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May 4, 2007

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 06-578C
NL&OS/CDS: R2
Docket No. 50-305
License No. DPR-43

**DOMINION ENERGY KEWAUNEE, INC.**
**KEWAUNEE POWER STATION**
**IMPLEMENTATION OF THE DOMINION STATISTICAL DNBR METHODOLOGY WITH VIPRE-D/WRB-1 AT KEWAUNEE POWER STATION**

In a January 31, 2006, public meeting with NRC staff, Dominion Energy Kewaunee, Inc. (DEK) presented a conceptual approach and implementation strategy for application of existing NRC-approved Dominion-Virginia nuclear core design and safety analysis methods to Kewaunee Power Station (KPS) (reference 1). Fundamental to the proposed approach was creation of a composite topical report (DOM-NAF-5) that would document the application of the relevant methodologies to KPS.

On August 16, 2006, DEK submitted DOM-NAF-5 without attachments A and B (reference 2). On December 6, 2006, Attachment A to DOM-NAF-5, containing Core Management Systems benchmark analysis results, was submitted (reference 3).

On April 16, 2007, DEK submitted Attachment B to DOM-NAF-5, containing RETRAN benchmark analysis results (reference 4). This submittal, in conjunction with References 2 and 3, provides the complete contents of topical report DOM-NAF-5. DEK will issue a consolidated version of the topical report, DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," including the supplemental material of Attachments A and B, following NRC review and approval.

In DOM-NAF-5, Section 3.5 (reference 2), DEK indicated that the KPS specific implementation of the Dominion Statistical DNBR Evaluation Methodology would be submitted for NRC review and approval. Attachment 1 to this letter provides the KPS plant specific application of the NRC approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," for KPS cores containing Westinghouse 422V+ fuel assemblies with the VIPRE-D/WRB-1 code correlation. Under 10 CFR 50.59 (c)(2)(vii) the Statistical Design Limit (SDL) constitutes a design basis limit for fission product barriers. Therefore, DEK requests NRC review and approval of the SDL documented in Attachment 1.

In order to support application of these methods to KPS cycle 29, DEK requests NRC review and approval of DOM-NAF-5 and the attached plant specific implementation and associated limit by September 30, 2007. A subsequent administrative license amendment request (LAR) to add DOM-NAF-5 to the KPS Technical Specifications list of analytical methods used for determining core operating limits is scheduled to be submitted in June 2007. The requested date for NRC staff approval of the LAR will be January 31, 2008. The requested LAR approval date supports application of DOM-NAF-5 to KPS cycle 29, which is scheduled to begin in April 2008.

Should you have any questions, please contact Mr. Craig D. Sly at 804-273-2784.

Very truly yours,

G. T. Bischof
Vice President - Nuclear Engineering

References:

1. Summary of Meeting on January 31, 2006, "To Discuss the Applicability of Dominion Safety and Core Design Methods to Kewaunee Power Station," (TAC No. MC 9566), (ADAMS Accession Number ML 060400098).
2. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated August 16, 2006 (ADAMS Accession Number ML 062370351).
3. Letter from G. T. Bischof (DEK) to NRC, "Attachment A to Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated December 6, 2006 (ADAMS Accession Number ML 0063410177).
4. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, "Application of Dominion Nuclear Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," dated April 16, 2007.

Attachment:

1. Implementation of the Dominion Statistical DNBR Methodology with VIPRE-D / WRB-1 at Kewaunee Power Station.

Commitments made in this letter: None

cc: Regional Administrator
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**Attachment 1**

# IMPLEMENTATION OF THE DOMINION STATISTICAL DNBR EVALUATION METHODOLOGY WITH VIPRE-D / WRB-1 AT KEWAUNEE POWER STATION

**KEWAUNEE POWER STATION**

**DOMINION ENERGY KEWAUNEE, INC.**

**IMPLEMENTATION OF THE DOMINION STATISTICAL DNBR EVALUATION METHODOLOGY WITH VIPRE-D/WRB-1 AT KEWAUNEE POWER STATION (KPS)**

Prepared By: ______________________ 4/9/07
Robert S. Brackmann Date

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Supervisor, Nuclear Safety Analysis

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# 1. Introduction

DOM-NAF-5 (Reference 7) was submitted to the NRC, in August 2006, to document application of Dominion nuclear core design and safety analysis methods to Kewaunee Power Station (KPS). This report provides the plant specific application for KPS cores containing Westinghouse 422V+ fuel assemblies, in accordance with Section 3.5 of DOM-NAF-5. Specifically, this report supports the application of U.S. Nuclear Regulatory Commission (USNRC) approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 1) to KPS. It provides the technical basis and documentation required by the USNRC to evaluate the plant specific application of VEP-NE-2-A methods to KPS. This application employs the VIPRE-D code with the Westinghouse WRB-1 Critical Heat Flux (CHF) correlation (VIPRE-D/WRB-1) for the thermal-hydraulic analysis of Westinghouse 422V+ fuel assemblies at KPS. In particular, Dominion requests the review and approval of the Statistical Design Limit (SDL) documented herein as per 10 CFR 50.59(c)(2)(vii) it constitutes a Design Basis Limit for Fission Products Barrier (DBLFPB).

The approval of a subsequent license amendment request (LAR) to add DOM-NAF-5 (Reference 7) to Section 6.9.a.4 of the KPS Technical Specifications, will provide Dominion with the ability to use the VIPRE-D/WRB-1 code/correlation set to perform licensing calculations for Westinghouse 422V+ fuel in KPS cores, using the Deterministic Design Limit (DDL) qualified in DOM-NAF-5 (which includes references to DOM-NAF-2-A) and the SDL documented herein. The intended applications are described in DOM-NAF-5.

## 2. Background

Kewaunee Power Station (KPS) became part of the Dominion nuclear fleet following Dominion's acquisition of KPS in July 2005. Currently, departure from nuclear boiling (DNB) analyses required to support the use of Westinghouse 422V+ fuel at KPS are performed using the Westinghouse Revised Thermal Design Procedure (RTDP) methodology which was approved for Kewaunee's fuel transition to Westinghouse's 422V+ fuel (Reference 5).

This report documents the plant specific application of the USNRC approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," (Reference 1) for KPS cores containing Westinghouse 422V+ fuel assemblies with the VIPRE-D/WRB-1 code/correlation in accordance with Section 3.5 of DOM-NAF-5.

In 1985, Virginia Power (Dominion) submitted to the USNRC Topical Report VEP-NE-2-A describing a proposed methodology for the statistical treatment of key uncertainties in core thermal-hydraulic DNBR analysis. The methodology provided DNBR margin through the use of statistical rather than deterministic uncertainty treatment. The methodology was reviewed and approved by the USNRC in May 1987, and the SER provided by the USNRC listed the following conditions for its use (Reference 8):

1) The selection and justification of the Nominal Statepoints used to perform the plant specific implementation must be included in the submittal (Sections 3.6 and 3.8).

2) Justification of the distribution, mean and standard deviation for all the statistically treated parameters must be included in the submittal (Section 3.2).

3) Justification of the value of model uncertainty must be included in the plant specific submittal (Section 3.4).

4) For the relevant CHF correlations, justification of the 95/95 DNBR limit and the normality of the M/P distribution, its mean and standard deviation must be included in the submission, unless there is an approved Topical Report documenting these (such as Reference 2).

This report provides the technical basis for implementation of the Dominion Statistical DNBR Evaluation Methodology for Westinghouse 422V+ fuel at Kewaunee with VIPRE-D/WRB-1, as well as the SDL obtained by this implementation. This report also documents that the existing Reactor Core Safety Limits and protection functions (OTΔT, OPΔT, FΔI, etc) do not require revision as a consequence of this implementation. The list of USAR transients for which the code/correlations will be applied is also included herein.

Section 3 of this report summarizes the implementation of the Dominion Statistical DNBR Evaluation Methodology to Westinghouse 422V+ fuel at Kewaunee Power Station with the VIPRE-D/WRB-1 code/correlation. Section 4 provides the necessary information for the plant specific application of the VIPRE-D/WRB-1 code/correlation to Kewaunee, including the SDL. The verification of the existing Reactor Core Safety Limits, Protection Setpoints and KPS USAR Chapter 14 events with the above DNBR limits is documented in Section 4.

# 3. Implementation of the Statistical DNBR Evaluation Methodology

## 3.1. Methodology Review

The Statistical DNBR Evaluation Methodology (Reference 1) is employed herein to determine a statistical DNBR limit for KPS. This new limit combines the correlation uncertainty with the DNBR sensitivities to uncertainties in key DNBR analysis input parameters. Even though the new DNBR limit (the Statistical Design Limit or SDL) is larger than the deterministic code/correlation design limit (DDL), its use is advantageous as the Statistical DNBR Evaluation Methodology permits the use of nominal values for operating initial conditions instead of requiring the application of evaluated uncertainties to the initial conditions for statepoint and transient analysis.

The SDL is developed by means of a Monte Carlo analysis. The variation of actual operating conditions about nominal statepoints due to parameter measurement and other key DNB uncertainties is modeled through the use of a random number generator. Two thousand random statepoints are generated for each nominal statepoint. The random statepoints are then supplied to the thermal-hydraulics code VIPRE-D, which calculates the minimum DNBR (MDNBR) for each statepoint. Each MDNBR is randomized by a code/correlation uncertainty factor as described in Reference 1 using the upper 95% confidence limit on the VIPRE-D/WRB-1 measured-to-predicted (M/P) CHF ratio standard deviation (Reference 2). The standard deviation of the resultant randomized DNBR distribution is increased by a small sample correction factor to obtain a 95% upper confidence limit, and is then combined Root-Sum-Square with code and model uncertainties to obtain a total DNBR standard deviation ($s_{total}$). The SDL is then calculated as:

$$SDL = 1 + 1.645 * s_{total} \qquad \text{[Eq. 3.1]}$$

in which the 1.645 multiplier is the z-value for the one-sided 95% probability of a normal distribution. This SDL thus provides peak fuel rod DNB protection at greater than 95/95.

As an additional criterion, the SDL is tested to determine the full core DNB probability when the peak pin reaches the SDL. This process is performed by summing the DNB probability of each rod in the core, using a bounding rod census curve and the DNB sensitivity to rod power. If necessary, the SDL is increased to reduce the full core DNB probability to 0.1% or less.

## 3.2. Uncertainty Analysis

This section is included herein to satisfy Condition 2 in the SER (Reference 8) of VEP-NE-2-A (Reference 1).

Consistent with VEP-NE-2-A, inlet temperature, pressurizer pressure, core thermal power, reactor vessel flow rate, core bypass flow, the nuclear enthalpy rise factor and the engineering enthalpy rise factor were selected as the statistically treated parameters in the implementation analysis. The magnitudes and functional forms of the uncertainties for the statistically treated parameters were derived in a rigorous analysis of plant hardware and measurement/calibration procedures, and have been summarized in Table 3.2-1.

The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and inlet temperature were quantified using all sensor, rack, and other components of a total uncertainty and combined in a manner consistent with their relative dependence or independence (Reference 6). Westinghouse quantified these uncertainties (Reference 6) for Kewaunee's transition to Westinghouse's 422V+ fuel (Reference 5). Total uncertainties were quantified at the $2\sigma$ level, corresponding to two-sided 95% probability. The standard deviations $\sigma$ were obtained by dividing the total uncertainty by 1.96, which is the z-value for the two-sided 95% probability of a normal distribution.

The magnitude and distribution of uncertainty for pressurizer pressure (system pressure) per the pressurizer pressure control system were quantified. The pressurizer pressure uncertainty is a normal, two-sided, 95% probability distribution with a magnitude of ±35.1 psi (Reference 6) and a standard deviation ($\sigma$) of 17.9 psi.

The magnitude and distribution of uncertainty on the average temperature (Tavg) per the Tavg rod control system were quantified. The average temperature uncertainty is a normal, two-sided, 95% probability distribution with a magnitude of ±4.9°F (Reference 6) and a standard deviation ($\sigma$) of 2.5°F.

The core power uncertainty is defined as a normal, two-sided, 95% probability distribution with a magnitude of 1.7% (Reference 6) and a standard deviation of 0.9%.

The reactor coolant system (RSC) flow uncertainty is defined as a normal, two-sided, 95% probability distribution with a magnitude of 2.7% (Reference 6) and a standard deviation of 1.4%.

The two-sided, 95/95 tolerance interval (95% probability, 95% confidence) for the measurement uncertainty of the nuclear enthalpy rise factor, $F_{\Delta H}^{N}$, is 3.5%. Conservatively, the measured $F_{\Delta H}^{N}$ uncertainty was defined as a normal distribution with a 4% tolerance interval for consistency with previous applications.

The magnitude and distribution of uncertainty on the engineering hot channel factor, $F_{\Delta H}^{E}$, was quantified as a normal probability distribution with a magnitude of ± 3.0%. The

Statistical DNBR Evaluation Methodology (Reference 1) treats the $F_{\Delta H}^{E}$ uncertainty as a uniform probability distribution.

The total core bypass flow consists of separate flow paths through the thimble tubes, direct leakage to the outlet nozzle, baffle joint leakage flow, upper head spray flow and core-baffle gap flow. These five components were each quantified based on the current Kewaunee core configuration, their uncertainties conservatively modeled and the flows and uncertainties totaled. The Monte Carlo analysis ultimately used a best estimate bypass flow of 5.1% with an uncertainty of 1.9%. The analysis assumed that the probability was uniformly distributed.

Table 3.2-1: Kewaunee Parameter Uncertainties (Reference 6)

| PARAMETER | NOMINAL VALUE | STANDARD DEVIATION | UNCERTAINTY | DISTRIBUTION |
|---|---|---|---|---|
| Pressure [psia] | 2250 | 17.9 psi | 35.1 psi at $2\sigma$ | Normal |
| Temperature [°F] | 541.2 | 2.5°F | 4.9°F at $2\sigma$ | Normal |
| Power [MW] | 1,772 | 0.9% | 1.7% at $2\sigma$ | Normal |
| Flow [gpm] | 186,000 | 1.4% | 2.7% at $2\sigma$ | Normal |
| $F_{\Delta H}^{N}$ | 1.635 | 2.0% | 4.0% at $2\sigma$ | Normal |
| $F_{\Delta H}^{E}$ | 1.0 | N/A | 3.0% | Uniform |
| Bypass [%] | 5.1 | N/A | 1.9% | Uniform |

## 3.3. CHF Correlations

The WRB-1/W-3 CHF correlations are used for the calculation of DNBRs in Westinghouse 422V+ fuel assemblies. Only WRB-1 is applicable to the operating conditions for which the Statistical DNBR Evaluation Methodology applies. Table 3.3-1 presents the Design Limit correlation data for VIPRE-D/WRB-1. The W-3 correlation is only used below the first mixing grid or when the operating conditions are outside of the range of validity of the WRB-1 CHF correlation, such as the main steam-line break evaluation, where there are reduced temperature and pressure. The W-3 CHF correlation is always used deterministically.

Table 3.3-1: CHF Code/Correlation Data (Reference 2)

| | **WRB-1** |
|---|---|
| **Average M/P** | 1.005 |
| **S(M/P)** | 0.083 |
| **n** | 945 |
| **K*** | 1.03963 |
| **K x S(M/P)** | 0.08629 |

## 3.4. Model Uncertainty Term

This section is included herein to satisfy Condition 3 in the SER (Reference 8) of the Statistical DNBR Evaluation Methodology Topical Report (Reference 1).

The VIPRE-D 20-channel production model for Kewaunee was used in the development of the VIPRE-D/WRB-1 SDL for Kewaunee. Since this is the production model that Dominion intends to use for all Kewaunee evaluations once the Topical Report DOM-NAF-5 is approved, and the VIPRE-D code becomes the approved code for the determination of the core operating limits in the Core Operating Limits Report (COLR), there is no additional uncertainty associated with the use of this model. In summary, it is concluded that no correction for model uncertainty is necessary, and the model uncertainty term is set to zero for the calculation of the total DNBR standard deviation.

## 3.5 Code Uncertainty

The code uncertainty accounts for any differences between Dominion's VIPRE-D and Westinghouse's THINC codes, with which the WRB-1 CHF data were correlated, and any

* K is a sample size correction factor that gives a one-sided 95% upper confidence limit on the estimated standard deviation of a given population. It can be calculated as:

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

effect due to the modeling of a full core with a correlation based upon bundle test data. These uncertainties are clearly independent of the correlation, the model, and parameter induced uncertainties. The code uncertainty was quantified at 5%; consistent with the factors specified for other thermal/hydraulic codes in Reference 1. The basis for this uncertainty is described in detail by USNRC staff in Reference 8. In Reference 8, the USNRC Staff refers to the 5% uncertainty as being a 2$\sigma$ value. The 5% code uncertainty is certainly conservative in light of the excellent VIPRE-D/VIPRE-W and VIPRE-D/CHF data comparisons. However, the 5% uncertainty serves as a conservative factor that may be shown to be wholly or partially unnecessary at a later time. A one-sided 95% confidence level on the code uncertainty is then 3.04% (=5.0%/1.645). The use of the 1.645 divisor (the one-sided 95% tolerance interval multiplier) is conservative since the USNRC Staff considers the 5% uncertainty to be a 2$\sigma$ value.

## 3.6. Monte Carlo Calculations

In order to perform the Monte Carlo analysis, nine Nominal Statepoints covering the full range of normal operation and anticipated transient conditions were selected. These statepoints must span the range of conditions over which the statistical methodology will be applied. Two statepoints were selected at each of the four Reactor Core Safety Limit (RCSL) pressures (2425, 2250, 2000, and 1800 psia). For each of the RCSLs, a high power, 120%, and low power, near the intercept of the DNBR limit line with the vessel exit boiling line, were chosen. In order to apply the methodology to low flow events, a low flow statepoint is also included (Statepoint Z). The inlet temperature used for each statepoint is calculated by determining the inlet temperature that would result in the desired MDNBR (1.24) for each statepoint. The selected Nominal Statepoints are listed in Tables 3.6-1.

Table 3.6-1: Nominal Statepoints for Westinghouse 422V+ Fuel at Kewaunee with VIPRED-/WRB-1

| STATE POINT | PRESSURIZER PRESSURE [psia] | INLET TEMPERATURE [°F] | POWER [%] | FLOW [%] | $F_{\Delta H}^{N**}$ | MDNBR |
|---|---|---|---|---|---|---|
| 1 | 2425.0 | 572.1 | 120% | 100% | 1.635 | 1.243 |
| 2 | 2250.0 | 562.5 | 120% | 100% | 1.635 | 1.244 |
| 3 | 2000.0 | 548.3 | 120% | 100% | 1.635 | 1.244 |
| 4 | 1800.0 | 536.2 | 120% | 100% | 1.635 | 1.244 |
| 5 | 2000.0 | 580.9 | 100% | 100% | 1.635 | 1.243 |
| 6 | 1800.0 | 563.2 | 104% | 100% | 1.635 | 1.244 |
| 7 | 2250.0 | 606.6 | 90% | 100% | 1.684 | 1.241 |
| 8 | 2425.0 | 623.6 | 85% | 100% | 1.709 | 1.242 |
| 9 | 2250.0 | 541.2 | 100% | 63.5% | 1.635 | 1.247 |

The Monte Carlo analysis itself consisted of 2000 calculations performed around each of the nine Nominal Statepoints. As described in Section 3.1, the DNBR standard deviation at each Nominal Statepoint was augmented by the code/correlation uncertainty, the small

** The part-power multiplier described in the Kewaunee Core Operating Limits Report (COLR) is used for less than 100% power statepoints.

sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation.

The Total $s_{Total}$, is obtained using the Root-Sum-Square method according to Equation 3.2:

$$s_{TOTAL} = \sqrt{s_{DNBR}^2 \cdot \left(1.0 + \sqrt{\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}^2 + \left\{\frac{1}{\sqrt{N}}\right\}^2}\right)^2 + F_C^2 + F_M^2} \qquad \text{[Equation 3.2]}$$

where

- $s_{DNBR}$ is the standard deviation for the Randomized DNBR distribution.

- The factor $\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}$ is the uncertainty in the standard deviation of the 2,000 Monte Carlo simulations, and provides a 95% upper confidence limit on the standard deviation.

- $1/\sqrt{N}$ is the uncertainty in the mean of the correlation. N is the number of number of degrees of freedom in the original correlation database.

- $F_c$ is the code uncertainty, that has been defined as 5% ($2\sigma$ value), i.e. 5.0%/1.645 =3.04% ($1\sigma$ value). See Section 3.1.5 in Reference 1.

- $F_M$ is the model uncertainty, which is 0.0 in our case as we are running the Monte Carlo simulation with the production model (see Section 3.1.4 in Reference 8).

Note that this equation differs slightly from the equation listed in Reference 1. It has an additional factor applied to the Randomized DNBR $s_{DNBR}$, the $1/\sqrt{N}$ factor to correct for the uncertainty in the mean of the correlation. This factor has been used in previous implementations of the Statistical DNBR Evaluation Methodology, such as Reference 3 and Reference 4.

The limiting peak fuel rod SDL was calculated to be 1.24 for VIPRE-D/WRB-1. The Monte Carlo Statepoint analysis is summarized in Table 3.6-2.

Table 3.6-2: Peak Pin SDL Results for Kewaunee 422V+ with VIPRE-D/WRB-1

| STATEPOINT | Randomized DNB $s_{DNBR3}$ | Total DNB $s_{TOTAL}$ | Pin Peak $SDL_{95/95}$ |
|---|---|---|---|
| 1 | 0.1309 | 0.1398 | 1.230 |
| 2 | 0.1338 | 0.1428 | 1.235 |
| 3 | 0.1339 | 0.1429 | 1.235 |
| 4 | 0.1298 | 0.1386 | 1.228 |
| 5 | 0.1306 | 0.1395 | 1.229 |
| 6 | 0.1342 | 0.1432 | **1.236** |
| 7 | 0.1320 | 0.1409 | 1.232 |
| 8 | 0.1299 | 0.1388 | 1.228 |
| 9 | 0.1310 | 0.1398 | 1.230 |

## 3.7. Full Core DNB Probability Summation

After the development of the peak pin 95/95 DNBR limit, the data statistics are used to determine the number of rods expected in DNB. The DNB sensitivity to rod power is estimated as $\partial(DNBR)/\partial(1/F\Delta h)$. The specific values of $\partial(DNBR)/\partial(1/F\Delta h)$, denoted $\beta$, are listed in Table 3.7-1.

To ensure that the calculations are conservative, a one-sided tolerance limit of $\beta$ is used:

$$\beta^* = \beta - t(\alpha,\nu)\cdot se(\beta)$$

in which:

- $\beta^*$ is the one-sided tolerance limit on $\beta$
- $t(\alpha,\nu)$ is the T-statistic with significance level $\alpha$ and $\nu$ degrees of freedom. For 2,000 observations at a 0.05 level of significance $t(0.05,2000) = 1.645$.
- $se(\beta)$ is the standard error of $\beta$.

The variable $1/F\Delta h$ is the most statistically significant independent variable in the linear regression model, yielding $R^2$ values larger than 99%. The value of the statistic parameter F of $1/F\Delta h$ was the largest for all statepoints, which indicates that the variable $1/F\Delta h$ accounts for the largest amount of the variation in the DNBR.

Table 3.7-1: $\partial(DNBR)/\partial(1/F\Delta h)$ Estimation for WRB-1

| STATEPOINT | $\beta$ | se($\beta$) | $\beta^*$ | $R^2$ |
|---|---|---|---|---|
| 1 | 4.05387 | 0.00914 | 4.03884 | 99.5% |
| 2 | 4.18659 | 0.00861 | 4.17243 | 99.6% |
| 3 | 4.31365 | 0.00878 | 4.29921 | 99.6% |
| 4 | 4.33980 | 0.00951 | 4.32416 | 99.5% |
| 5 | 4.16476 | 0.00962 | 4.14894 | 99.5% |
| 6 | 4.14116 | 0.00951 | 4.12551 | 99.5% |
| 7 | 4.19940 | 0.00997 | 4.18301 | 99.5% |
| 8 | 4.03524 | 0.01006 | 4.01869 | 99.5% |
| 9 | 4.41089 | 0.00936 | 4.39549 | 99.5% |

A representative fuel rod census curve used for the determination of the SDL is listed in Table 3.7-2. The full core DNB probability summation will be reevaluated on a reload basis to verify the applicability of the fuel rod census ($F_{\Delta H}{}^{N}$ versus % of core with $F_{\Delta H}{}^{N}$ greater than or equal to a given $F_{\Delta H}$ limit) used in the implementation analysis. The DNB probability summation for VIPRE-D/WRB-1 is summarized in Table 3.7-3.

Table 3.7-2: Representative Fuel Rod Census
for a Maximum Peaking Factor FΔh = 1.635

| MAXIMUM % OF FUEL RODS IN CORE WITH FΔh ≥ to: | FΔh LIMIT |
|---|---|
| 0.0 | 1.6350 |
| 0.1 | 1.6330 |
| 0.2 | 1.6300 |
| 0.3 | 1.6270 |
| 0.4 | 1.6230 |
| 0.5 | 1.6194 |
| 0.6 | 1.6174 |
| 0.7 | 1.6154 |
| 0.8 | 1.6124 |
| 0.9 | 1.6104 |
| 1.0 | 1.6074 |
| 1.5 | 1.5977 |
| 2.0 | 1.5896 |
| 2.5 | 1.5805 |
| 3.0 | 1.5725 |
| 4.0 | 1.5594 |
| 5.0 | 1.5514 |
| 6.0 | 1.5453 |
| 7.0 | 1.5412 |
| 8.0 | 1.5362 |
| 9.0 | 1.5315 |
| 10.0 | 1.5271 |
| 20.0 | 1.4676 |
| 30.0 | 1.4213 |
| 40.0 | 1.3709 |
| **PEAK** | **1.635** |

Table 3.7-3: Full Core DNB Probability Summation for Kewaunee 422V+ with VIPRE-D/WRB-1

| STATEPOINT | $S_{TOTAL}$ | % of Rods in DNB | Full Core $SDL_{99.9}$ |
|---|---|---|---|
| **1** | 0.1398 | 0.0992 | 1.232 |
| **2** | 0.1428 | 0.0991 | 1.236 |
| **3** | 0.1429 | 0.0983 | 1.234 |
| **4** | 0.1386 | 0.0983 | 1.224 |
| **5** | 0.1395 | 0.0992 | 1.229 |
| **6** | 0.1432 | 0.0990 | **1.238** |
| **7** | 0.1409 | 0.0984 | 1.232 |
| **8** | 0.1388 | 0.0996 | 1.230 |
| **9** | 0.1398 | 0.0988 | 1.225 |

## 3.8. Verification of Nominal Statepoints

Condition 1 of the USNRC's safety evaluation report for Reference 1 (Reference 8) requires that the Nominal Statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may potentially be applied.

It is therefore necessary to demonstrate that $s_{total}$ as calculated herein is maximized for any conceivable set of conditions at which the core may approach the SDL. To do so, a regression analysis is performed using as dependent variable the unrandomized DNBR standard deviations at each Nominal Statepoint (i.e. the raw MDNBR results obtained from the Monte Carlo simulation). The Nominal Statepoint pressures, inlet temperatures, powers and flow rates are used as the independent variable. If no clear trend appears in the plot it can be concluded that the standard deviation has been maximized. If a clear trend is displayed, the regression function is determined. This regression equation is evaluated to determine the values of the independent variable for which the standard deviation would be maximized, and it is verified that the Nominal Statepoints selected bound those conditions. In addition, the residuals of the regression are plotted again against all the independent variables, and it is verified that no trends are discernible.

Table 3.8-1 shows the $R^2$ coefficients obtained for the verification of the nominal statepoints. The largest linear curve fit $R^2$ coefficient is 9.33%, thus confirming that there is not dependence.

An evaluation of all the data, linear fits, and $R^2$ coefficients indicates that there are no discernible trends in the database. Therefore, it was concluded that $s_{TOTAL}$ had been maximized for any conceivable set of conditions at which the core may approach the SDL and that the selected Nominal Statepoints provide a bounding standard deviation for any set of conditions to which the methodology may potentially be applied. Figure 3.8-1 displays a sample regression plot for WRB-1 and clearly shows the trends discussed above.

Table 3.8-1: $R^2$ Coefficients for the Verification of the Nominal Statepoints for Kewaunee 422V+ with VIPRE-D/WRB-1

| | $R^2$ Linear Regression |
|---|---|
| **Pressure** | 6.97% |
| **Temperature** | 4.93% |
| **Flow Rate** | 8.33% |
| **Power** | 9.33% |

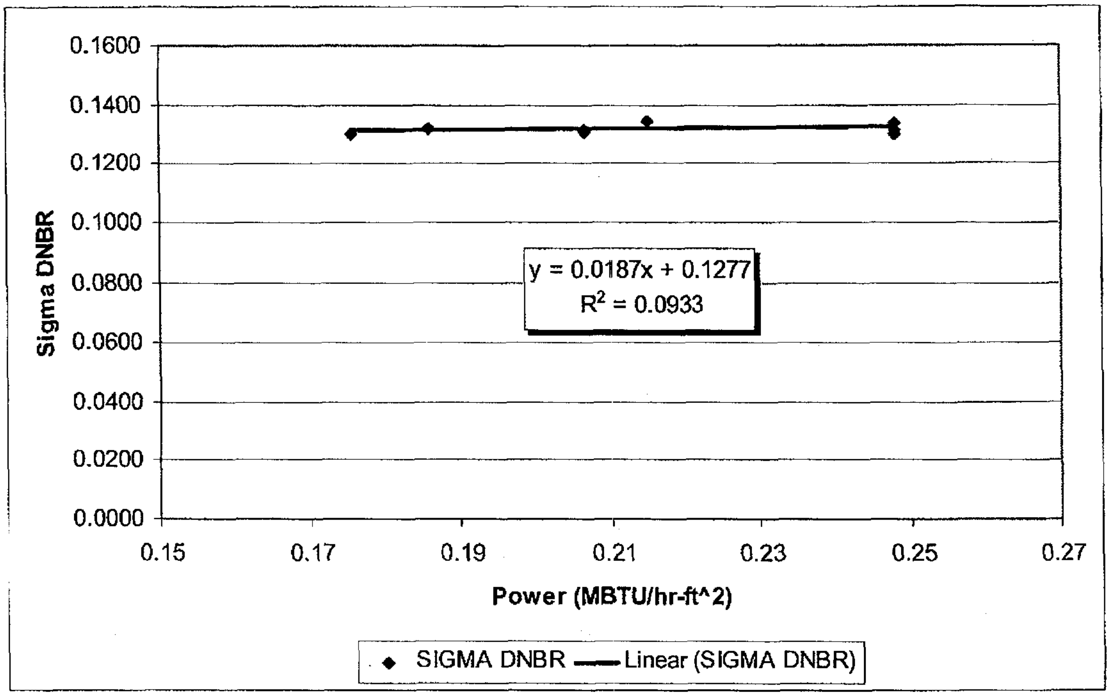


Figure 3.8-1: Variation of the Unrandomized Standard Deviation with Power for the WRB-1 CHF Correlation

## 3.9. Scope of Applicability

This section is included herein to satisfy Condition 4 in the SER (Reference 8) of VEP-NE-2-A (Reference 1).

The Statistical DNBR Evaluation Methodology may be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical (RWFS) which is initiated from zero power), and to the Loss of Flow analysis and the Locked Rotor Accident. The accidents to which the methodology is applicable are listed in Table 3.9-1. This table corresponds to Table 3.5.1 in Reference 7. The range of application is consistent with previous applications of Dominion Statistical DNBR Evaluation Methodology for other Dominion PWR applications (North Anna and Surry). This methodology will not be applied to accidents that are initiated from zero power where the parameter uncertainties are higher.

The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and hot channel uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

Table 3.9-1: USAR Transients Analyzed with VIPRE-D/WRB-1/W-3 for Kewaunee (Reference 7, Table 3.5.1)

| ACCIDENT | KPS USAR SECTION | APPLICATION |
|---|---|---|
| Rod cluster control assembly bank withdrawal from subcritical | 14.1.1 | DET-DNB |
| Rod cluster control assembly bank withdrawal at power | 14.1.2 | STAT-DNB |
| Rod cluster control assembly misalignment / Dropped rod/bank | 14.1.3 | STAT-DNB |
| Uncontrolled boron dilution | 14.1.4 | Non-DNB |
| Full and partial loss of forced reactor coolant flow | 14.1.8 | STAT-DNB |
| Startup of an inactive reactor coolant loop | 14.1.5 | Non-DNB |
| Loss of external electrical load and/or turbine trip | 14.1.9 | STAT-DNB |
| Loss of normal feedwater | 14.1.10 | Non-DNB |
| Loss of offsite power | 14.1.12 | Non-DNB |
| Excessive heat removal due to feedwater system malfunction | 14.1.6 | STAT-DNB |
| Excessive load increase | 14.1.7 | STAT-DNB |
| Rupture of a main steam pipe | 14.2.5 | DET-DNB |
| Locked reactor coolant pump rotor or shaft break | 14.1.8 | STAT-DNB |

## 3.10. Summary of Analysis

The steps of the SDL derivation analysis may be summarized as follows:

In accordance with the Statistical DNBR Evaluation Methodology, 2,000 random statepoints are generated about each nominal statepoint and VIPRE-D is then executed to obtain MDNBRs. The standard deviation for the distribution of 2,000 MDNBRs is referred to as the unrandomized standard deviation. At the limiting Nominal Statepoint (W), the standard deviation of the randomized DNBR distributions, which is the unrandomized corrected for CHF correlation uncertainty, was found to be 0.1342. This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation of 0.1432, as listed in Table 3.6-2. The use of 0.1432 in Equation 3.1 yields a peak pin DNBR limit of 1.236 with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core of 1.238. Therefore the VIPRE-D/WRB-1 SDL for Westinghouse 422V+ fuel is set to 1.24.

# 4. Application of VIPRE-D/WRB-1/W-3 to KPS

VIPRE-D/WRB-1 together with the Statistical DNBR Evaluation Methodology will be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWFS), and to the Complete Loss of Flow event and the Locked Rotor Accident. The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and $F_{\Delta H}^{E}$ uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

In addition, there are a few events that will be evaluated with the VIPRE-D/W-3 and deterministic models because they do not meet the applicability requirements of the Statistical DNBR Evaluation Methodology (see Table 3.9-1, DET-DNB events). These events will be initiated from bounding operating conditions considering the nominal value and the appropriate uncertainty value, and require the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and $F_{\Delta H}^{E}$ uncertainties. The events modeled deterministically are limited by the deterministic design limit (DDL) stated in DOM-NAF-2-A (Reference 2).

## 4.1. VIPRE-D/WRB-1 Statistical Design Limit (SDL) for Kewaunee

The Statistical Design Limit for Kewaunee cores containing Westinghouse 422V+ fuel assemblies with the VIPRE-D/WRB-1 code was derived in Section 3 of this report. The SDL for VIPRE-D/WRB-1 is determined to be 1.24. The SDL limit provides a peak fuel rod DNB protection with at least 95% probability at a 95% confidence level and a 99.9% DNB protection for the full core. This SDL is plant specific as it already includes the Kewaunee specific uncertainties for the key parameters accounted for in the application of the Statistical DNBR Evaluation Methodology. Therefore, this limit is applicable to the analysis of statistical DNB events of Westinghouse 422V+ fuel in Kewaunee cores with the VIPRE-D/WRB-1 code.

## 4.2. Verification of Existing Reactor Core Safety Limits, Protection Setpoints and KPS USAR Chapter 14 Events

This section is included herein to satisfy Condition 3 in the SER (Reference 8) of VEP-NE-2-A (Reference 1).

To demonstrate that the DNB performance of the Westinghouse 422V+ fuel is acceptable, Dominion performed calculations for full-core configurations of Westinghouse 422V+ fuel. The calculations were performed using the VIPRE-D/WRB-1 and VIPRE-D/W-3 code/correlation pairs and selected statepoints including: the reactor core safety limits (RCSL), axial offset limits (AO), rod withdrawal from subcritical (RWFS), rod withdrawal at power (RWAP), loss of flow (LOFA), locked rotor events (LOCROT), hot zero power steam line break (MSLB), dropped rod limit line (DRLL), and static rod misalignment (SRM). 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 (RCSL); (b) limiting axial flux shapes at several axial offsets (AO); and (c) low flow (LOFA and LOCROT). The statepoints for the RWFS and MSLB were evaluated with deterministic DNB methods. The remaining statepoints were evaluated using statistical DNB methods. The evaluation criterion for these analyses is that the minimum DNBR must be equal to or greater than the applicable safety analysis limit (SAL) listed below.

The results of the calculations demonstrate that the minimum DNBR values are equal to or greater than the applicable safety analysis limit for all of the analyses that are performed to address statepoints of the Reactor Core Safety Limits, the OT$\Delta$T, OP$\Delta$T and F$\Delta$I trip setpoints, as well as all the evaluated Chapter 14 events (including the LOFA and LOCROT) with an $F\Delta h$ of 1.635 (COLR limit of 1.7 divided by the measurement uncertainty of 1.04 = 1.635).

| **VIPRE-D/WRB-1** | |
|---|---|
| DDL | 1.17 |
| SDL | 1.24 |
| SAL | 1.31 |
| **VIPRE-D/W-3** | |
| DDL (≥1000 psia) | 1.30 |
| DDL (<1000 psia) | 1.45 |
| SAL (≥1000 psia) | 1.42 |
| SAL (<1000 psia) | 1.58 |

# 5. Conclusions

This report supports the application of USNRC approved VEP-NE-2-A to KPS. It provides the technical basis and documentation required to evaluate the plant specific application of VEP-NE-2-A methods to KPS. This application employs the VIPRE-D code with the Westinghouse WRB-1 Critical Heat Flux (CHF) correlation (VIPRE-D/WRB-1) for the thermal-hydraulic analysis of Westinghouse 422V+ fuel assemblies at KPS. In particular, Dominion requests the review and approval of the Statistical Design Limit (SDL) of 1.24 as documented herein as per 10 CFR 50.59(c)(2)(vii) it constitutes a Design Basis Limit for Fission Products Barrier (DBLFPB).

## 6. References


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

2. Fleet Report, DOM-NAF-2, Rev 0.0-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," R. M. Bilbao y León, August 2006.

3. Letter from E. S. Grecheck (VEPCO) to US NRC Document Control Desk, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 - Proposed Technical Specification Changes Addition of Analytical Methodology to COLR," Serial No. 05-419, July 5, 2005.

4. Letter from W. L. Stewart (VEPCO) to US NRC Document Control Desk, "Virginia Electric and Power Company Surry Power Station Units 1 and 2 - Proposed Technical Specification Changes - FΔh Increase/Statistical DNBR Methodology," Serial No. 91-374, July 8, 1991.

5. Letter from Mark Warner (NMC) to US NRC Document Control Desk, "Kewaunee Nuclear Power Plant License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel", Serial No. 02-067, July 2, 2002.

6. Technical Report, WCAP-15591, Rev. 1 (proprietary), "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology Kewaunee Nuclear Plant," December 2002.

7. Letter from G. T. Bischof (Dominion) to Document Control Desk (USNRC), "Request for Approval of Topical Report DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," Serial No. 06-578 dated August 16, 2006.

8. Letter from L. B. Engle (NRC) to W. L. Stewart (Virginia Power), "Statistical DNBR Evaluation Methodology, VEP-NE-2, Surry Power Station, Units No. 1 & No. 2 (Surry-1&2) and North Anna Power Station, Units No. 1 & No. 2 (NA-1&2)," Serial No. 87-335 dated May 28, 1987.