. The length of the transient was 195 seconds, with a 0.5-second time step. COBRA analysis of record and VIPRE-D calculations exhibited similar behavior, and the MDNBR results show a maximum deviation of less that 0.4% (see Figure B.7.3-1).

The second sample transient selected to perform this verification was the Locked Rotor Transient (LOCROT). Forcing functions for the LOCROT transient were obtained from the SPS UFSAR. The length of the transient was 9.5 seconds, with a 0.025-second time step. COBRA analysis of record and VIPRE-D calculations exhibited similar behavior, and the MDNBR results show a maximum deviation of less that 1.6% (see Figure B.7.3-2).

The transient analyses demonstrate that VIPRE-D/WRB-1 is capable of performing stable transient calculations and the results obtained are essentially the same as the COBRA/WRB-1 results documented in the SPS UFSAR.

Figure B.7.3-1: VIPRE-D FWMAL Transient Sample Calculation Results

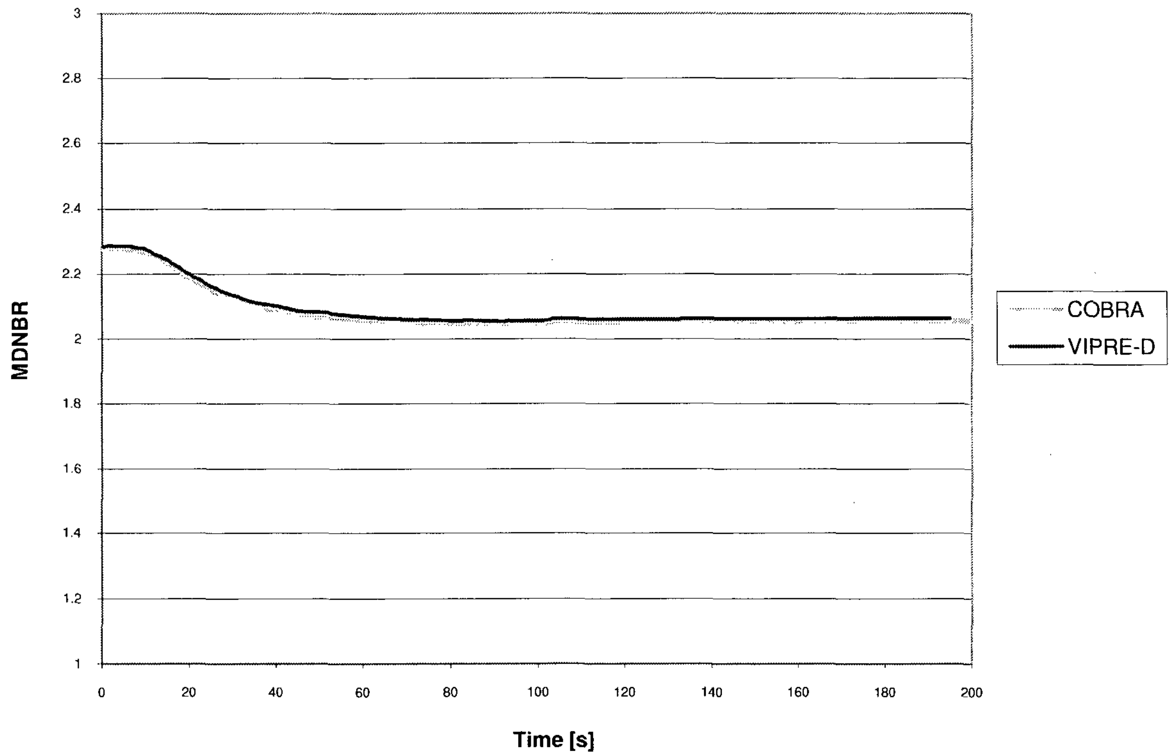
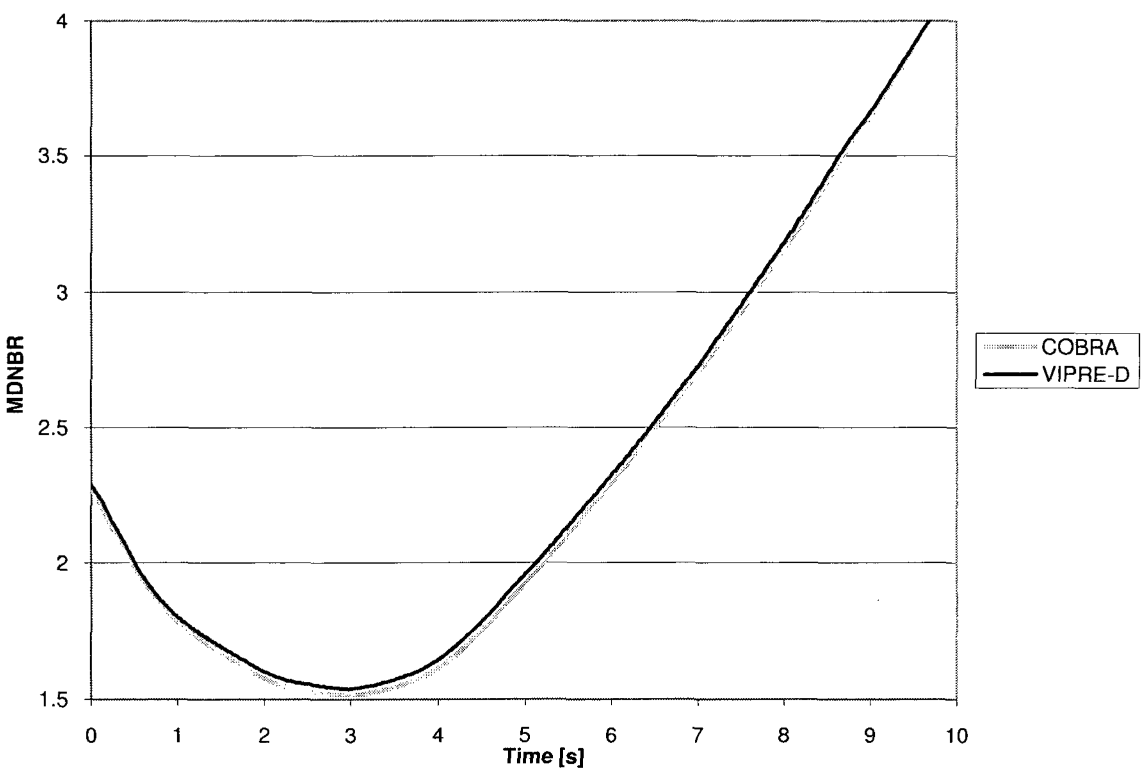


Figure B.7.3-2: VIPRE-D LOCROT Transient Sample Calculation Results



B.8 CONCLUSIONS

The WRB-1 correlation has been qualified with Dominion's VIPRE-D computer code. Table B.8-1 summarizes the DNBR design limits for VIPRE-D/WRB-1 that yields a 95% non-DNB probability at a 95% confidence level. The limit of 1.17 from VIPRE-D is the same limit as found with three other, approved code packages: COBRA (Reference B3), THINC (Reference B1), and Westinghouse's version of VIPRE-01 (Reference B6). Westinghouse WRB-1 CHF correlation has been approved by the USNRC for use with Westinghouse 15x15 and 17x17 "R" grid type fuel, and with Westinghouse 14x14, 15x15 and 17x17 OFA-type fuel products.

Table B.8-1: DNBR Limits for WRB-1

WRB-1 CHF CORRELATION DESIGN LIMITS				
	Dominion VIPRE-D	Dominion COBRA	Westinghouse THINC	Westinghouse VIPRE-01
DNBR limit	**1.17**	1.17	1.17	1.17

Table B.8-2 summarizes the applicability and the ranges of validity for VIPRE-D/WRB-1, which are the same as those on page 2 of the Dominion COBRA SER in Reference B3.

Table B.8-2: Range of Validity for VIPRE-D/WRB-1

Pressure [psia]	1,440 to 2,490
Mass Velocity [Mlbm/hr-ft^2]	0.9 to 3.7
Thermodynamic Quality at CHF	≤ 0.30
Local Heat Flux [Mbtu/hr- ft^2]	≤ 1.0
Mixing Vane Grid Spacing [in]	> 13.0

Finally, extensive code benchmark calculations have confirmed that the VIPRE-D/WRB-1 models created using the modeling guidelines specified in Section 4 in the main body of this report produce essentially the same results as USNRC approved equivalent Dominion COBRA/WRB-1 models.

B.9 REFERENCES

B1. Technical Report, WCAP-8762-P-A, "New Westinghouse Correlation *WRB-1* for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids", F. E. Motley, et al., July 1984.

B2. Technical Report, EPRI NP-2609, "Parametric Study of CHF Data, Volume 3, Part 1; Critical Heat Flux Data," C. F. Fighetti, & D.G. Reddy, September 1982.

B3. Topical Report, VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," R.C. Anderson & N. P. Wolfhope, July 1990.

B4. Technical Report, "Tables for Normal Tolerance Limits, Sampling Plans, and Screening," R. E. Odeh and D. B. Owen, 1980.

B5. Technical Report, "Assessment of the Assumption of Normality (employing individual observed values)," American National Standards Institute, ANSI N15.15.1974.

B6. Technical Report, WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, P. Schueren, and A. Meliksetian, October 1999.

B7. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4," May 1, 1986.

B8. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), "Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, 'VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores,' (TAC No. M79498)," October 30, 1993.

B9. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," R. C. Anderson, June 1987.

B10. Technical Report, "Boiling Crisis and Critical Heat Flux," TID-25887, 1972.

B11. Letter from A.C. Thadani (NRC) to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P, WCAP-9227-NP, Reactor Core Response to Excessive Secondary Steam Releases," 1989.

Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code

NUCLEAR ANALYSIS AND FUEL DEPARTMENT
DOMINION
RICHMOND, VIRGINIA

Approved by NRC Safety Evaluation dated April 22, 2009

CLASSIFICATION/DISCLAIMER

ABSTRACT

This appendix documents Dominion's qualification of the Westinghouse WRB-2M correlation with the VIPRE-D code. This qualification was performed against the same CHF experimental database used by Westinghouse to develop and license the correlation. This appendix summarizes the data evaluations that were performed to qualify the VIPRE-D/WRB-2M code/correlation pair, and to develop the corresponding DNBR design limit.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

CHF	Critical Heat Flux
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
FLC	Form Loss Coefficient
HTRF	Heat Transfer Research Facility at Columbia University
IFM	Intermediate Flow Mixer
LPD	Low Pressure Drop
M/P	Ratio of Measured-to-Predicted CHF
MIFM	Modified Intermediate Flow Mixer
MPS	Millstone Power Station
MVG	Mixing Vane Grid
NMVG	Non-Mixing Vane Grid
P/M	Ratio of Predicted-to-Measured CHF (equivalent to DNBR)
PWR	Pressurized Water Reactor
RFA	Robust Fuel Assembly
USNRC	US Nuclear Regulatory Commission

C.1 PURPOSE

Dominion currently uses the Westinghouse 17x17 Robust Fuel Assembly (RFA) fuel product at Millstone Power Station (MPS), Unit 3. The thermal-hydraulic analysis of this Westinghouse fuel product requires the use of the Westinghouse WRB-2M CHF correlation (Reference C1). In fact, the Westinghouse WRB-2M CHF correlation has been approved by the USNRC for use with the 17x17 RFA fuel design with or without the Intermediate Flow Mixer (IFM) grids (Reference C1).

To be licensed for use, a critical heat flux (CHF) correlation must be tested against experimental data that span the anticipated range of conditions over which the correlation will be applied. Furthermore, the population statistics of the database must be used to establish a departure from nucleate boiling ratio (DNBR) design limit such that the probability of avoiding departure from nucleate boiling (DNB) will be at least 95% at a 95% confidence level.

This addendum documents Dominion's qualification of the WRB-2M correlation with the VIPRE-D code. This qualification was performed against the data from the Columbia University Heat Transfer Research facility (HTRF) for the Modified Vantage 5H and Modified Vantage 5H/IFM fuel types (Reference C1). This is the same set of the Columbia-EPRI CHF database used by Westinghouse in the qualification of the WRB-2M correlation with the VIPRE-01 code (Reference C1). This addendum summarizes the data evaluations that were performed to qualify the VIPRE-D/WRB-2M code/correlation pair, and to develop the corresponding DNBR design limits for the correlation.

C.2 APPLICABILITY

Dominion intends to use the VIPRE-D/WRB-2M code/correlation for Westinghouse 17x17 RFA fuel products, with or without modified intermediate flow mixers (MIFM) in a PWR reactor. When evaluating this type of fuel outside of the range of validity of the WRB-2M CHF correlation, Dominion intends to use the VIPRE-D/W-3 code/correlation pair. W-3 is one of the CHF correlations contained in the USNRC approved generic version of VIPRE-01 (References C3 and C4), and it has already been approved for use with the VIPRE-D code (Reference C5).

The intended VIPRE-D/WRB-2M applications discussed in this addendum are consistent with the generic intended applications listed in the main body of this report (Section 2.0 in Reference C5). Also, more specifically, Dominion intends to use VIPRE-D/WRB-2M to analyze the transients delineated in Table 2.1-1 in Section 2.0 of the main body of this report (Reference C5). The qualification of the WRB-2M correlation with the VIPRE-D code has been performed following the modeling guidelines described in Section 4 of this report (Reference C5).

This Addendum is submitted to the USNRC for review and approval in order to meet the USNRC's requirement #2 listed in the VIPRE-01 SER, as outlined in Section 2.2 in the main body of this report (Reference C5).

C.3 DESCRIPTION OF THE WESTINGHOUSE WRB-2M CHF CORRELATION

In pressurized water reactor (PWR) cores, the energy generated inside the fuel pellets leaves the fuel rods at their surface in the form of heat flux, which is removed by the reactor coolant system flow. The normal heat transfer regime in this configuration is nucleate boiling, which is very efficient. However, as the capacity of the coolant to accept heat from the fuel rod surface degrades, a continuous layer of steam (a film) starts to blanket the tube. This heat transfer regime, termed film boiling, is less efficient than nucleate boiling and can result in significant increases of the fuel rod temperature for the same heat flux. Since the increase in temperature may lead to the failure of the fuel rod cladding, PWRs are designed to operate in the nucleate boiling regime and protection against operation in film boiling must be provided.

The heat flux at which the steam film starts to form is called CHF or the point of DNB. For design purposes, the DNBR is used as an indicator of the margin to DNB. The DNBR is the ratio of the predicted CHF to the actual local heat flux under a given set of conditions. Thus, DNBR is a measure of the thermal margin to film boiling and its associated high temperatures. The greater the DNBR value (above 1.0), the greater the thermal margin.

The CHF cannot be predicted from first principles, so it is empirically correlated as a function of the local thermal-hydraulic conditions, the geometry, and the power distribution measured in the experiments. Since a CHF correlation is an analytical fit to experimental data, it has an associated uncertainty, which is quantified in a DNBR design limit. A calculated DNBR value greater than this design limit provides assurance that there is at least a 95% probability at the 95% confidence level that a departure from nucleate boiling will not occur.

Correlations to predict the occurrence of CHF have undergone evolutions as nuclear fuel designs have changed. Westinghouse developed the WRB-1 CHF correlation for the prediction of DNB for Westinghouse fuel assemblies with mixing vane grids (MVGs). Subsequently, Westinghouse developed the WRB-2 CHF correlation for the prediction of DNB in Westinghouse fuel assemblies with MVGs and intermediate flow mixing grids (IFMs). More recently, Westinghouse has modified their nuclear fuel design to reduce fuel rod mechanical wear and to further improve thermal/hydraulic performance.

The new fuel design includes modified low pressure drop (LPD) mixing vane grids and modified intermediate flow mixing grids (MIFMs). This new fuel design is called the modified Vantage 5H and Modified Vantage 5H/IFM depending on whether the MIFMs have been included. (When the design includes the MIFM grids, it has also been referred to as the Robust Fuel Assembly (RFA)). CHF tests with the modified grids were conducted at the Columbia HTRF with and without control rod guide thimbles and with and without MIFM grids. Although the new data was successfully correlated by Westinghouse using the WRB-2 CHF correlation, a better correlation, the WRB-2M CHF correlation (a modification of the WRB-2 CHF correlation), was obtained by incorporation of a multiplier 'M' (Reference C1). The sensitivity of WRB-2M to other parameters, such as various power shapes is very similar to WRB-2. Thus, WRB-2M is applicable for 17x17 fuel with 0.374 inch OD rods, and Modified LPD grids with or without MIFM's. The range of applicable parameters is given in Table C.5-3. The WRB-2M correlation has been approved by the NRC (Reference C1).

The WRB-2M DNB correlation was developed from test bundles simulating the RFA fuel design with only a cosine axial power shape. As part of a scoping study for new grid designs, DNB tests were performed with the uniform axial power shape at the Columbia University test loop. Although the grid designs were not the same, the mixing vanes of the test bundles were similar to the RFA fuel design. When compared to the data from those tests, the WRB-2M measured-to-predicted (M/P) CHF average ratio was lower than 1.0. No significant trend in M/P was observed with respect to key parameters such as local flow rate, local equilibrium quality, and pressure. Based on this comparison with the test results, Westinghouse decided to adjust the DNB predictions for the WRB-2M DNB correlation. The adjustment factor does not constitute a change in the methodology as described in the licensing basis. The NRC staff has reviewed the adjustment factor and its consequences and found it acceptable (Reference C6).

The W-3 correlation is used when conditions are outside the range of the WRB-2M DNB correlation. Specifically, the W-3 correlation is applied to the lower portion of the fuel assemblies in the rod withdrawal from subcritical event because of the bottom peaked axial power distribution assumed, and in the steam line break event because of the low pressures involved. The W-3 correlation with a correlation limit of 1.30 is used below the fuel assembly first mixing vane grid for the rod withdrawal from subcritical event. For the steam line break event, the W-3 correlation is used with a correlation limit of 1.45 in the pressure range of 500 to 1000 psia and 1.30 for pressures above 1000 psia (Reference C8). The Westinghouse W-3 CHF correlation is described on page 10 in Reference C7.

C.4 DESCRIPTION OF THE VIPRE-D/WRB-2M DATABASE AND TEST ASSEMBLIES

The WRB-2M CHF correlation was developed from CHF data obtained at the Columbia University HTRF using full-scale, electrically heated rod bundle test sections (Reference C1). The Dominion qualification of WRB-2M in VIPRE-D was performed against the same test data from the Columbia-EPRI CHF database for Westinghouse 17x17 fuel. Dominion used the CHF experimental data used by Westinghouse to develop the WRB-2M correlation. No data point was deleted or excluded.

The HTRF test assemblies had a 5x5 geometry, thus the test assemblies used by Dominion to qualify the VIPRE-D/WRB-2M code/correlation pair have a 5x5 geometry. These 5x5 test bundles have essentially a 17x17 subchannel geometry (Reference C1). Table C.4-1 provides a summary of the key information about each test.

Table C.4-1: Summary of CHF Tests

TEST	MATRIX	AXIAL HEAT FLUX SHAPE	PIN OD / GUIDE TUBE OD [inches]	HEATED LENGTH [inches]	MIFM grids?	NUMBER OF TESTS
A-1	5 x 5	Non-Uniform	0.374 / -	168	Yes	66
A-2	5 x 5	Non-Uniform	0.374 / 0.474	168	Yes	77
A-3	5 x 5	Non-Uniform	0.374 / 0.474	168	No	70
A-4	5 x 5	Non-Uniform	0.374 / -	168	No	28

C.5 VIPRE-D/WRB-2M RESULTS AND COMPARISON TO VIPRE-01

Reference C1 describes the mathematical model for each separate test section by providing the bundle and cell geometry, the rod radial peaking values, the rod axial flux shapes, the types, axial locations and form losses associated to the spacer grids, as well as the thermocouple locations. Reference C1 also provides the data for each CHF observation within a test, including power, flow, inlet temperature, pressure and CHF axial location.

Each test section was modeled for analysis with the VIPRE-D thermal-hydraulic computer code as a full assembly model following the modeling methodology discussed in Section 4 in the main body of this report. For each set of bundle data, VIPRE-D produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc.) at every axial node along the heated length of the test section. The ratio of measured-to-predicted CHF (M/P) is the variable that is normally used to evaluate the thermal-hydraulic performance of a code/correlation pair. The measured CHF is the local heat flux at a given location, while the predicted CHF is calculated by the code using the WRB-2M CHF correlation. The ratio of these two values provides the M/P ratio, which is the inverse of the DNB ratio. M/P ratios are frequently used to validate CHF correlations instead of DNB ratios, because their distribution is usually a normal distribution, which simplifies their manipulation and statistical analysis.

This section summarizes the VIPRE-D results and the associated significant statistics. This section also shows the variation of the M/P ratio with each independent variable to demonstrate that there are no biases in the data. Finally, it provides the VIPRE-D overall statistics for the WRB-2M tests and generates the DNBR design limit for the WRB-2M CHF correlation with VIPRE-D.

The WRB-2M correlation was developed by Westinghouse by correlating the CHF experimental results obtained in the tests as described in Reference C1. Westinghouse also used these test data to calculate a DNBR design limit of 1.14 for the WRB-2M correlation (Reference C1). Dominion used these experimental data, as described in section C.4, to develop the VIPRE-D/WRB-2M DNBR limit. Table C.5-1 summarizes the relevant statistics for each test, and calculates the aggregate statistics for the entire set of data.

Table C.5-1: Summary of VIPRE-D Results

Test	Number of Tests	M/P Ratio Average	M/P Ratio STDEV	M/P Ratio Max	M/P Ratio Min
A-1	66	1.0178	0.0789	1.2114	0.8287
A-2	77	0.9834	0.0538	1.1017	0.8614
A-3	70	1.0144	0.0559	1.1444	0.8954
A-4	28	0.9731	0.0490	1.1002	0.8688
Thimble	147	0.9982	0.0568	1.1444	0.8614
Typical	94	1.0045	0.0740	1.2114	0.8287
With MIFM	143	0.9993	0.0685	1.2114	0.8287
Without MIFM	98	1.0026	0.0570	1.1444	0.8688
All Results	241	1.0006	0.0640	1.2114	0.8287

One-sided tolerance theory (Reference C2) is used for the calculation of the VIPRE-D/WRB-2M DNBR design limit. This theory allows the calculation of a DNBR limit so that, for a DNBR equal to the design limit, DNB will be avoided with 95% probability at a 95% confidence level.

First, it is necessary to verify that the overall distribution for the M/P ratios is a normal distribution, because all the statistical techniques used below assume that the original data distribution is normal. To evaluate if the distribution is normal, the D' normality test was applied. A value of D' equal to 1047.04 was obtained for the VIPRE-D/WRB-2M database. This D' value is within the range of acceptability for 241 data points with a 95% confidence level (1038.60 to 1066.75)[1]. Thus, it is concluded that the M/P distribution for the VIPRE-D/WRB-2M database is indeed normal. Based on the results listed in Table C.5-1, the deterministic DNBR design limit can be calculated as:

$$DNBR_L = \frac{1.0}{M/P - K_{N,C,P} \cdot \sigma_{M/P}} \qquad [C.5.1]$$

where

- M/P = average measured to predicted CHF ratio
- $\sigma_{M/P}$ = standard deviation of the measured to predicted CHF ratios of the database
- $K_{N,C,P}$ = one-sided tolerance factor based on N degrees of freedom, C confidence level, and P portion of the population protected. This number can be obtained from Table 1.4.4 of Reference C2.

[1] From Table 5 in Reference C9
D' Lower Limit (241) [P = 0.025] = 1038.60
D' Upper Limit (241) [P = 0.975] = 1066.75

Normally, the number of degrees of freedom would be the total number of data minus one. However, because Westinghouse used these experimental data to correlate the 6 constants that appear in the WRB-2M correlation, the total number of degrees of freedom must be corrected to account for this. In addition, the standard deviation of the database needs to be corrected accordingly to account for this reduced number of degrees of freedom:

$$N = n - 1 - 6 \qquad [C.5.2]$$
$$\sigma_N = \sigma_{M/P} \cdot [(n - 1) / N]^{1/2}$$

Then, the DNBR design limit for the VIPRE-D and the WRB-2M correlation can be calculated as shown in Table C.5-2.

Table C.5-2: Statistical Analysis of WRB-2M Design Limit

Number of data	n		241
Degrees of freedom	N	= n - 1 - 6	234
Average M/P	M/P		1.0006
Standard Deviation	$\sigma_{M/P}$		0.0640
Corrected Standard Deviation	σ_N	$= \sigma_{M/P} \cdot [(n - 1) / N]^{1/2}$	0.0648
Owens Factor	K(N,0.95,0.95)		1.8170
WRB-2M Design limit	$DNBR_L$	$= 1 / (M/P - K(N,0.95,0.95) \cdot \sigma_N)$	1.1327

Even though this is not a large database, correcting for the number of constants in the WRB-2M correlation has no significant effect, and it is more conservative to make the correction. The calculated DNBR limit results in a value of **1.14**. This is the same number reported by Westinghouse in Reference C1 and has been approved by the NRC.

Table C.5-3 summarizes the ranges of validity for the VIPRE-D/WRB-2M correlation. These ranges, are identical to those submitted by Westinghouse and already approved by the NRC (Reference C1).

Table C.5-3: Range of Validity for WRB-2M

	VIPRE-D
Pressure [psia]	1495 to 2425
Mass Velocity [Mlbm/hr-ft^2]	0.97 to 3.1
Thermodynamic Quality at CHF	-0.1 to 0.29

Figures C.5-1 through C.5-4 display the performance of the M/P ratio, and its distributions as a function of the pressure, mass velocity and quality. The objective of these plots is to show that there are no biases in the M/P ratio distribution, and that the performance of the WRB-2M correlation is independent of the three independent variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes. These plots also show that all the tests in the WRB-2M database are within 3.5 standard deviations from the average. Figures C.5-5 through C.5-7 display the performance of the P/M ratio (i.e., the DNBR) against the major independent variables for the WRB-2M database. These plots also include the DNBR design limit line. It can be seen that only six data points (2.49% of the database) are above the DNBR design limit, and that these data in excess of the limit are distributed over the variable ranges tested.

A more formal determination of the lack of bias of the average M/P ratio can be done using the analysis of variance test (ANOVA) shown in Table C.5-4. ANOVA tests are normally applied to highly controlled situations, but they can be somewhat useful in CHF testing and correlation. However, the ANOVA test cannot be used as the sole measure of the performance of a CHF correlation, but it would indicate an extremely bad mismatch (with a very large F statistic). The variables analyzed were pressure, quality, mass velocity and test cell type. The ANOVA results for VIPRE-D/WRB-2M slightly exceed the critical values of F for pressure and quality, but other comparisons prove the hypothesis that all the groups belong to the same distribution; i.e., that there is no bias of the results regarding the analyzed variables. Furthermore, when looking at the figures in this section, there does not appear to be any trend or bias in the data. Therefore, it can be concluded that the WRB-2M M/P ratio database is independent of the pressure, quality, and mass velocity.

Table C.5-4: M/P CHF Performance by Independent Variable Grouping of the WRB-2M Database at 95% Confidence Level

Grouping	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Pressures					
Below 1575 psia	29	0.9831	0.0610	1.1153	0.8746
1575 - 1850 psia	67	0.9967	0.0666	1.1327	0.8614
1850 – 2250 psia	74	1.0210	0.0680	1.2114	0.9058
Above 2250 psia	71	0.9903	0.0534	1.1002	0.8287
	$F_{distribution}$ = 4.0812		$F_{critical}(3,237)$ = 2.6427		
Analysis by Qualities					
Below 5%	42	1.0107	0.0521	1.1193	0.8688
5% to 10%	71	1.0126	0.0581	1.1302	0.8862
10% to 15%	66	1.0051	0.0633	1.1744	0.8794
15% to 20%	46	0.9718	0.0628	1.1444	0.8287
Above 20%	16	0.9858	0.0961	1.2114	0.8336
	$F_{distribution}$ = 3.6650		$F_{critical}(4,236)$ = 2.4099		
Analysis by Mass Velocities					
Below 1.25 Mlbm/hr-ft^2	32	0.9915	0.0645	1.1444	0.8688
1.25 – 1.75 Mlbm/hr-ft^2	75	1.0055	0.0642	1.1302	0.8614
1.75 – 2.25 Mlbm/hr-ft^2	74	1.0023	0.0602	1.1744	0.8287
2.25 – 2.75 Mlbm/hr-ft^2	51	0.9911	0.0637	1.1264	0.8336
2.75 – 3.25 Mlbm/hr-ft^2	9	1.0323	0.0879	1.2114	0.9063
	$F_{distribution}$ = 1.1176		$F_{critical}(4,236)$ = 2.4099		
Analysis by Geometry Type					
With MIFM Grids	143	0.9993	0.0685	1.2114	0.8287
Without MIFM Grids	98	1.0026	0.0571	1.1444	0.8688
	$F_{distribution}$ = 0.1580		$F_{critical}(1,239)$ = 3.8807		
Analysis by Thimble vs. Typical					
Thimble	147	0.9982	0.0568	1.1444	0.8614
Typical	94	1.0045	0.0740	1.2114	0.8287
	$F_{distribution}$ = 0.5546		$F_{critical}(1,239)$ = 3.8807		
All Data WRB-2M					
All Data	241	1.0006	0.0640	1.2114	0.8287

Figure C.5-1: Measured vs. Predicted CHF for VIPRE-D/WRB-2M Database

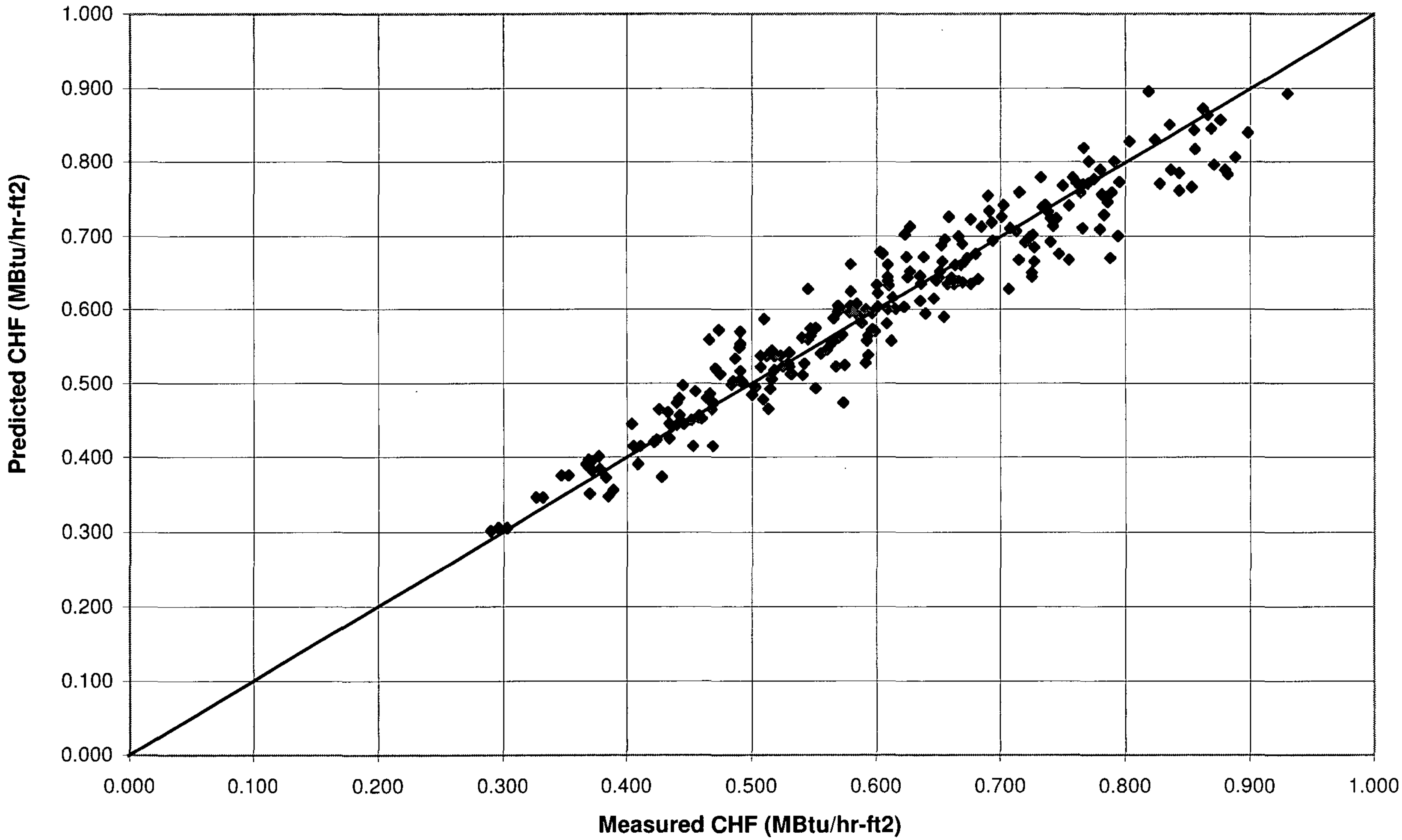
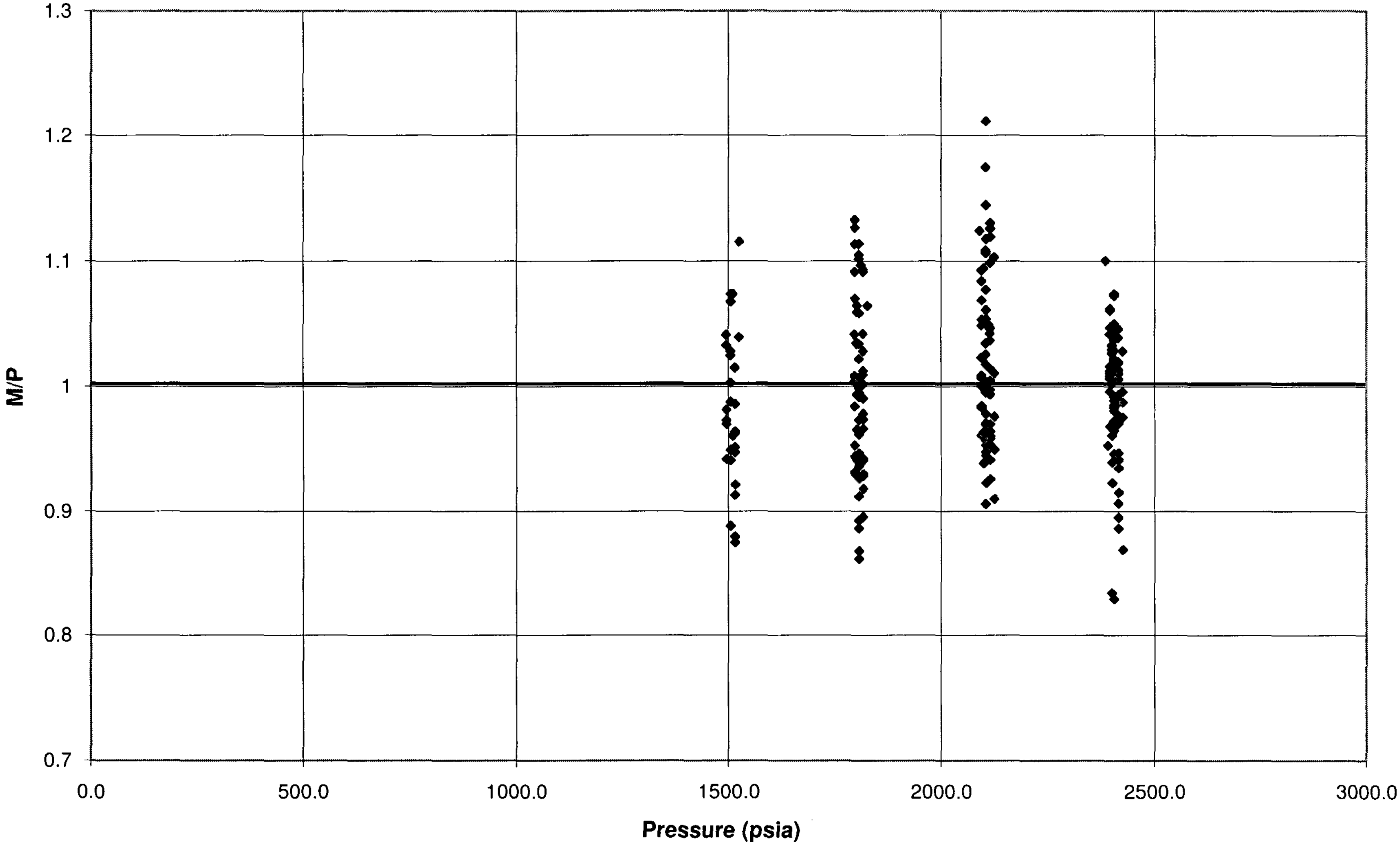


Figure C.5-2: M/P vs. Pressure for VIPRE/WRB-2M Database



[redacted]

Figure C.5-3: M/P vs. Mass Velocity for VIPRE-D/WRB-2M Database

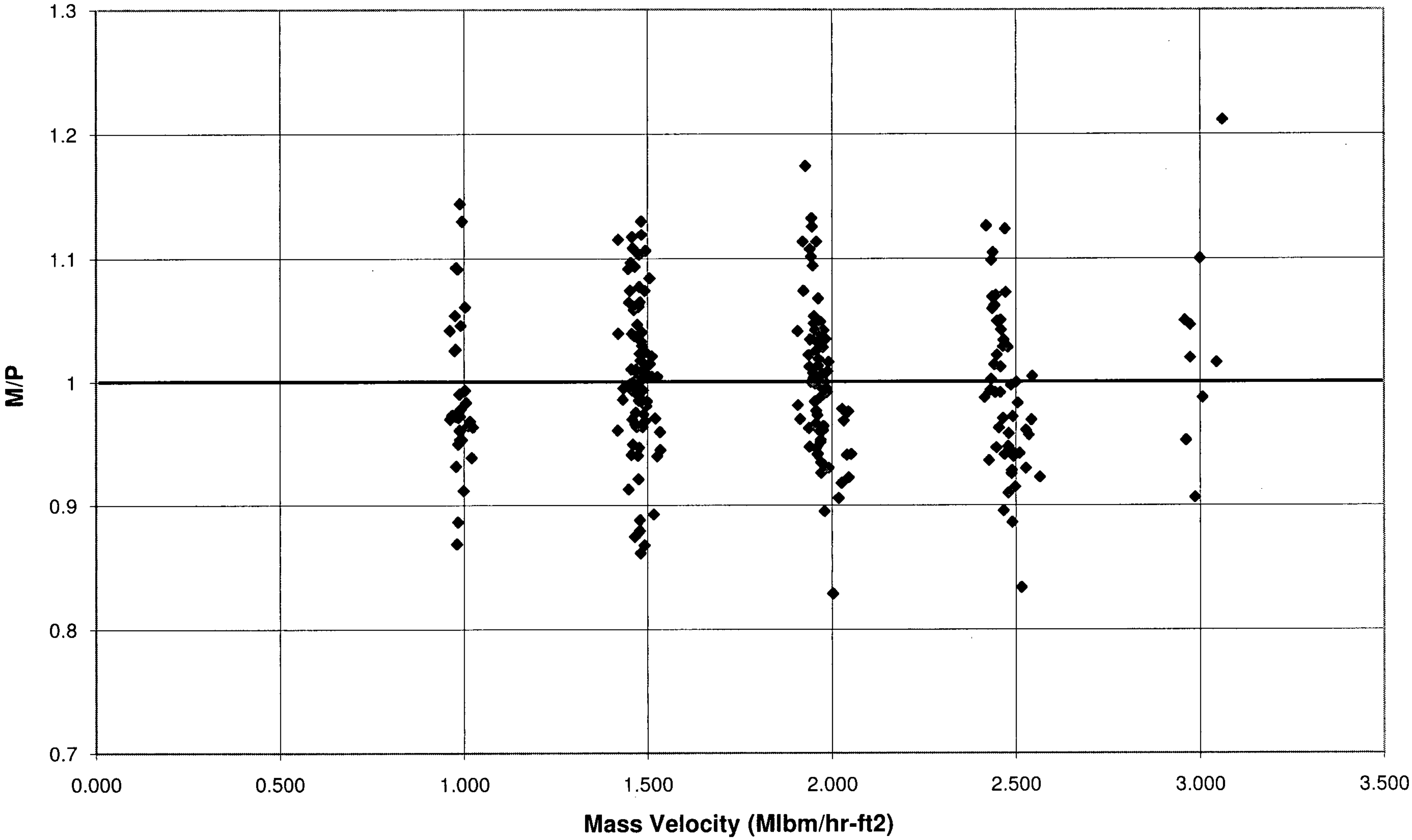


Figure C.5-4: M/P vs. Quality for VIPRE-D/WRB-2M Database

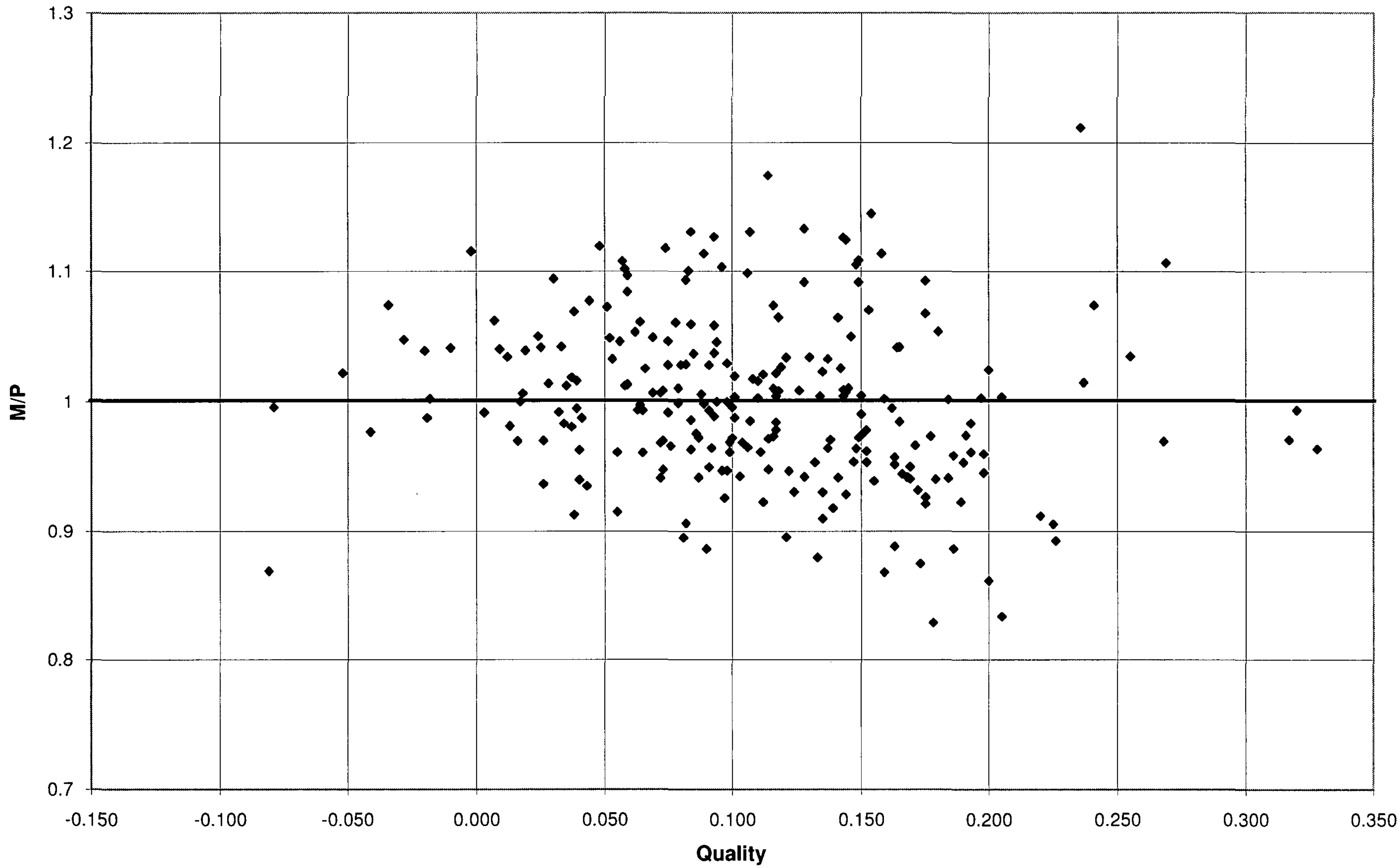


Figure C.5-5: DNBR vs. Pressure for VIPRE-D/WRB-2M Database

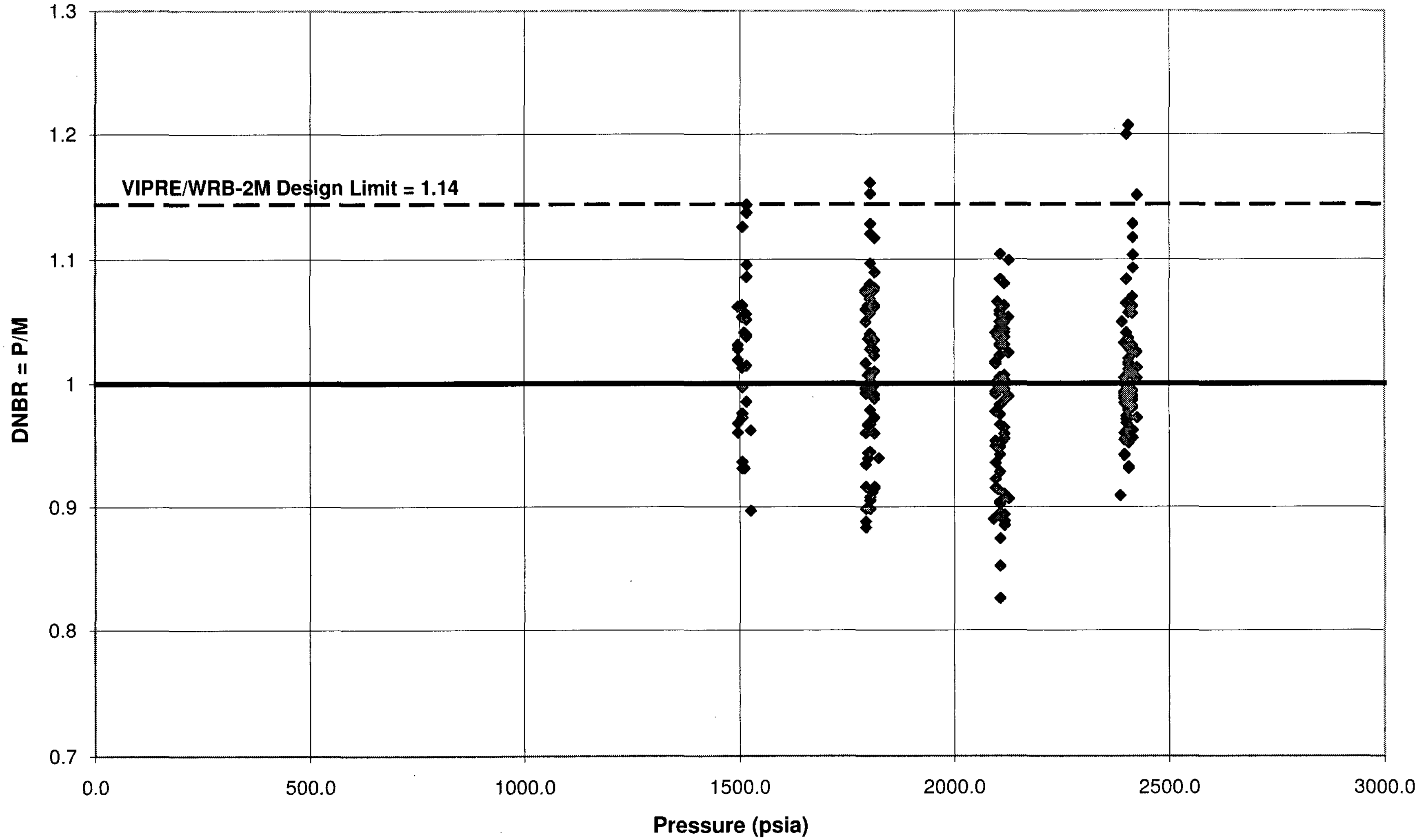


Figure C.5-6: DNBR vs. Mass Velocity for VIPRE-D/WRB-2M Database

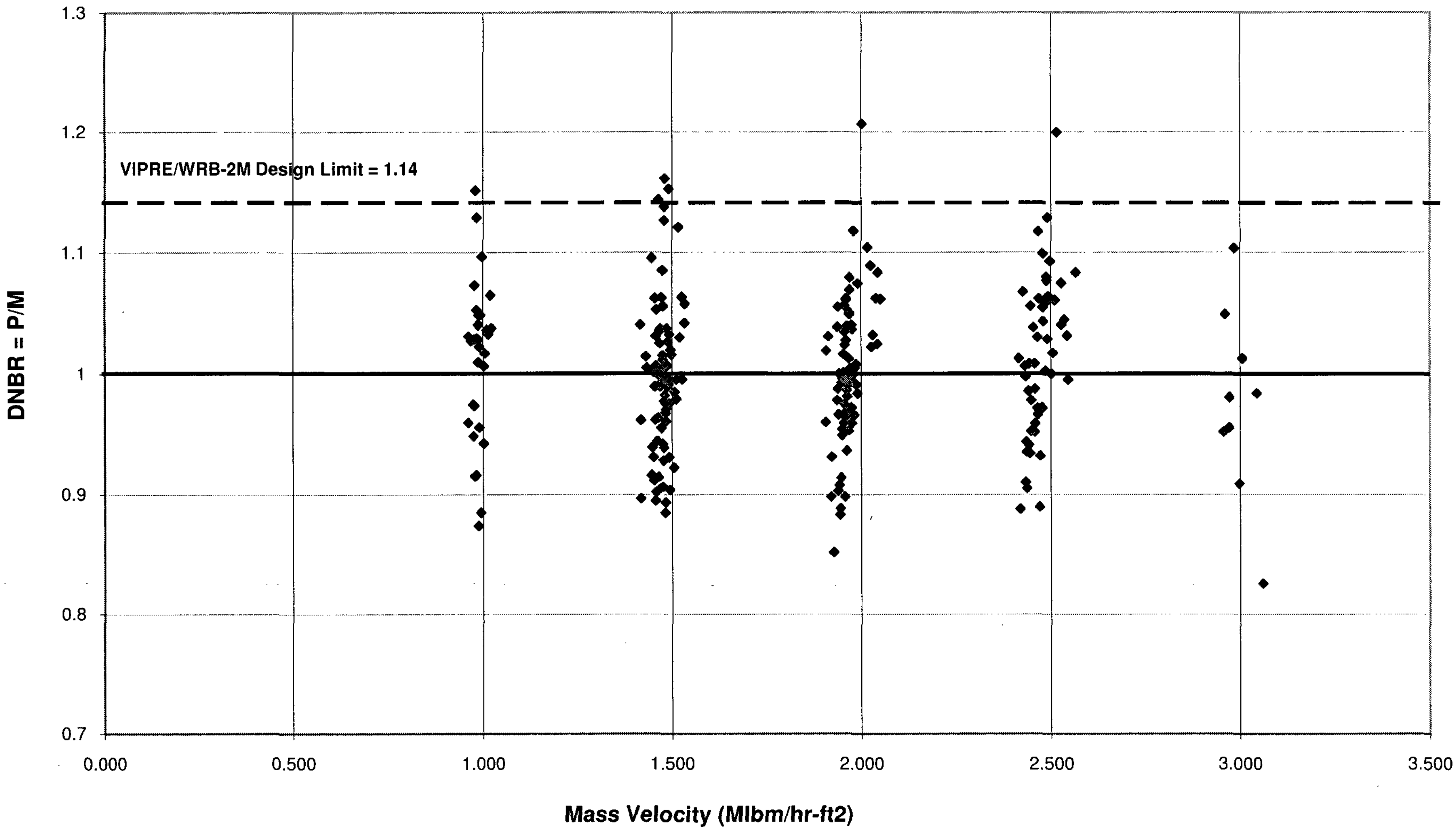
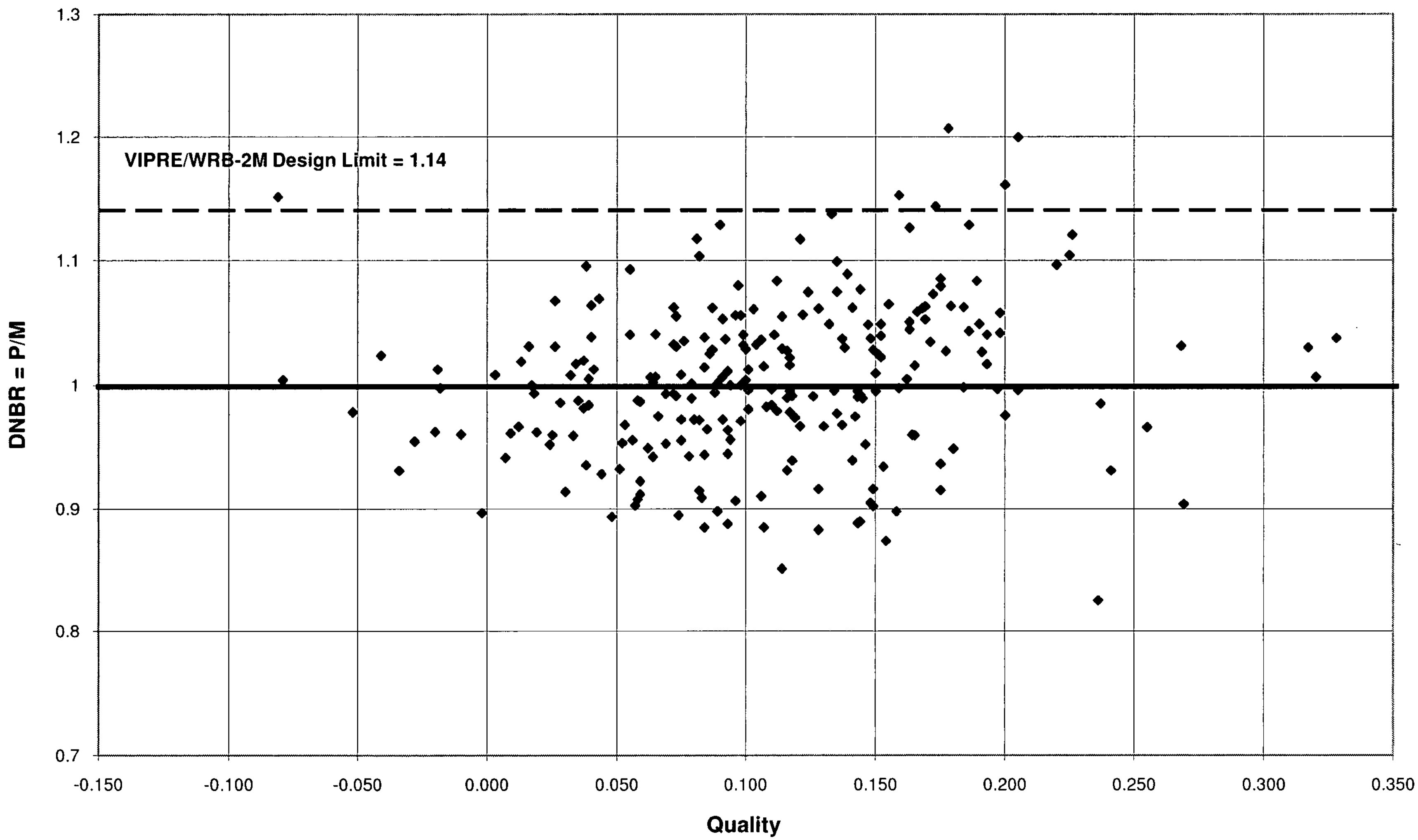
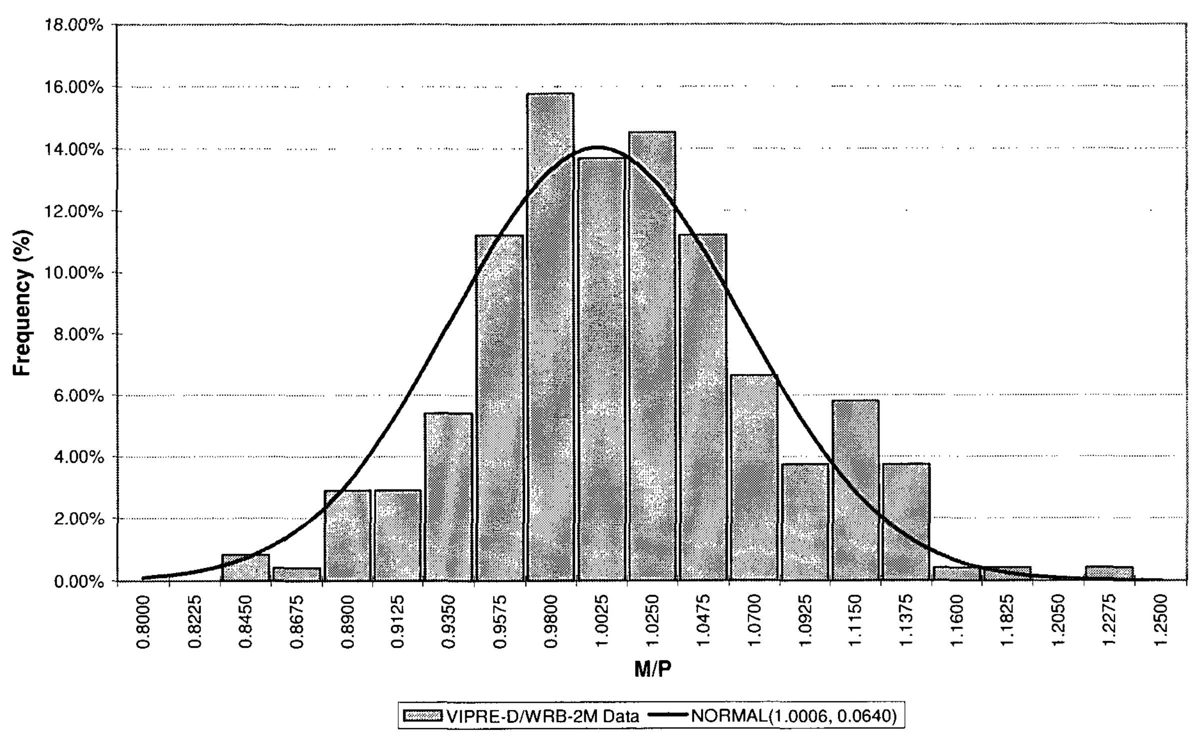


Figure C.5-7: DNBR vs. Quality for VIPRE-D/WRB-2M Database



The 241 data points of the VIPRE-D/WRB-2M M/P distribution calculated by Dominion were used to create the empirical probability density function. These data points were distributed among 21 equal bins that covered the entire range of M/P in the VIPRE D/WRB-2M distribution, and the frequency of data in each bin was determined. The resulting empirical probability density functions for the VIPRE-D/WRB-2M distribution were then compared with the probability density function of a normal distribution of mean 1.0006 and standard deviation 0.0640, which is the mean and standard deviation for the VIPRE-D/WRB-2M distribution calculated in Section C.5 above. Figure C.5-8 displays the resulting empirical probability density function for the VIPRE-D/WRB-2M M/P distribution, and compares it with the probability density function of the normal distribution of mean 1.0006 and standard deviation 0.0640.

Figure C.5-8: VIPRE-D/WRB-2M Probability Density Function



C.6 CONCLUSIONS

The WRB-2M correlation has been qualified with Dominion's VIPRE-D computer code. Table C.6-1 summarizes the DNBR design limits for VIPRE-D/WRB-2M that yields a 95% non-DNB probability at a 95% confidence level. The limit of 1.14 from VIPRE-D is the same limit as found with Westinghouse's version of VIPRE (Reference C1). The Westinghouse WRB-2M CHF correlation has been approved by the USNRC for use with Westinghouse 17x17 with 0.374 inch OD rods, and Modified LPD grids with or without MIFM's in a PWR reactor.

Table C.6-1: DNBR Limits for WRB-2M

	Dominion VIPRE-D	Westinghouse VIPRE-01
DNBR limit	**1.14**	1.14

Table C.6-2 summarizes the applicability and the ranges of validity for VIPRE-D/WRB-2M, which are the same as those on page 4-2 of Reference C1.

Table C.6-2: Range of Validity for VIPRE-D/WRB-2M

Pressure [psia]	1495[2] to 2425
Mass Velocity [Mlbm/hr-ft^2]	0.97 to 3.1
Thermodynamic Quality at CHF	-0.1 to 0.29

[2] The NRC SER [Serial #09-290] for WRB-2M (Appendix C) typographically listed the beginning of the pressure range as 1405 psia and stated it should match the WCAP-15025-P-A applicability range of 1495 to 2425 psia. Also a header in the applicability table, Table 1, in the NRC SER listed WNG-1, which should be WRB-2M.

The original submittal by Dominion to the NRC [Serial #08-0174] for Appendix C had the beginning of the pressure range was listed as 1440 psia and stated that the applicability range should match those which are on page 4-2 of WCAP-15025-P-A. The WCAP-15025-P-A [page 4-2] pressure applicability range is 1495 to 2425 psia.

C.7 REFERENCES

C1. Technical Report, WCAP-15025-P-A, “Modified WRB-2 Correlation, WRB-2M, for predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” L.D. Smith, et al, April 1999.

C2. Technical Report, "Tables for Normal Tolerance Limits, Sampling Plans and Screening," R. E. Odeh and D. B. Owen, 1980.

C3. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), “Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, ‘VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,’ Volumes 1, 2, 3 and 4,” May 1, 1986.

C4. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), “Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, ‘VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores,’ (TAC No. M79498),” October 30, 1993.

C5. Fleet Report, DOM-NAF-2-A, including Appendixes A and B, “Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code,” Rosa M. Bilbao y León, August 2006. (ML062650184)

C6. Letter from D. S. Collins (NRC) to J. A. Gresham (Westinghouse), “Modified WRB-2 Correlation WRB-2M for Predicting Critical Heat Flux in 17X17 Rod Bundles with Modified LPD Mixing Vane Grids,” February 3, 2006.

C7. Technical Report, “Boiling Crisis and Critical Heat Flux,” TID-25887, 1972.

C8. Letter from A.C. Thadani (NRC) to W.J. Johnson (Westinghouse), “Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P, Reactor Core Response to Excessive Secondary Steam Releases.” 1989.

C9. Technical Report, “Assessment of the Assumption of Normality (employing individual observed values),” American National Standards Institute, ANSI N15.15.1974.

Fleet Report DOM-NAF-2, Rev. 0.1-A
Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code

Attachment 1

Request for Additional Information
Set 1
Questions and Responses

(31 pages)

Dominion Resources Services, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion

June 30, 2005

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 05-328
NL&OS/ETS: R0
Docket Nos. 50-280/281
50-338/339
50-336/423
License Nos. DPR-32/37
NPF-4/7
DPR-65/NPF-49

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 ADDITIONAL INFORMATION ON
TOPICAL REPORT DOM-NAF-2:
REACTOR CORE THERMAL-HYDRAULICS USING THE VIPRE-D COMPUTER CODE
INCLUDING APPENDICES A AND B

In letters dated September 30, 2004 and January 13, 2005 (Serial Nos. 04-606 and 05-020, respectively), Dominion and DNC submitted the Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code" and associated Appendices A and B, for NRC review and approval. VIPRE-01 is a core thermal-hydraulics computer code developed by EPRI, approved by the NRC, and currently in use throughout the nuclear industry. VIPRE-D is the Dominion version of VIPRE-01, which has been enhanced by the addition of several vendor specific CHF correlations. In a May 19, 2005 letter, the NRC requested additional information to complete their review of VIPRE-D and the associated appendices.

On May 25, 2005 Dominion and NRC held a public meeting on VIPRE-D licensing issues at which Dominion discussed the proposed scope for the RAIs. The NRC staff agreed that the proposed scope for the responses was acceptable. The attachment to this letter provides the detailed responses discussed in the scope at the public meeting, including appropriate references and supporting information. If you have further questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services
Virginia Electric and Power Company
Dominion Nuclear Connecticut, Inc.

Attachment

Commitments made in this letter: None

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck who is Vice President – Nuclear Support Services, of Virginia Electric and Power Company and Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of these companies, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 30th day of June, 2005.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

Attachment

(Serial No. 05-328)

Request for Additional Information

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)

REQUEST FOR ADDITIONAL INFORMATION

PROPOSED TOPICAL REPORT DOM-NAF-2

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

DOMINION NUCLEAR CONNECTICUT (DNC)

NRC Question 1

Departure from Nucleate Boiling Ratio (DNBR) is sensitive to the turbulent mixing coefficient, Please justify the value of the turbulent mixing coefficient. Stating it is conservative is insufficient. Please show the sensitivity of the DNBR to the turbulent mixing coefficient.

Dominion/DNC Response:

The turbulent mixing coefficient (ABETA) is obtained empirically by the fuel vendor. Both AREVA and Westinghouse have conducted several subchannel mixing tests in pressurized water loops at Reynolds numbers similar to that of a PWR core under single phase and two-phase flow conditions (References 13, 14, 15) to determine ABETA. It has been determined experimentally that the value of ABETA is a function of grid spacing, and as such, it is dependent on the fuel product design. A turbulent mixing coefficient value of 0.038 has been validated by the NRC for analyzing Westinghouse 17 x 17 and 15 x 15 fuel assemblies with mixing vane grids having a spacer span of 26 inches or less (Reference 15). The Dominion submittal for the AREVA Fuel Transition (Section 4.2.5 in Reference 5) documented a turbulent mixing coefficient of 0.038 for the AREVA Advanced Mark-BW fuel. Therefore, a turbulent mixing coefficient value of 0.038 is applicable to current AREVA and Westinghouse fuel designs, and will be used by Dominion in VIPRE-D models for current fuel products. The turbulent mixing coefficient used by VIPRE-D models for future fuel designs will be provided by the fuel vendor.

As agreed during the May 25, 2005 Dominion-NRC public meeting, the sensitivity of the DNBR to the turbulent mixing coefficient has not been provided herein.

NRC Question 2

Please describe the basis for the use of the drift flux correlations employed as part of the Electric Power Research Institute (EPRI) void model. Please describe the flow regime/regimes that the EPRI drift velocity correlation is applicable to and show that this applies to the flow regimes experienced during DNB in the plant analyses. Comparisons of the drift velocity correlation/correlations to void data in rod bundles and small pipes would be desirable. This could be done using all other particular inputs and correlation choices (and code corrections) included to show the effect/ability to continue to predict the test data presented in Volume 4 of the VIPRE Manual entitled "Applications" dated 1987. Since *the EPRI void model appears to employ only the drift velocity correlation applicable to churn turbulent bubbly flow (for void fractions less than 0.3), please explain and justify why this correlation is applied to slug and annular flow where Critical Heat Flux (CHF) can occur. Define the limitations of the drift flux correlations (i.e. pressure range, flow conditions, etc.). What distribution parameter is assumed? Since more voiding may occur near the walls of the hot rods, how does the distribution parameter account for this condition in the drift flux modeling? Please explain.*

Dominion/DNC Response:

VIPRE-01 has been approved by the NRC (References 10 and 11). 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. Therefore, none of the models for bulk void, subcooled void and two-phase friction factor present in VIPRE-01 has been modified in any way in VIPRE-D, including the EPRI models.

The EPRI models for bulk void, subcooled void and two-phase friction factor are described in detail in the VIPRE-01 documentation (Section 2.6 of Reference 16 and Section 2.7 of Reference 7), including their underlying assumptions and ranges of validity. Sections 2.5 and 3.0 of Reference 9 document the VIPRE-01 comparisons performed by the code developer. VIPRE-01 two-phase friction factor models were evaluated against experimental data from the FRIGG rod bundle test loop. VIPRE-01 void models were evaluated against experimental data from the FRIGG rod bundle test loop, ANL void test and Martin void measurements at high pressure. The comparisons show that although all the models available in VIPRE-01 match the experimental data reasonably well, the EPRI set of correlations compare more favorably with the measured data. In consequence, the EPRI models for bulk void, subcooled void and two-phase friction factor are the default selections in VIPRE-01 for PWR analysis.

Based on the evaluation of the benchmark calculations discussed above, NRC staff concluded that the EPRI models for bulk void, subcooled void and two-phase friction are acceptable for use in licensing calculations (Reference 10).

The benchmark studies mentioned above, as well as the NRC approval of the EPRI models provided Dominion a good starting point for the selection of void and two-phase friction multiplier models. In addition, Dominion performed another sensitivity study to determine the most suitable set of models for Dominion applications. This sensitivity analysis, which is summarized in Section 5.4 of DOM-NAF-2, provides justification for Dominion's modeling selections, thus fulfilling condition (3) of the SER for VIPRE-01 MOD-01 (Reference 10).

Dominion performed a detailed analysis of the correlations available in the code, and four sets of correlations were chosen based on the compatibility of the modeling assumptions used in deriving the various correlations. 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

Although other relevant parameters such as void fraction or quality were verified throughout the benchmark evaluation, Dominion's acceptance criterion was based on the DNBR performance of the analyzed cases. The EPRI models for bulk void, subcooled void and two-phase friction factor provided the best DNBR comparison for all the following:

- VIPRE-D/BWU against the CHF experimental database for the AREVA BWU CHF correlations (DOM-NAF-2 Appendix A).
- VIPRE-D/BWU against the LYNXT/BWU code results for a set of representative operating conditions (DOM-NAF-2 Section 5.0)
- VIPRE-D/WRB-1 against the CHF experimental database for the Westinghouse WRB-1 CHF correlation (DOM-NAF-2 Appendix B).
- VIPRE-D/WRB-1 against the COBRA/WRB-1 code results for a set of representative operating conditions (DOM-NAF-2 Appendix B Section B.7).

Consistent with the results of these external and in-house benchmark studies, Dominion has demonstrated that the selection of the EPRI set of void and two-phase friction models is acceptable for application in DOM-NAF-2.

During the May 25, 2005 NRC-Dominion public meeting, Dominion agreed to provide some graphic comparisons to show VIPRE-D's performance for void fraction and quality. Figures 1 and 2 provide a comparison between the VIPRE-D/WRB-1 and the COBRA/WRB-1 codes. In particular, these figures show the void fraction and quality axial distributions in the hot channel at the time of minimum DNBR (3.0 seconds) for a Locked Rotor Transient event at Surry Power Station. Additional description and results associated with this transient were included in Section B.7.3 of Appendix B to DOM-NAF-2. In addition to the VIPRE-D/WRB-1 with the EPRI-EPRI-EPRI set of models, Figures 1 and 2 show VIPRE-D/WRB-1 with the LEVI-SMITH-HOMOGENEOUS set of models, which is the set of bulk void, subcooled void and two-phase friction models closest to COBRA/WRB-1 modeling in Reference 12 (LEVY-SMITH-BAROCZY).

Figures 1 and 2 show that the different void and two-phase friction models result in comparable void fraction and quality trends for COBRA and VIPRE-D. However, as noted above, DNBR comparisons are the main criterion used to evaluate the performance of VIPRE-D against other thermal-hydraulics codes and models. These comparisons demonstrated the acceptability of the EPRI models for bulk void, subcooled void and two-phase friction factor for use in the Dominion VIPRE-D models.

Figure 1: Locked Rotor Transient Equilibrium Quality Results at Time 3.0 Seconds

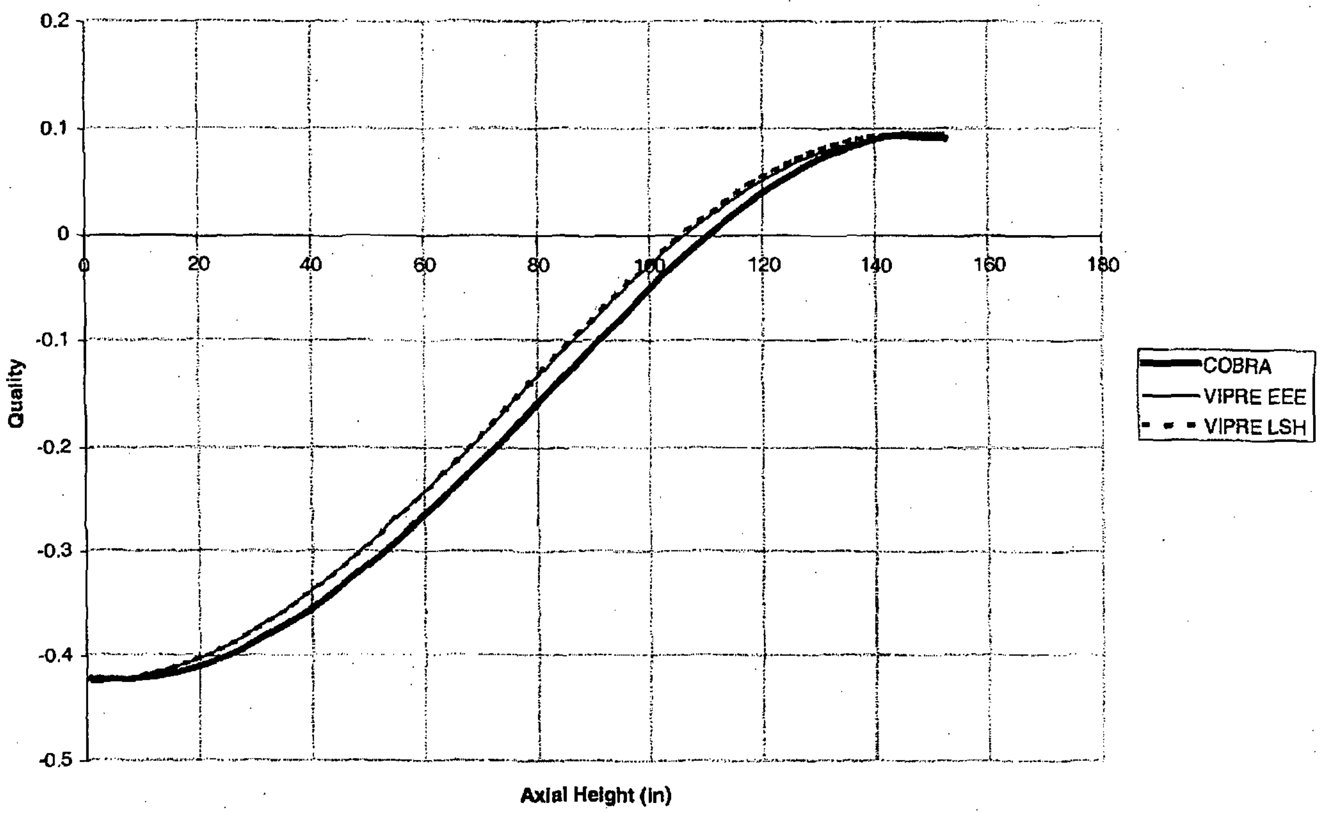
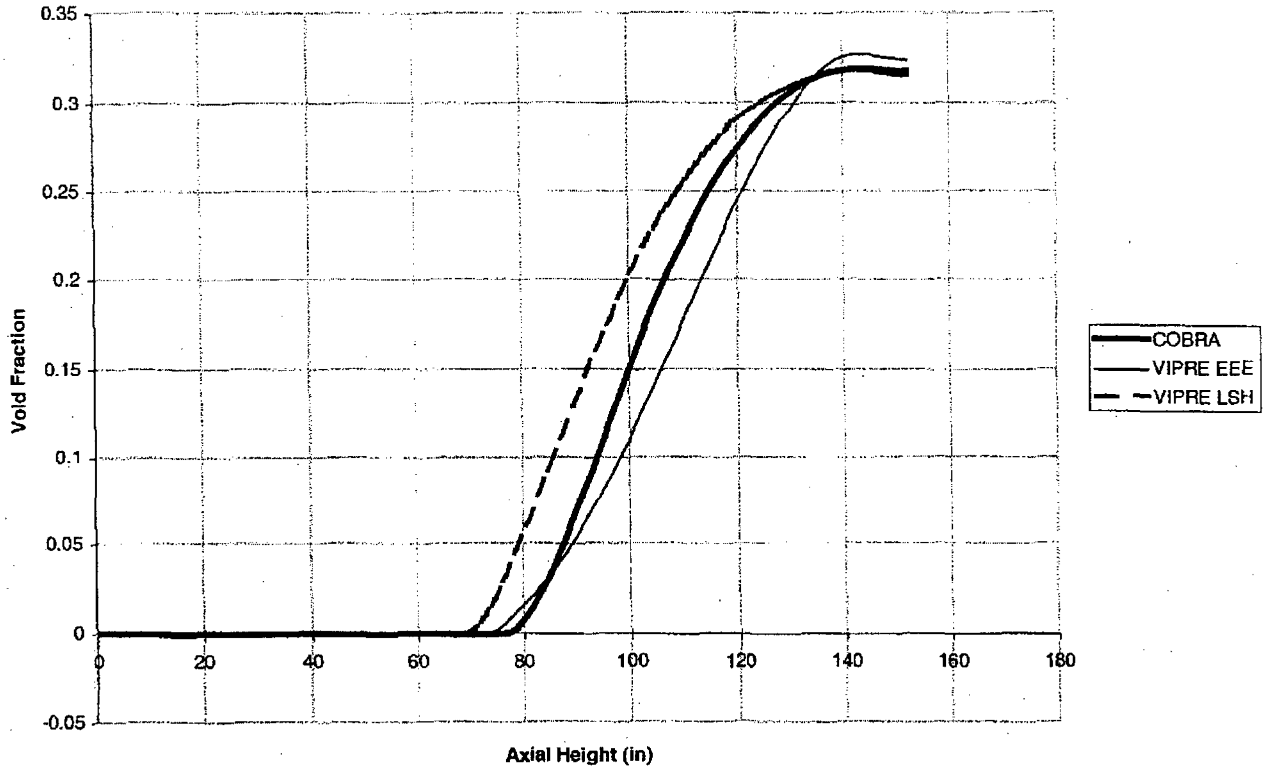


Figure 2: Locked Rotor Transient Void Fraction Results at Time 3.0 Seconds



NRC Question 3

Please explain the basis for choosing the EPRI bulk void model. Please describe the transient test cases employed to determine the values given in Table 5.4-1 and show the void distributions for the VIPRE and LYNXT codes at the initiation of DNB in some example cases. Also provide justification for the choice of the subcooled boiling model as well as the two-phase friction multiplier. Please show comparisons of the VIPRE-D code to data for these models. Please also describe the limitations and identify the ranges of applicability of each of these correlations.

Dominion/DNC Response:

The basis for choosing the EPRI bulk void, subcooled void and two-phase friction multiplier models is provided in the response to Question 2.

The transient test cases employed to determine the values in Table 5.4-1 of DOM-NAF-2 are the same statepoints that were used in Section 5.1 of DOM-NAF-2 to benchmark VIPRE-D/BWU against LYNXT/BWU, and were also used to support the Licensing Amendment Request (LAR) for the Framatome ANP Fuel Transition (Reference 5). These statepoints were 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. These various statepoints provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power $F\Delta H$), 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). These statepoints cover the full range of conditions and axial offsets in UFSAR Chapter 15 evaluations (except for MSLB that was evaluated separately), and were specifically selected to challenge the three BWU CHF correlations.

In the May 25, 2004 NRC-Dominion public meeting, Dominion indicated that LYNXT models use different (proprietary) void models, and that as a consequence, void distribution comparisons for VIPRE-D/BWU and LYNXT/BWU would not be expected to match. A comparison of the void and quality distributions for COBRA/WRB-1 and VIPRE-D/WRB-1 is provided in the response to Question 2.

NRC Question 4

In section 4.12, please explain what "nearly identical" means. Identifying the percent difference between the two results would be helpful or show the plot of the two DNBR calculations.

Dominion/DNC Response:

In the few occasions in which VIPRE-D did not converge when using the default values of the convergence criteria and damping factors, and these criteria and/or damping factors were adjusted to ensure numerical convergence, the difference in the DNBR results reported by the code was 0.5% or less.

NRC Question 5

Section 5.3 describes a comparison with the COBRA code but states the Minimum DNBR (MDNBR) results are different because the analyses use different fuel types and CHF correlations. Please provide the latest comparisons between the codes using the same fuel type and CHF correlations. Please show the channel void distribution and quality at several selected times during the events. Show the steamline break, feedline break, and loss-of-flow events.

Dominion Response:

Section B.7 in Appendix B to Topical Report DOM-NAF-2 includes comparisons of VIPRE-D/WRB-1 with COBRA/WRB-1 for Surry Power Station. These comparisons use the same fuel type (Westinghouse 15x15 SIF) and the same CHF correlation (Westinghouse WRB-1).

The comparisons include:
- A statepoint safety analysis evaluation of 164 statepoints obtained from the Surry UFSAR Chapter 14 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.
- A Main Steam Line Break statepoint application.
- A Feedwater Malfunction Transient (FWMAL).
- A Locked Rotor Transient (LOCROT).

Selected examples comparing channel void and quality distributions were provided in the answer to Question 2.

NRC Question 6

The qualification document identifies the DNBR limits for the correlations for several pressure groups. Please explain what DNBR limit is applied or how the situation is handled when the range of validity is exceeded for the other parameters identified in Table A.5.2. Please also define the quality range for the correlation.

Dominion/DNC Response:

The qualification of the BWU CHF correlations with the VIPRE-D code was performed consistent with AREVA's approved Topical Report (Reference 4). The evaluations performed for the various pressure groups as well as the resulting limits were included in Appendix A for completeness. However, Dominion will only apply the DNBR limits listed in Table A.5-1:

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 at or above 594 psia	1.18
VIPRE-D/BWU-N	
DNBR limit below 1200 psia	1.39
DNBR limit at or above 1200 psia	1.22

The ranges of validity for the three BWU CHF correlations, including quality, are listed in Table A.5-2:

	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 Low Flow Main Steam Line Break (MSLB) statepoint has been identified as the only event that might fall outside of the range of validity of the BWU correlations (for the mass velocity). This issue was already identified in the Framatome Fuel Transition Submittal (Reference 5). The Low Flow MSLB statepoint results in minimum DNBR values of the order of 5.8, which are obviously very far from being limiting. In this case Dominion will use the DNBR limit corresponding to the pressure at which the event takes place.

NRC Question 7

CHF is also sensitive to the axial power distribution. Since the correlations were developed from data with uniform or symmetric power distributions, please justify applicability of the correlations to the asymmetric power distributions that may be limiting in the plant calculations. Explain how the correlations are applied and describe any correction factors that may be applied to accommodate skewed distributions.

Dominion/DNC Response:

This issue is a generic concern with all CHF correlations, since these correlations were developed using a limited number of uniform and non-uniform axial power distributions. Both the AREVA BWU set of correlations (qualified with VIPRE-D in Appendix A) and the Westinghouse WRB-1 correlation (qualified in Appendix B) include a correction factor, the F-factor proposed by Tong (Reference 8), that is used to correct for non-uniform axial power shapes. The Tong Factor has been extensively verified by both Westinghouse (Reference 6) and AREVA (Reference 4) with numerous CHF test data, that over the years have included uniform and non-uniform power shapes. The use of this F-factor has been previously approved by the NRC staff.

In addition to the verifications performed by both vendors, Dominion performed a statepoint safety analysis evaluation using symmetric and non-symmetric, positively and negatively skewed axial power shapes to benchmark VIPRE-D/BWU with LYNXT/BWU and VIPRE-D/WRB-1 with COBRA/WRB-1. The results of these evaluations do not show differences in the performance of the CHF correlations dependent on the form or uniformity of the axial power distributions.

NRC Question 8

Please justify applicability of the steady-state DNB correlations to steamline breaks since these events have rapid depressurizations where the steady-state correlations may not be applicable. These transients may also transition through slug and annular flow. As such, please justify the use of the EPRI bulk boiling drift flux model since it only applies to bubbly flow.

Dominion/DNC Response:

The issue of the applicability of steady-state CHF correlations to transient analysis is generic to all thermal hydraulics codes and CHF correlations. Section 6.7 of Reference 9 provides a study to determine the applicability of steady-state CHF correlations to transient analysis for VIPRE-01. The NRC staff review concluded that the studies have shown that the transient CHF for power ramp and flow coastdown transients are higher than the steady-state CHF, and that, except for very rapid depressurization events (LOCA), the use of CHF correlations developed with steady-state CHF data can correctly or conservatively predict the transient CHF when the instantaneous local fluid conditions are used (Reference 10).

Dominion's Main Steam Line Break (MSLB) DNBR evaluations are performed at the limiting statepoint. Dominion does not perform full DNBR transient analysis for this event. For MSLB all the DNB limiting statepoints occur after the pressurizer has drained and the upper head has flashed. Therefore, pressure is changing relatively slowly and the quasi steady-state assumption is appropriate.

The use of the EPRI bulk boiling void correlation was justified in the response to Question 2.

NRC Question 9

Please discuss whether the slip option will be used and if so justify the slip ratio employed in the DNBR calculations.

Dominion/DNC Response:

Dominion VIPRE-D models do not use the slip option.

NRC Question 10

Since the conduction model will not be used, please explain how the stored energy in the rod is accounted for. Please explain why use of the dummy rod model is conservative since the conduction model does not include the effects of gap conductance and initial stored energies. Please also describe how the heat flux is calculated for use as input to the VIPRE-D code.

Dominion/DNC Response:

Conduction models are typically used to perform fuel temperature calculations and to simulate delay of energy transport. Dominion does not plan to use VIPRE-D for these applications. The use of the dummy rod model is consistent with previously approved Dominion methodologies (Reference 12).

The use of the dummy rod model requires the user to provide the fuel rod surface heat flux as one of the operating input conditions. For steady-state statepoint analysis this value is easily calculated based on core thermal power. Fuel rod surface heat flux forcing functions for transient calculations are provided by an NRC-approved transient system code (e.g., RETRAN). Transient system codes account for fuel conduction, gap conductance, and delayed energy transport effects. Therefore, the use of the dummy rod model appropriately includes the relevant effects into the analysis.

Appendix A to DOM-NAF-2

NRC Question 1

The appropriate statistical analysis of the data, which form Tables A.3.1-1, A.3.2-1, and A.3.3-1 is an analysis of variance of a mixed-effects model.

a) Give the appropriate analysis of variance tables for these mixed-effects models.

b) Formulate the appropriate statistical hypothesis tests to justify the values for M/P and $\sigma_{M/P}$ used in Eq. A.1.1 based on the data in Tables A.3.1-1, A.3.2-1, and A.3.3-1,

c) For those cases where individual DNBR design limits were developed for each low pressure group, how were the results of the above analysis of variance taken into account?

Dominion/DNC Response:

As clarified in our May 25, 2005 meeting with the NRC staff, this question refers to Tables A.4.1-2, A.4.2-2 and A.4.3-2.

a) The qualification of the BWU CHF correlations with the VIPRE-D code was performed consistent with AREVA's approved Topical Report (Reference 4). An analysis of variance (ANOVA) test was performed for each one of the three BWU correlations. This analysis was not included in the original submission because it was not deemed to provide additional substantial information to the qualification.

An ANOVA test divides the database in several groupings according to a given variable and then evaluates whether or not the distributions for each one of the groups appear to belong to the overall distribution. If all the groupings belong to the same distribution it can be deduced that the total population does not show a bias with respect to that particular variable Even though it was recognized that ANOVA tests cannot be used as the sole measure of the performance of a CHF correlation, they can be useful to indicate an extremely bad mismatch (very large F statistic). The variables analyzed were mass velocity, pressure, quality, test cell type and axial flux shape type.

The results of the ANOVA tests for the BWU-Z, BWU-ZM and BWU-N correlations are provided in Tables 1 through 5. The results for the BWU-ZM (Table 3) and BWU-N (Table 4) correlations prove the ANOVA hypothesis: all the groups analyzed belong to the same distribution, i.e. there is no bias of the results regarding the analyzed variables. The results for the BWU-Z correlation (Table 1) show F values slightly above the critical value, but still reasonably small. Following the AREVA approach in Reference 4, an additional ANOVA test was performed excluding the low pressure data from the BWU-Z database. This treatment is consistent with the fact that the low pressure data were taken separately by determining a separate DNBR design limit. Table 2 summarizes the results of this second ANOVA analyses. While the values of the F statistic do decrease somewhat in most cases, they are still slightly above the critical value of F for the appropriate number of degrees of freedom. An additional ANOVA test was also performed for BWU-N without the low pressure data (Table 5), and the trends observed are the same as those shown in Table 4. For all cases, these are the same trends showed by AREVA in Reference 4.

Table 1: M/P CHF Performance by Independent Variable Grouping using the entire BWU-Z database at 95% confidence level

	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Mass Velocities					
0.5 $Mlbm/hr\text{-}ft^2$	9	0.9559	0.1147	1.1083	0.7833
1.0 $Mlbm/hr\text{-}ft^2$	90	1.0120	0.1088	1.2974	0.6980
1.5 $Mlbm/hr\text{-}ft^2$	97	0.9934	0.0726	1.1609	0.8321
2.0 $Mlbm/hr\text{-}ft^2$	134	0.9721	0.0878	1.1612	0.7717
2.5 $Mlbm/hr\text{-}ft^2$	85	1.0018	0.0908	1.1693	0.7667
3.0 $Mlbm/hr\text{-}ft^2$	71	1.0216	0.0891	1.1481	0.7261
3.5 $Mlbm/hr\text{-}ft^2$	42	0.9853	0.0749	1.1148	0.8387
	$F_{distribution} = 3.4995$		$F_{critical}(6,521) = 2.1159$		
Analysis by Pressures					
Below 1250 psia	64	1.0437	0.0936	1.2974	0.8342
1250 - 1649 psia	111	0.9789	0.0881	1.1683	0.7770
1650 - 1949 psia	108	0.9722	0.0916	1.1693	0.7667
1950 - 2249 psia	116	1.0089	0.0914	1.1481	0.6980
Above 2250 psia	129	0.9914	0.0796	1.1576	0.7669
	$F_{distribution} = 8.3625$		$F_{critical}(4,523) = 2.3889$		
Analysis by Qualities					
Below 5%	34	0.9634	0.0794	1.0840	0.7261
5% to 10%	84	1.0415	0.0740	1.1693	0.8306
10% to 15%	127	0.9886	0.0884	1.1576	0.7717
15% to 20%	122	0.9750	0.0774	1.1479	0.8126
20% to 25%	90	0.9755	0.0913	1.1411	0.6980
25% to 30%	29	0.9969	0.1039	1.1609	0.7669
30% to 40%	18	1.0343	0.1006	1.1642	0.8484
Above 40%	24	1.0553	0.1198	1.2974	0.8342
	$F_{distribution} = 7.9433$		$F_{critical}(7,520) = 2.0271$		
Analysis by Test Cell Type					
Unit Cell	218	1.0077	0.0894	1.1683	0.7669
Guide Tube	170	0.9849	0.0917	1.1693	0.7261
Other	140	0.9877	0.0899	1.2974	0.6980
	$F_{distribution} = 3.6760$		$F_{critical}(2,525) = 3.0128$		
All Data BWU-Z					
All Data	528	0.9950	0.0907	1.2974	0.6980

Table 2: M/P CHF Performance by Independent Variable Grouping using the BWU-Z database without the low pressure data at 95% confidence level

	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Mass Velocities					
1.25 $Mlbm/hr\text{-}ft^2$	70	0.9909	0.1025	1.1457	0.6980
1.5 $Mlbm/hr\text{-}ft^2$	86	0.9876	0.0700	1.1576	0.8321
2.0 $Mlbm/hr\text{-}ft^2$	119	0.9626	0.0853	1.1492	0.7717
2.5 $Mlbm/hr\text{-}ft^2$	77	0.9994	0.0945	1.1693	0.7667
3.0 $Mlbm/hr\text{-}ft^2$	70	1.0201	0.0889	1.1481	0.7261
3.5 $Mlbm/hr\text{-}ft^2$	42	0.9853	0.0749	1.1148	0.8387
	$F_{distribution} = 4.2356$		$F_{critical}(5,458) = 2.2337$		
Analysis by Pressures					
1250 – 1649 psia	111	0.9789	0.0881	1.1683	0.7770
1650 – 1949 psia	108	0.9722	0.0916	1.1693	0.7667
1950 - 2249 psia	116	1.0089	0.0914	1.1481	0.6980
Above 2250 psia	129	0.9914	0.0796	1.1576	0.7669
	$F_{distribution} = 3.8436$		$F_{critical}(3,460) = 2.6243$		
Analysis by Qualities					
< 10%	116	1.0186	0.0839	1.1693	0.7261
10% - 15%	122	0.9859	0.0887	1.1576	0.7717
15% - 20%	115	0.9737	0.0787	1.1479	0.8126
> 20%	111	0.9746	0.0951	1.1411	0.6980
	$F_{distribution} = 6.7560$		$F_{critical}(3,460) = 2.6243$		
Analysis by Test Cell Type					
Unit Cell	188	1.0024	0.0880	1.1683	0.7669
Guide Tube	150	0.9810	0.0931	1.1693	0.7261
Other	126	0.9761	0.0805	1.1457	0.6980
	$F_{distribution} = 4.1660$		$F_{critical}(2,461) = 3.0153$		
BWU-Z without low pressure data					
All Data	464	0.9883	0.0883	1.1693	0.6980

Table 3: M/P CHF Performance by Independent Variable Grouping using the entire BWU-ZM database at 95% confidence level

	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Mass Velocities					
<1.25 Mlbm/hr-ft^2	36	1.0338	0.0712	1.2110	0.8828
1.5 Mlbm/hr-ft^2	32	1.0256	0.1039	1.2299	0.8142
2.0 Mlbm/hr-ft^2	41	0.9859	0.0890	1.1467	0.8255
2.5 Mlbm/hr-ft^2	18	0.9918	0.0843	1.1607	0.7793
3.0 Mlbm/hr-ft^2	10	1.0480	0.0809	1.1699	0.9612
3.5 Mlbm/hr-ft^2	11	1.0228	0.0676	1.1324	0.9195
	$F_{distribution} = 1.9439$		$F_{critical}(5,142) = 2.2779$		
Analysis by Pressures					
< 1000 psia	11	0.9758	0.0545	1.0749	0.8832
1000 - 1500 psia	15	1.0202	0.0840	1.2110	0.9125
1500 - 2000 psia	53	1.0249	0.0982	1.1844	0.8142
> 2000 psia	69	1.0099	0.0833	1.2299	0.7793
	$F_{distribution} = 1.0528$		$F_{critical}(3,144) = 2.6674$		
Analysis by Qualities					
< 10%	11	1.0004	0.0622	1.1249	0.9195
10% - 15%	39	1.0111	0.1023	1.1699	0.7793
15% - 20%	35	1.0082	0.0737	1.1680	0.8783
> 20%	63	1.0209	0.0897	1.2299	0.8496
	$F_{distribution} = 0.2794$		$F_{critical}(3,144) = 2.6674$		
Analysis by Test Cell Type					
Unit Cell	76	1.0230	0.1000	1.2299	0.8142
Guide Tube	72	1.0041	0.0715	1.1747	0.7793
	$F_{distribution} = 1.7240$		$F_{critical}(1,146) = 3.9059$		
All Data BWU-ZM					
All Data	148	1.0138	0.0875	1.2299	0.7793

Table 4: M/P CHF Performance by Independent Variable Grouping using the entire BWU-N database at 95% confidence level

	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Mass Velocities					
0.5 Mlbm/hr-ft^2	147	1.0052	0.1469	1.3507	0.6707
1.0 Mlbm/hr-ft^2	172	0.9870	0.1331	1.3251	0.6635
1.5 Mlbm/hr-ft^2	194	1.0018	0.0985	1.2983	0.7262
2.0 Mlbm/hr-ft^2	208	1.0086	0.0758	1.2541	0.7560
2.5 Mlbm/hr-ft^2	149	0.9980	0.0808	1.2061	0.7747
3.0 Mlbm/hr-ft^2	124	1.0050	0.0767	1.1845	0.8154
3.5 Mlbm/hr-ft^2	96	1.0102	0.0882	1.2826	0.8002
	$F_{distribution} = 0.9171$		$F_{critical}(6,1083) = 2.1069$		
Analysis by Pressures					
Below 1250 psia	40	1.0309	0.1066	1.2569	0.6985
1250 - 1649 psia	192	1.0092	0.1110	1.3251	0.7396
1650 - 1949 psia	198	0.9900	0.0999	1.3507	0.7372
1950 - 2249 psia	446	1.0021	0.1002	1.2868	0.6635
Above 2250 psia	214	1.0002	0.1069	1.3030	0.6930
	$F_{distribution} = 1.6829$		$F_{critical}(4,1085) = 2.3801$		
Analysis by Qualities					
Below 5%	222	1.0129	0.0788	1.1834	0.7867
5% to 10%	192	0.9993	0.0758	1.1845	0.8041
10% to 15%	234	1.0051	0.0885	1.2787	0.7372
15% to 20%	158	1.0110	0.1029	1.3030	0.7747
20% to 25%	106	0.9766	0.1393	1.3251	0.6746
25% to 30%	45	0.9821	0.1371	1.2983	0.6635
30% to 40%	56	0.9915	0.1341	1.2917	0.7264
Above 40%	77	1.0008	0.1533	1.3507	0.6707
	$F_{distribution} = 1.8027$		$F_{critical}(7,1082) = 2.0180$		
Analysis by Test Cell Type					
Unit Cell	553	1.0031	0.1140	1.3251	0.6635
Guide Tube	385	1.0012	0.0889	1.2003	0.6707
Intersection	152	0.9988	0.1004	1.3507	0.8002
	$F_{distribution} = 0.1133$		$F_{critical}(2,1087) = 3.0040$		
Analysis by Axial Flux Shape Type					
Uniform	355	0.9915	0.1200	1.3251	0.6635
Non Uniform – Symmetric	568	1.0069	0.0968	1.3507	0.6985
Non Uniform - Asymmetric	167	1.0063	0.0875	1.2003	0.7346
	$F_{distribution} = 2.6068$		$F_{critical}(2,1087) = 3.0040$		
All Data BWU-N					
All Data	1090	1.0018	0.1038	1.3507	0.6635

Table 5: M/P CHF Performance by Independent Variable Grouping using the BWU-N database without the low pressure data at 95% confidence level

Grouping	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Mass Velocities					
1.25 $Mlbm/hr\text{-}ft^2$	304	0.9945	0.1387	1.3507	0.6635
1.5 $Mlbm/hr\text{-}ft^2$	188	1.0001	0.0995	1.2983	0.7262
2.0 $Mlbm/hr\text{-}ft^2$	202	1.0074	0.0765	1.2541	0.7560
2.5 $Mlbm/hr\text{-}ft^2$	143	0.9961	0.0814	1.2061	0.7747
3.0 $Mlbm/hr\text{-}ft^2$	117	1.0040	0.0772	1.1845	0.8154
3.5 $Mlbm/hr\text{-}ft^2$	96	1.0102	0.0882	1.2826	0.8002
	$F_{distribution} = 0.6302$		$F_{critical}(5,1044) = 2.2227$		
Analysis by Pressures					
1250 – 1649 psia	192	1.0092	0.1110	1.3251	0.7396
1650 – 1949 psia	198	0.9900	0.0999	1.3507	0.7372
1950 - 2249 psia	446	1.0021	0.1002	1.2868	0.6635
Above 2250 psia	214	1.0002	0.1069	1.3030	0.6930
	$F_{distribution} = 1.1604$		$F_{critical}(3,1046) = 2.6134$		
Analysis by Qualities					
< 5%	222	1.0129	0.0788	1.1834	0.7867
5% - 10%	186	0.9986	0.0764	1.1845	0.8041
10% - 15%	227	1.0041	0.0891	1.2787	0.7372
15% - 20%	151	1.0076	0.1027	1.3030	0.7747
20% - 25%	102	0.9734	0.1409	1.3251	0.6746
> 25%	162	0.9924	0.1430	1.3507	0.6635
	$F_{distribution} = 2.4577$		$F_{critical}(5,1044) = 2.2227$		
Analysis by Test Cell Type					
Unit Cell	537	1.0033	0.1141	1.3251	0.6635
Guide Tube	385	1.0012	0.0889	1.2003	0.6707
Intersection	128	0.9884	0.0979	1.3507	0.8002
	$F_{distribution} = 1.0784$		$F_{critical}(2,1047) = 3.0043$		
Analysis by Axial Flux Shape Type					
Uniform	355	0.9915	0.1200	1.3251	0.6635
Non Uniform – Symmetric	528	1.0051	0.0959	1.3507	0.7045
Non Uniform - Asymmetric	167	1.0063	0.0875	1.2003	0.7346
	$F_{distribution} = 2.1303$		$F_{critical}(2,1047) = 3.0043$		
BWU-N without low pressure data					
All Data	1050	1.0007	0.1036	1.3507	0.6635

b) As clarified in our May 25, 2005 meeting with the NRC staff, this question refers to Eq. A.4.1.1 and Tables A.4.1-1, A.4.2-1 and A.4.3-1. The validity of this equation is based on two assumptions: 1) the average M/P is 1.0 and 2) the M/P distribution is normal. These two assumptions were demonstrated in Appendix A:

CORRELATION	AVERAGE M/P = 1.0	M/P DISTRIBUTION NORMAL
BWU-Z	Table A.4.1-1	Page A-13
BWU-ZM	Table A.4.2-1	Page A-23
BWU-N	Table A.4.3-1	Page A-33 (Hypernormal distribution, also seen by AREVA in Reference 4)

c) Different DNBR limits were developed for low pressure groups for the BWU-Z and BWU-N correlations following the guidance in Reference 4. The plots (not the ANOVA tests displayed in Tables 1 and 4 respectively) seemed to indicate a poorer performance of the correlation at low pressures, and for that reason a separate DNBR limit was calculated at low pressures. In those cases, a new ANOVA analysis was performed excluding the low-pressure data (Table 2 for BWU-Z and Table 5 for BWU-N). These ANOVA results were similar to the previous results. All Dominion results were similar to AREVA results in Reference 4.

NRC Question 2

As in Appendix B, you state that the plots show that there are no biases in the M/P ratio distributions, and that the performance of the CHF correlations is independent of the three variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes. The plots again suggest but do not demonstrate that the claims made in those sentences are true. Please give the appropriate statistical analysis, that demonstrates the truth of the claim.

Dominion/DNC Response:

As discussed in the response to Question 1 for Appendix A, an ANOVA analysis was performed for each correlation to formally demonstrate that the performance of the BWU CHF correlations is not biased by the three independent variables present in the correlations (mass velocity, pressure and quality). Please refer to the response to Question 1 for analyses demonstrating no biases in the M/P distributions.

NRC Question 3

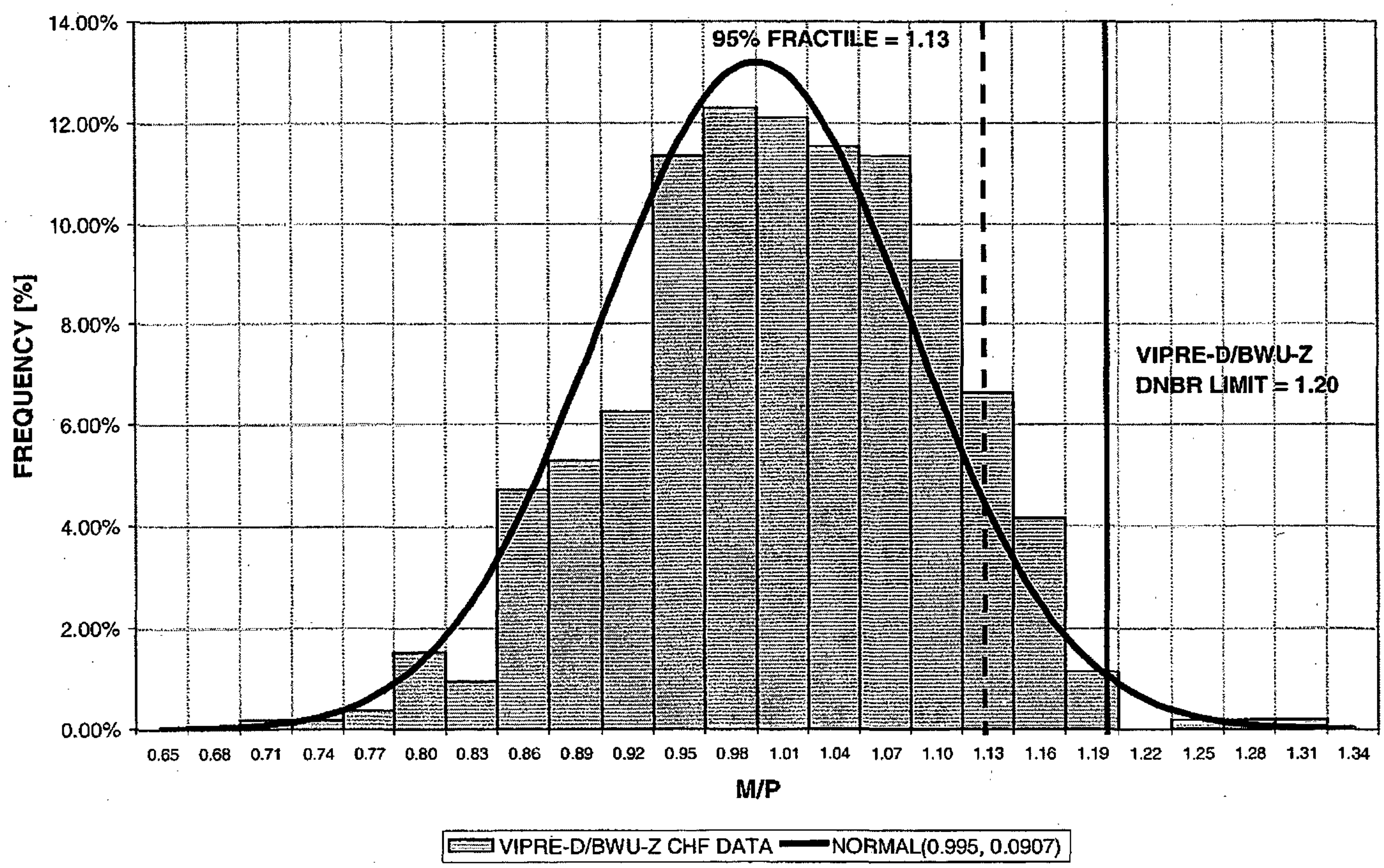
Please show the empirical probability density functions for the M/P values used in the analyses together with the estimate of the 95-percent fractile for each correlation.

Dominion/DNC Response:

The probability density functions for the BWU-Z, BWU-ZM and BWU-N correlations are shown in Figures 3, 4 and 5.

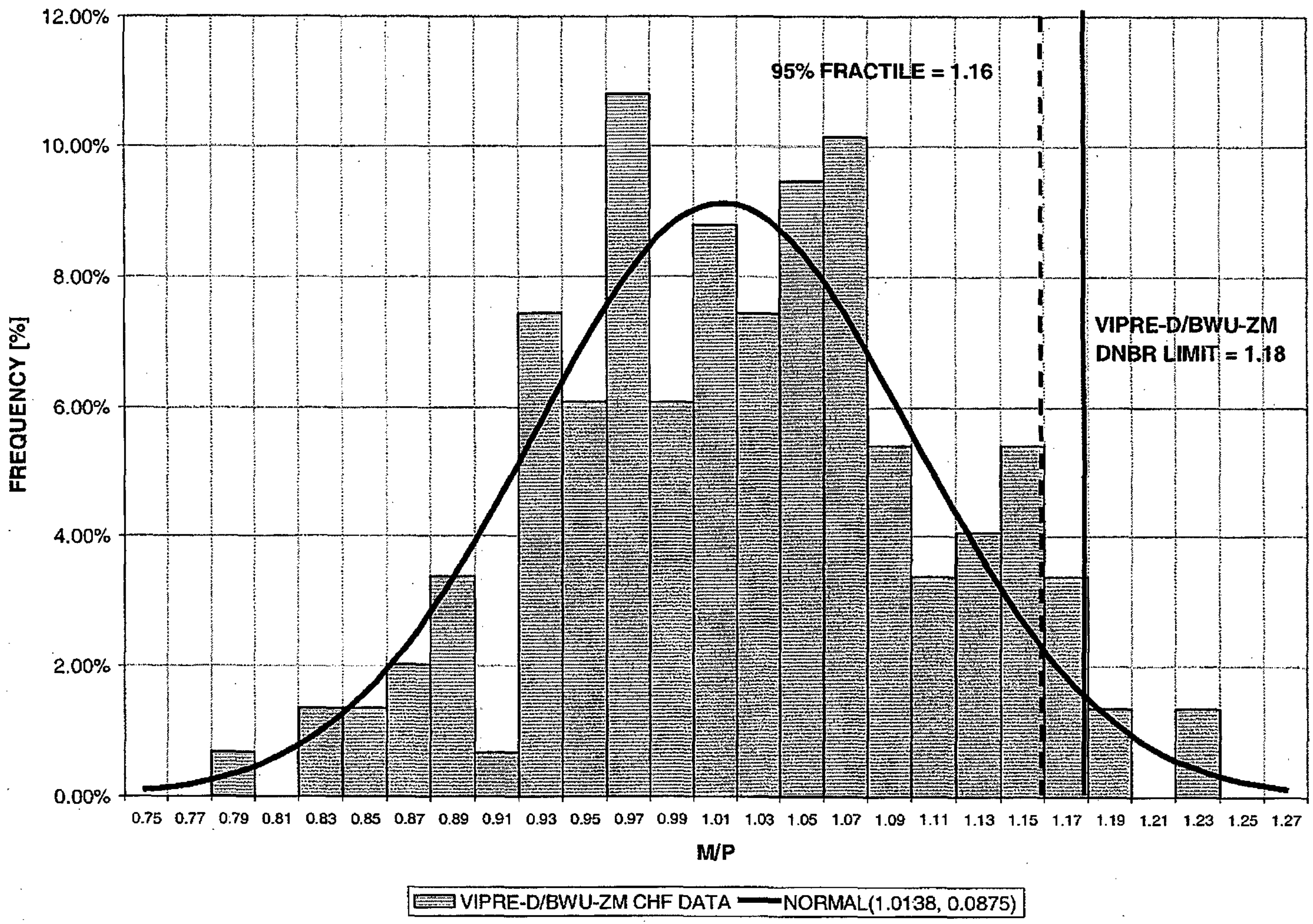
The 528 datapoints of the VIPRE-D/BWU-Z M/P distribution summarized in Section A.4.1 of DOM-NAF-2 Appendix A were used to create the empirical probability density function. These datapoints were distributed among 24 equal bins that covered the entire range of M/P in the VIPRE-D/BWU-Z distribution, and the frequency of data in each bin was determined. The resulting empirical probability density function for the VIPRE-D/BWU-Z M/P distribution was then compared with the probability density function of a normal distribution of mean 0.995 and standard deviation 0.0907, which are the mean and standard deviation calculated for the VIPRE-D/BWU-Z M/P distribution in Section A.4.1 of DOM-NAF-2 Appendix A. Figure 3 also displays the obtained 95% fractile (1.13) for the data and the VIPRE-D/BWU-Z DNBR limit obtained in Section A.4.1 of DOM-NAF-2 Appendix A (1.20).

Figure 3: VIPRE-D/BWU-Z Probability Density Function



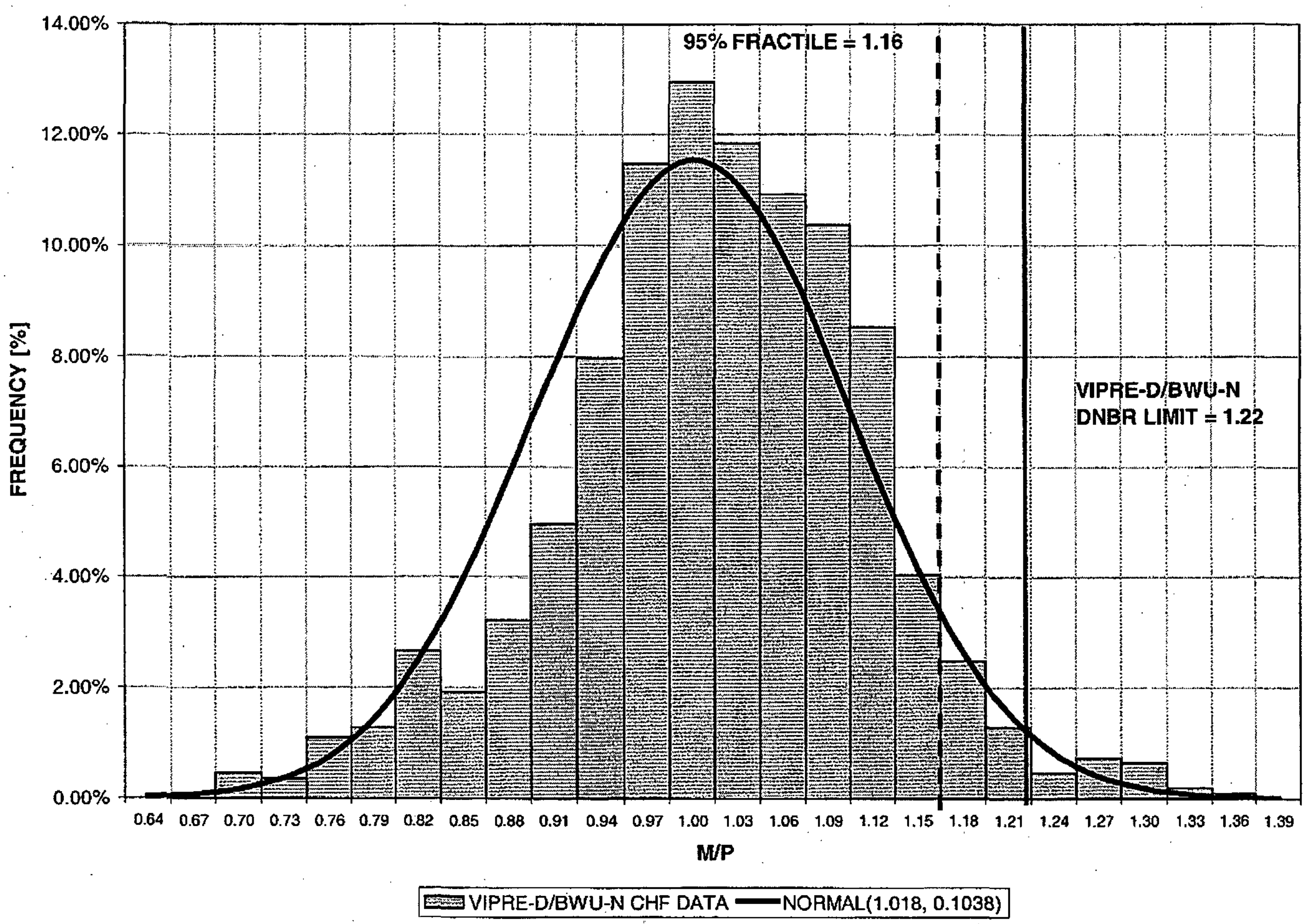
The 148 datapoints of the VIPRE-D/BWU-ZM M/P distribution summarized in Section A.4.2 of DOM-NAF-2 Appendix A were used to create the empirical probability density function. These datapoints were distributed among 27 equal bins that covered the entire range of M/P in the VIPRE-D/BWU-ZM distribution, and the frequency of data in each bin was determined. The resulting empirical probability density function for the VIPRE-D/BWU-ZM M/P distribution was then compared with the probability density function of a normal distribution of mean 1.0138 and standard deviation 0.0875, which are the mean and standard deviation calculated for the VIPRE-D/BWU-ZM M/P distribution in Section A.4.2 of DOM-NAF-2 Appendix A. Figure 4 also displays the obtained 95% fractile (1.16) for the data and the VIPRE-D/BWU-ZM DNBR limit obtained in Section A.4.2 of DOM-NAF-2 Appendix A (1.18).

Figure 4: VIPRE-D/BWU-ZM Probability Density Function



The 1090 datapoints of the VIPRE-D/BWU-N M/P distribution summarized in Section A.4.3 of DOM-NAF-2 Appendix A were used to create the empirical probability density function. These datapoints were distributed among 26 equal bins that covered the entire range of M/P in the VIPRE-D/BWU-N distribution, and the frequency of data in each bin was determined. The resulting empirical probability density function for the VIPRE-D/BWU-N M/P distribution was then compared with the probability density function of a normal distribution of mean 1.0018 and standard deviation 0.1038, which are the mean and standard deviation calculated for the VIPRE-D/BWU-N M/P distribution in Section A.4.3 of DOM-NAF-2 Appendix A. Figure 5 also displays the obtained 95% fractile (1.16) for the data and the VIPRE-D/BWU-N DNBR limit obtained in Section A.4.3 of DOM-NAF-2 Appendix A (1.22).

Figure 5: VIPRE-D/BWU-N Probability Density Function



Appendix B VIPRE-D

NRC Question 1

The appropriate statistical analysis of the data that form Table B.6-1 is an analysis of variance of a mixed-effects model.

a) *Give the appropriate analysis of variance table for this mixed-effects model.*
b) *Formulate the appropriate statistical hypothesis tests to justify the values for M/P and $\sigma_{M/P}$ used in Eq. B.6.1.*

Dominion/DNC Response:

a) The qualification of the WRB-1 CHF correlations with the VIPRE-D code was performed consistent with Dominion's approved Topical Report for COBRA/WRB-1 (Reference 12) and Westinghouse's approved Topical Report (Reference 6). An analysis of variance (ANOVA) test was performed but was not included in the original submission because it was not deemed to provide additional substantial information to the qualification. As mentioned in the response to Question 1 on Appendix A, it was recognized that ANOVA tests cannot be used as the sole measure of the performance of a CHF correlation, but they can be useful to indicate an extremely bad mismatch. The variables analyzed were mass velocity, pressure, quality and test cell type.

The ANOVA results for VIPRE-D/WRB-1 (Table 6) exceed the critical values of F for some comparisons, but other comparisons prove the hypothesis that all the groups belong to the same distribution, i.e. that there is no bias of the results regarding the analyzed variables. These are the same trends discussed in the NRC's Safety Evaluation of WRB-1 (Reference 6).

Table 6: M/P CHF Performance by Independent Variable Grouping using the entire WRB-1 database at 95% confidence level

	Number of Data	Average M/P	Standard Deviation	Maximum M/P	Minimum M/P
Analysis by Mass Velocities					
Below 1.25 Mlbm/hr-ft^2	58	0.9849	0.0753	1.1158	0.8005
1.25 – 1.75 Mlbm/hr-ft^2	131	1.0125	0.0970	1.3069	0.7632
1.75 – 2.25 Mlbm/hr-ft^2	247	0.9930	0.0914	1.2711	0.7600
2.25 – 2.75 Mlbm/hr-ft^2	203	1.0129	0.0688	1.1715	0.8276
2.75 – 3.25 Mlbm/hr-ft^2	159	1.0074	0.0755	1.1858	0.8337
Above 3.25 Mlbm/hr-ft^2	147	1.0137	0.0787	1.2356	0.8017
	$F_{distribution}$ = 2.6870		$F_{critical}(5,939)$ = 2.2236		
Analysis by Pressures					
Below 1575 psia	227	1.0201	0.0851	1.3069	0.8017
1575 - 1850 psia	179	0.9878	0.0861	1.2711	0.7632
1850 - 2250 psia	277	1.0076	0.0774	1.1873	0.7600
Above 2250 psia	262	1.0014	0.0814	1.2356	0.8005
	$F_{distribution}$ = 5.4330		$F_{critical}(3,941)$ = 2.6144		
Analysis by Qualities					
Below 5%	262	1.0000	0.0762	1.3069	0.8017
5% to 10%	199	1.0119	0.0778	1.2034	0.8226
10% to 15%	247	1.0090	0.0810	1.2711	0.8005
15% to 20%	169	0.9927	0.0942	1.2312	0.7600
20% to 25%	68	1.0222	0.0922	1.2234	0.7662
	$F_{distribution}$ = 2.4209		$F_{critical}(4,940)$ = 2.3814		
Analysis by Geometry Type					
4x4	456	0.9941	0.0847	1.3069	0.7600
5x5	489	1.0154	0.0794	1.2711	0.7632
	$F_{distribution}$ = 15.9428		$F_{critical}(1,943)$ = 3.8513		
Analysis by Thimble vs. Typical					
Thimble	207	0.9910	0.0771	1.2229	0.7873
Typical	738	1.0091	0.0838	1.3069	0.7600
	$F_{distribution}$ = 7.8372		$F_{critical}(1,943)$ = 3.8513		
All Data WRB-1					
All Data	945	1.0051	0.0827	1.3069	0.7600

d) The validity of equation B.6.1 is based on two assumptions: 1) the average M/P is 1.0 and 2) the M/P distribution is normal. These two assumptions were demonstrated in Appendix B:

CORRELATION	AVERAGE M/P = 1.0	M/P DISTRIBUTION NORMAL
WRB-1	Table B.6-1	Page B-12

NRC Question 2

On page B-14 you state, "These plots show that there are no biases in the M/P ratio distributions, and that the performance of the WRB-1 CHF correlation is independent of the three variables of interest. The plots show a mostly uniform scatter of the data and no obvious trends or slopes." The plots suggest but do not demonstrate that the claims made in those sentences are true. Please give the appropriate statistical analysis that demonstrates the truth of the claim.

Dominion/DNC Response:

As discussed in the response to Question 1 for Appendix B, an ANOVA analysis was performed to formally demonstrate that the performance of the WRB-1 CHF correlation is not biased by the three independent variables present in the correlation (mass velocity, pressure and quality). Please refer to the response to Question 1 for an analysis demonstrating no biases in the M/P distribution.

NRC Question 3

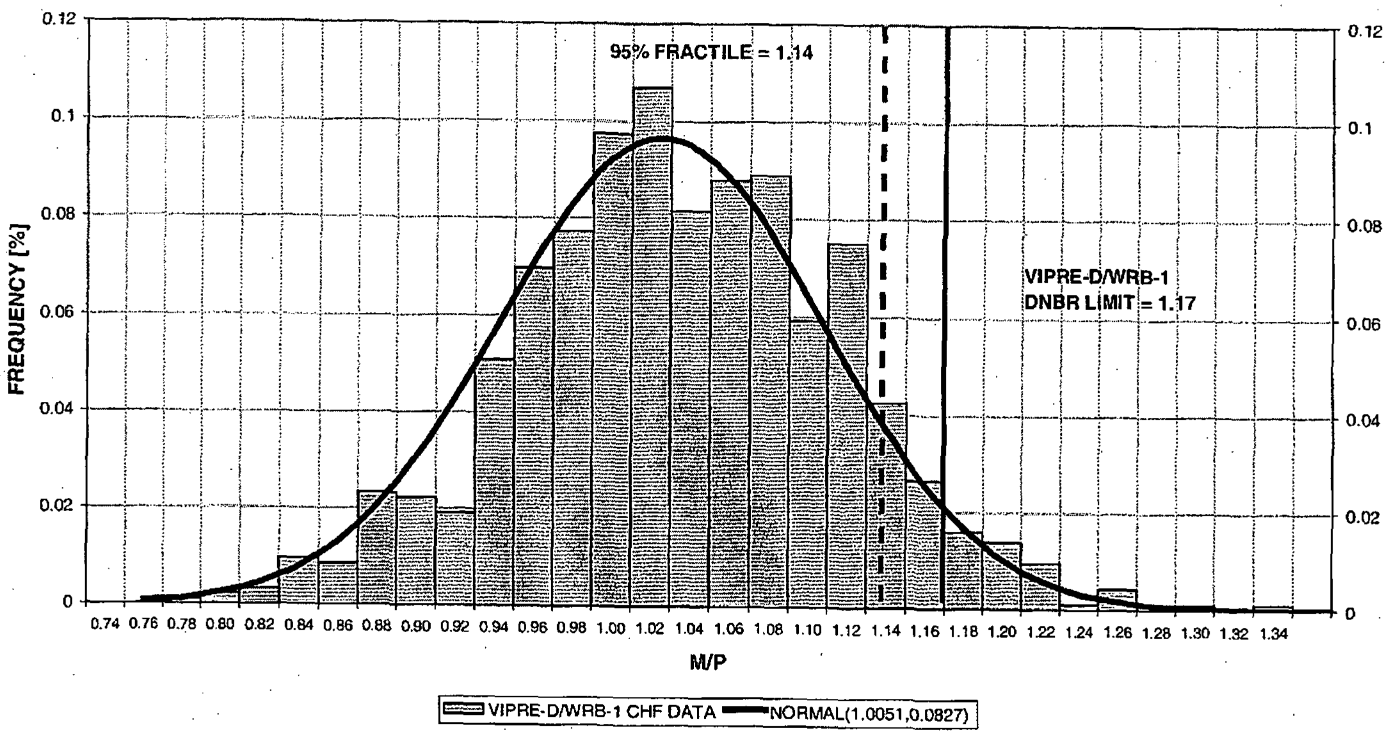
Please show the empirical probability density function for the M/P values used in the analyses together with the estimate of the 95-percent fractile.

Dominion/DNC Response:

The probability density function for the WRB-1 correlation is shown in Figure 6.

The 945 datapoints of the VIPRE-D/WRB-1 M/P distribution summarized in Section B.6 of DOM-NAF-2 Appendix B were used to create the empirical probability density function. These datapoints were distributed among 31 equal bins that covered the entire range of M/P in the VIPRE-D/WRB-1 distribution, and the frequency of data in each bin was determined. The resulting empirical probability density function for the VIPRE-D/WRB-1 M/P distribution was then compared with the probability density function of a normal distribution of mean 1.0051 and standard deviation 0.0827, which are the mean and standard deviation calculated for the VIPRE-D/WRB-1 M/P distribution in Section B.6 of DOM-NAF-2 Appendix B. Figure 6 also displays the obtained 95% fractile (1.14) for the data and the VIPRE-D/WRB-1 DNBR limit obtained in Section B.6 of DOM-NAF-2 Appendix B (1.17).

Figure 6: VIPRE-D/WRB-1 Probability Density Function



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

Attachment 2

Information Regarding a LYNXT Error Supporting the
Request for Approval of Fleet Report DOM-NAF-2

(5 pages)

Dominion Resources Services, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion

September 8, 2005

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 05-020A
NL&OS/ETS: R0
Docket Nos. 50-280/281
50-338/339
50-336/423
License Nos. DPR-32/37
NPF-4/7
DPR-65/NPF-49

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
INFORMATION REGARDING A LYNXT ERROR SUPPORTING THE 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

In a September 28, 2004 letter (Serial No. 04-406), Dominion/DNC submitted 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," for NRC review and approval. Dominion/DNC was recently notified by Framatome ANP of an error identified in the LYNXT code, which was used to benchmark the VIPRE-D code. Although the impact of this error is considered negligible and should not affect the NRC's review of VIPRE-D, Dominion/DNC is providing a description of the error and the impact for your information. The attachment to this letter summarizes the error and the impact on the DNB benchmark analyses.

Although the docket number is identified for each Dominion/DNC unit, Dominion/DNC 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.

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

Very truly yours,

L N Hartz

Leslie N. Hartz
Vice President – Nuclear Engineering
Virginia Electric and Power Company
Dominion Nuclear Connecticut, Inc.

Attachment

Commitments made in this letter: None

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Attachment 1

Topical Report DOM-NAF-2

05-020A

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

INFORMATION REGARDING A LYNXT ERROR

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

Characterization of the LYNXT Error

During the development of a new version of the LYNXT computer code, Framatome ANP detected an error that affects the LYNXT DNB predictions for previous code versions. It was discovered that two different surface fits to the ASME steam tables were used to calculate the values for the thermodynamic quality for a given local coolant condition within the code. The difference in the two values is based on two different water property subroutines used to generate the saturated liquid enthalpy (h_f) and the latent heat of vaporization (h_{fg}), which are needed to calculate the thermodynamic quality given the local coolant enthalpy. One thermodynamic quality definition was being used in the flow field calculations and the second definition was being used in the calculation of the DNBR. Although the two definitions are very close (maximum observed differences less than 0.2 percent in h_f and h_{fg}) there is an impact on DNBR predictions. Neither surface fit is incorrect by itself, but the inconsistency of using two different values for the quality was characterized as a code error.

Generic Impact of the LYNXT Error

Due to the location of the error in the LYNXT code, there is no impact to the LYNXT/BWU code/correlation limits reported in Framatome ANP's Topical Report BAW-10199 and Addendum 2. The differences between the DNBR results provided by the corrected and the uncorrected versions of the LYNXT code are extremely small, but observable.

Specific Impact to DOM-NAF-2

Topical Report DOM-NAF-2 describes Dominion's use of the VIPRE-D code, including modeling and qualification for Pressurized Water Reactors (PWR) thermal-hydraulic design. The Topical is entirely based on VIPRE-D calculations, and it does not rely on LYNXT results. However, the Topical includes information about VIPRE-D benchmarks to the NRC-approved code, LYNXT, to assist the NRC in the review of the VIPRE-D Topical Report.

- For the 173 North Anna statepoints used in Section 5.1 in the Topical Report DOM-NAF-2, the average difference between LYNXT predictions with the error and without the error is less than 0.02%, and the maximum difference is less than 0.15%.

- The maximum change to any numerical value reported in Section 5 of the main body of Topical Report DOM-NAF-2 regarding benchmark DNBR calculations between VIPRE-D and LYNXT is 0.02%.

- The comparisons between the corrected LYNXT and VIPRE-D are slightly better than the comparisons between the uncorrected LYNXT and VIPRE-D.

Appendix A to Topical Report DOM-NAF-2 documents Dominion's qualification of the BWU-N, BWU-Z and BWU-ZM correlations with VIPRE-D. Tables A.4.1-3 and A.4.3-3 of Appendix A list the LYNXT/BWU code/correlation limits for information in comparison to the calculated values for VIPRE-D/BWU. Since there is no impact to the LYNXT/BWU code-correlation limits reported in Framatome ANP's Topical Report BAW-10199 (including Addendum 2), Appendix A to Topical Report DOM-NAF-2 is not affected by this error.

Appendix B to Topical Report DOM-NAF-2 documents Dominion's qualification of the WRB-1 correlation with VIPRE-D. Appendix B to Topical Report DOM-NAF-2 is not affected by this error.

Conclusion

Based on the above discussion, Dominion has concluded that the LYNXT error has a negligible impact to the information provided in Topical Report DOM-NAF-2, including Appendixes A and B.