Designated Original

Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion®

August 16, 2006

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

Serial No. 06-578
NL&OS/CDS: R-1
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
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)"

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

On June 13, 2006, DEK and NRC staff engaged in subsequent discussions that focused on the content of the topical report. Attachment 1 provides the subject topical report, DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)." Dominion Energy Kewaunee, Inc. intends to submit supplemental material to support the NRC review of this document, in the form of Attachments A and B to DOM-NAF-5, in subsequent transmittals. The current schedule is to submit Attachment A by September 30, 2006 and Attachment B by January 31, 2007. An additional submittal (scheduled for transmittal by January 31, 2007) will provide a license amendment request (LAR) to include DOM-NAF-5 among the reference methodology reports in the KPS Technical Specifications.

In order to support application of these methods to KPS Cycle 29, DEK requests NRC staff review and approval of DOM-NAF-5 by July 15, 2007. The requested date for NRC staff approval of the forthcoming LAR is anticipated to be January 31, 2008.

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

Very truly yours,

Gerald T. Bischof
Vice President - Nuclear Engineering

A001

Reference:

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), (Accession Number ML 060400098).

Attachment:

1. DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," dated July 2006.

Commitments made in this letter: None

cc: Mr. J. L. Caldwell
Administrator, Region
U. S. Nuclear Regulatory Commission
Region III
2443 Warrenville Road
Suite 210
Lisle, Illinois 60532-4352

Mr. D. H. Jaffe
Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O-7-D-1
Washington, D. C. 20555

Mr. S. C. Burton
NRC Senior Resident Inspector
Kewaunee Power Station

Attachment 1

TOPICAL REPORT DOM-NAF-5

APPLICATION OF DOMINION NUCLEAR CORE DESIGN AND SAFETY ANALYSIS METHODS TO THE KEWAUNEE POWER STATION (KPS), DATED JULY 2006

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)

NUCLEAR ANALYSIS AND FUEL DEPARTMENT
DOMINION
RICHMOND, VIRGINIA
July, 2006

Prepared by:

Rosa M. Bilbao y León
Gary L. Darden
Kurt F. Flaig
Robert A. Hall
John T. Holly
Noval A. Smith

Recommended for Approval:

Joel R. Harrell

J. R. Harrell
Supervisor, Nuclear Fuel Projects

Approved:

Kerry L. Basehore

K. L. Basehore
Director, Nuclear Analysis and Fuel

Classification/Disclaimer

Abstract

DOM-NAF-5 is a Dominion topical report that documents justification for application of Dominion nuclear core design and safety analysis methods to Kewaunee Power Station (KPS). This report:

a) Describes Dominion nuclear core design and safety analysis methods, and
b) Documents assessments of the applicability of Dominion nuclear core design and safety analysis methods to KPS.

Based on the applicability assessments of the Dominion nuclear core design and safety analysis methods described herein, the Dominion methods were determined to be applicable to KPS, and can be employed in design and licensing analyses for KPS. The applicability of certain methods (i.e., VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code"; and DOM-NAF-1, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations") require further demonstration through detailed validation analyses that supplement DOM-NAF-5 (see Attachments A and B). A License Amendment Request (LAR), including a plant-specific and fuel-specific application analysis to define a Departure from Nucleate Boiling Ratio (DNBR) Statistical Design Limit (SDL), is required to support implementation of these methods at KPS. The LAR will request addition of Dominion topical reports, through DOM-NAF-5, as reference methodologies in the KPS Core Operating Limits Report (COLR) and in Technical Specification (TS) 6.9.a.4. Other conforming Technical Specification changes are to be incorporated into the LAR, as needed to reflect use of Dominion methods.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF ATTACHMENTS

1.0 Introduction

Kewaunee Power Station (KPS) became part of the Dominion nuclear fleet following Dominion's acquisition of KPS in July 2005. In addition to KPS, the Dominion nuclear fleet presently includes Surry Power Station (SPS), Millstone Power Station (MPS), and North Anna Power Station (NAPS). Dominion nuclear core design and safety analysis methods were developed for application to the original Dominion nuclear power stations (SPS and NAPS) in the 1980's. Over the years, these analysis methods have been successfully applied in numerous analytical, operational, and regulatory support activities.

DOM-NAF-5 is a Dominion topical report that documents justification for application of Dominion nuclear core design and safety analysis methods to KPS. This report:

a) Describes Dominion nuclear core design and safety analysis methods, and
b) Documents assessments of the applicability of the Dominion nuclear core design and safety analysis methods to KPS.

Section 2.0 of the report identifies the analysis methods that are in the scope of application considered for this report. The following methods are outside the scope of application considered in this report:

a) Containment response and containment integrity analysis methods
b) Radiological analysis methods
c) Fuel rod design and analysis methods (Note: transient fuel rod thermal response for specific transient events is in scope. The transient fuel rod thermal response is to be calculated using the approved RETRAN hot-spot model as described in Reference 1. With this exception, the responsibility for fuel rod design calculations is to reside with the fuel vendor using the approved methods described in the Core Operating Limits Report.)
d) Small break and large break loss of coolant accident (LOCA) analysis methods
e) Control Rod Ejection analysis methods

The reload design and safety analysis process performed by the current KPS fuel supplier (Westinghouse) is essentially the same process as the Dominion process, but it is performed with approved Westinghouse design and analysis methods. The current KPS Core Operating Limits Report (COLR) references the approved Westinghouse design and analysis methods for KPS. These Westinghouse design and analysis methods will remain applicable to KPS, and Dominion intends to retain the Westinghouse methods in the COLR after approval of DOM-NAF-5 to facilitate orderly transition to Dominion analyses.

Section 3.0 of this report describes the various in-scope design and analysis methodologies, and documents assessments of the applicability of those methodologies to KPS. Section 4.0 presents the conclusions derived from the methods applicability assessments.

As described herein, Dominion nuclear core design and safety analysis methods have been determined to be applicable to KPS, and can be employed in design and licensing analyses for KPS. The applicability of certain methods (e.g., VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code"; and DOM-NAF-1, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations") requires further demonstration through detailed validation analyses that supplement DOM-NAF-5 (see Attachments A and B). A License Amendment Request (LAR), including a plant-specific and fuel-specific application analysis to define a Departure from Nucleate Boiling Ratio (DNBR) Statistical Design Limit (SDL), is required to support implementation of these methods at KPS. The LAR will request addition of Dominion Topical Reports, through DOM-NAF-5, as reference methodologies in the KPS Core Operating Limits Report (COLR) and in Technical Specification (TS) 6.9.a.4. Other conforming Technical Specification changes are to be incorporated into the LAR, as needed to reflect use of Dominion methods.

2.0 Dominion Nuclear Core Design and Safety Analysis Methodologies

2.1 Dominion Methods to be Applied to KPS

Dominion currently applies its nuclear core design and safety analysis methods to its nuclear power stations, while the fuel vendor is responsible for fuel design analyses and reload fuel performance assessments. Dominion has performed reload design and safety analyses for approximately 65 reload cores at Surry and North Anna using both vendor and Dominion-developed tools. Dominion will apply the Dominion nuclear core design and safety analysis methods to KPS in the same manner it applies these methods to the other plants in the fleet. The KPS fuel vendor will retain responsibility for licensing the fuel design, for performing fuel rod design analysis, and for reload fuel performance assessment. For KPS, the fuel vendor will also perform certain specific safety analyses, e.g. small break and large break LOCA analyses.

Dominion has established a process for control and maintenance of its NRC-approved nuclear core design and safety analysis methodologies. Section 2.3 of Topical Report VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," (Reference 6) refers to this process. This process was further defined in responses to Requests for Additional Information (RAIs) on Dominion's Reload Nuclear Design Methodology (Reference 23). The NRC reviewed the Dominion analysis methods control and maintenance process and found it acceptable, as discussed in their Safety Evaluation Report (SER) for Dominion's Reload Nuclear Design Methodology (Reference 7). The Dominion analysis methods applied to KPS are to be controlled and maintained using these approved processes.

The Dominion nuclear core design methods within the scope of this report are:

a) VEP-FRD-42 (current version: VEP-FRD-42, Rev. 2.1-A), "Reload Nuclear Design Methodology" (Reference 6)

b) VEP-NE-1 (current version: VEP-NE-1, Rev. 0.1-A), "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications" (Reference 8)

c) DOM-NAF-1 (current version: DOM-NAF-1, Rev. 0.0-P-A), "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations" (Reference 9)

The Dominion safety analysis methods within the scope of this report are:

d) VEP-FRD-41 (current version: VEP-FRD-41, Rev. 0.1-A), "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code" (Reference 1)

e) VEP-NE-2 (current version: VEP-NE-2-A), "Statistical DNBR Evaluation Methodology" (Reference 3)

f) DOM-NAF-2 (current version: DOM-NAF-2), "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code" (Reference 2)

Each of the above methods is assessed for applicability to KPS in Section 3.0 using the Applicability Assessment Methodology described below in Section 2.2. Throughout the remainder of this report, each of these reports is cited without reference to the revision. Section 5.0, References, cites the current report versions as of the date of this report (July 2006).

2.2 Applicability Assessment Methodology

Dominion analysis methods are to be applied to KPS in a manner consistent with the conditions and limitations described in the Dominion topical reports and in applicable NRC Safety Evaluation Reports (SERs). Any differences from the methods described in the Dominion Topical Reports that are required for application of the methods to KPS are identified and addressed through the Applicability Assessment Methodology described herein.

The following systematic evaluation process is applied herein to assess the application of candidate methodologies to KPS:

a) Each method is described, including its purpose, key features, and dependencies. Descriptions include:

- Key phenomena/conditions predicted by the method
- General calculation approach or assumptions
- Types of reactor conditions for which the method is used

b) Conditions and limitations associated with each method are identified, including:

- Regulatory limitations in NRC Safety Evaluation Reports (SER)
- Physical limitations (e.g., plant systems, plant features & conditions)
- Limitations in Dominion Topical Reports (e.g., specific modeling approaches or inherent assumptions)

c) Each method is assessed with respect to the identified Conditions and Limitations. The assessment effort ranges from written evaluations, to validation and benchmark analyses and detailed comparisons of results.

d) The results of the applicability assessment are documented for each method. Results from some of the more involved assessment efforts are presented in Attachments A and B.

3.0 Applicability Assessments

3.1 Applicability Assessment of Reload Nuclear Design Methods - VEP-FRD-42, "Reload Nuclear Design Methodology"

3.1.1 Description of Methodology

The Dominion reload nuclear design methodology includes calculational and process elements that are employed in the design and evaluation of reload nuclear cores. The major activities of the methodology are: 1) determination and fulfillment of cycle energy requirements; 2) determination of a core loading pattern; and 3) a reload safety evaluation that confirms acceptable behavior for the reload core under predicted design basis accident conditions.

The Dominion reload nuclear design methods, as documented in VEP-FRD-42 (Reference 6), consist of the following elements:

a) Analytical Models (e.g., Studsvik Core Management System (CMS) Models, VEPCO RETRAN Models, Core Thermal-Hydraulics VIPRE-D Models)
b) Analytical Methods (e.g., Core Depletions, Core Reactivity Parameters and Coefficients, Core Reactivity Control, Safety Analysis, Statistical DNB)
c) Reload Design Process (e.g., Core Loading Pattern Design & Optimization, Key Parameter Treatment in Nuclear Design Analyses)
d) Reload Safety Evaluation Process
e) Nuclear Design Report, Operator Curves & Core Follow Data

The Dominion methodology for designing a reload core is an iterative process. The process involves determining a core loading pattern which provides the required total cycle energy and then demonstrating through analysis or evaluation that the plant will continue to meet all applicable safety criteria after considering the changes associated with the reload core. Should the characteristics of the proposed loading pattern cause any safety analysis criteria to be exceeded, the loading pattern is revised.

Reload safety evaluation and analysis criteria are established using a bounding analysis concept. This approach employs a list of key analysis parameters with the limiting direction of each parameter identified. This allows reload core characteristics to be compared with the parameter values assumed in the reference analyses for various transients and accidents. For a proposed core reload design, if all key analysis parameters are conservatively bounded, then the reference safety analysis applies, and no further analysis is necessary. If one or more key analysis parameters are not bounded, then further analysis or evaluation of the transient or accident in question is performed. Occasionally, the applicable safety analyses are revised, or changes are made in the operating requirements (e,g., Technical Specifications or Core Operating Limits Report (COLR) changes) to ensure that plant operation will satisfy the applicable safety analysis criteria for the proposed loading pattern.

Topical Report WCAP-9272, "Westinghouse Reload Safety Evaluation," dated March 1978 (Reference 10) describes the Westinghouse methodology used for reload safety evaluation. WCAP-9272 forms the basis for Dominion's reload methodology as described in Topical Report VEP-FRD-42. The Westinghouse methodology defines the specific key parameters for use in accident analyses and provides limiting directions for consideration in reload evaluations.

The reload core design is evaluated by comparing the reload core parameters against the assumptions in the current safety analyses. Safety analysis (accident analysis) is the study of nuclear reactor behavior under

accident conditions. The accident analyses consider all relevant aspects of the plant and core including the operating procedures and limits on controllable plant parameters and the engineered safety, shutdown, and containment systems.

There are two stages in the typical safety analysis process. First, steady state nuclear calculations are performed for the core conditions assumed in the accident analysis. The nuclear parameters derived from these calculations are called the core physics related key analysis parameters and serve as input to the second stage. The second stage is the actual dynamic accident analysis, which yields the accident results that are applicable for these key analysis parameter values. The accident analyses are transient calculations that usually model the core nuclear kinetics and those parts of the plant systems that have a significant impact on the events under consideration.

The Kewaunee Updated Safety Analysis Report (USAR) documents acceptable plant safety via detailed results of accident analyses performed with the bounding values of key analysis parameters. Plant safety is demonstrated if accident analysis results meet the applicable acceptance criteria. The reload core design is evaluated by comparing the core physics related key analysis parameters against the assumptions in the current safety analyses. The reload evaluation process is complete if the acceptance criteria delineated in the USAR are satisfied with the reload core implemented. If an accident reanalysis is necessary, more detailed analysis methods and/or Technical Specifications changes may be required to meet the acceptance criteria. Such changes are to be processed in accordance with the applicable regulatory processes.

In summary, the overall reload evaluation process includes the following steps:

a) Determine bounding key analysis parameters, which constitute the current limits for reload cores.
b) Perform (or confirm) accident analysis using the bounding key analysis parameters and conservative assumptions.
c) Establish a proposed core loading pattern that provides the required total cycle energy.
d) Determine, for the proposed core loading pattern, the value for each key analysis parameter.
e) Compare key reload analysis parameters to current limits.
f) Evaluate whether an accident reanalysis is needed based on the effect the reload key analysis parameters may have on the accident.
g) Perform reanalysis of specific affected accidents, change operating limits, or revise the loading pattern, as necessary.

Key attributes of the reload nuclear design methodology include the following elements:

a) An analysis framework in which safety analyses establish acceptable limit values for reload core key analysis parameters, while nuclear and fuel design codes confirm each core's margin to these limits
b) The use of bounding key parameter values in reference safety analyses
c) Recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data
d) Representation of key fuel features via detailed inputs in core design and safety analysis models
e) Fuel modeling using approved critical heat flux correlations demonstrated to be applicable and within the range of qualification and identified in the plant COLR section of the TS

3.1.2 Conditions and Limitations

There are specific and inherent conditions and limitations that are associated with application of the methods documented in VEP-FRD-42 to KPS:

a) Regulatory limitations in NRC Safety Evaluation Reports (SER)

i. Inherent limitation for use at North Anna and Surry Power Stations
ii. Prior to its use for fuel types other than Westinghouse and Framatome ANP Advanced Mark-BW fuel, confirm that the impact of the fuel design and its specific features can be accurately modeled with the VEPCO nuclear design and safety analysis codes and methods. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42, these proposed changes are required to be submitted for prior NRC review and approval.

b) Physical limitations (e.g., plant systems, plant features & conditions)

None identified (The methods of VEP-FRD-42 are not dependent on such physical limitations)

c) Limitations in Dominion Topical Reports (e.g., specific modeling approaches or inherent assumptions)

None identified

3.1.3 Assessment

The Dominion reload nuclear design methods and the current KPS reload nuclear design methods are both based on the Westinghouse Topical Report WCAP-9272. Thus, the Dominion and KPS reload nuclear design methods are similar since they have a common basis in the Westinghouse reload safety evaluation methods.

KPS and the other Dominion Westinghouse nuclear units use Westinghouse designs for nuclear steam supply system (NSSS) and reactor protection system (RPS). KPS and the other Dominion Westinghouse units have many design and operating similarities. The specific differences in NSSS, RPS and fuel features for KPS are all capable of being reflected via modeling inputs in the analytical methods of VEP-FRD-42. These differences do not impact the execution of the key VEP-FRD-42 methodology elements for the design and evaluation of reload cores.

The reload safety evaluation and analysis process for Dominion and KPS uses a bounding analysis concept. The method that is used for Dominion and KPS employs a list of key analysis parameters and limiting directions of the key analysis parameters for various transients and accidents.

The key analysis parameters and the limiting directions of those key parameters for the various transients and accidents were evaluated. The key analysis parameters and their limiting direction for KPS are the same as the key analysis parameters and limiting direction for the other Dominion Westinghouse units. The design basis accidents and transients in the safety analyses were also evaluated. The design basis transients and accidents for KPS are similar to the design basis transients and accidents for the other Dominion Westinghouse units. Based on the key analysis parameters assessment, and the design basis transients and accidents that are evaluated in the reload safety evaluation process, the Dominion reload safety evaluation and analysis methods have been determined to be applicable to KPS.

3.1.4 Summary

Dominion reload nuclear design methods documented in VEP-FRD-42 are concluded to be applicable to KPS and can be applied to KPS licensing analysis for reload core design and reload safety evaluation.

3.2 Applicability Assessment of Relaxed Power Distribution Control Methods – VEP-NE-1, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications"

3.2.1 Description of Methodology

The relaxed power distribution control (RPDC) method is the Dominion method for axial power distribution control. RPDC involves a variable axial flux difference (delta-I) band power distribution control strategy that uses a widened full power delta-I band, and provides for an increasing delta-I band with decreasing power. The widened delta-I band is based on maintaining an approximately constant analysis margin to the design bases limits at all power levels.

RPDC provides several benefits to plant operation, described as follows. The ability to return to power after a trip, particularly at end-of-cycle (EOC), is enhanced. Control rod motion necessary to compensate for delta-I band restrictions is reduced to only that motion needed to maintain operation within a much wider band. The reactor coolant system boration and dilution requirements are decreased due, in part, to the reduced control rod motion. There is generally an enhancement of operational flexibility. The RPDC methodology allows the Dominion units to operate with additional flexibility while at the same time ensuring that the design bases limits are met with an appropriate margin.

RPDC also involves the formulation of Technical Specification surveillance and COLR limits for Total Peaking Factor (FQ). The FQ surveillance uses the measured core axial position-dependent FQ (FQ(Z)) augmented by a non-equilibrium operation multiplier (N(Z)) in order to verify compliance with the peaking factor limits. This FQ surveillance is a required element of the RPDC method.

The objective of the RPDC analysis is to determine acceptable delta-I bands that maintain margin to all the applicable design bases criteria and at the same time provide enhanced delta-I operating margin. Because the RPDC delta-I band is an analysis output quantity rather than a fixed input, power shapes that adequately bound the potential delta-I range must be generated. These power shapes must include the effect of combinations of the key parameters such as burnup, control rod position, xenon distribution, and power level. Dominion has developed the methodology to generate the large number of power shapes required for RPDC analyses.

After the power shapes have been created, proposed delta-I bands are chosen such that all shapes within the delta-I bands satisfy the COLR FQ(Z) limit. For the normal operation analysis, power levels spanning the 50% to 100% range are investigated to establish the RPDC delta-I limits. Further verification of the proposed delta-I bands is performed via two different limiting shape evaluations, one based on Loss of Coolant Accident (LOCA) FQ considerations and the other based on a Loss of Flow Accident (LOFA) thermal/hydraulic evaluation. Delta-I bands can be narrowed to satisfy the requirements of these evaluations, if necessary.

Condition II or Abnormal Operation events, which may be the result of system malfunctions or operator errors, are also analyzed for RPDC. This RPDC analysis examines the more limiting of these Condition II events and verifies on a cycle-to-cycle basis that the Over-Power Delta-T (OPΔT) and the Over-Temperature Delta-T (OTΔT) setpoints are conservative. The OPΔT and OTΔT setpoints were designed primarily to provide transient and steady state protection against fuel centerline melt and DNB, respectively.

The RPDC methodology takes advantage of the large amount of margin to the design bases limits available at reduced power levels and provides delta-I limits at all power levels that are less restrictive than under Constant Axial Offset Control (CAOC) operation. The RPDC methodology may be summarized as follows:

a) A full range of normal-operation power shapes is obtained by combining the key parameters upon which each shape is dependent: xenon distribution, core burnup, boron concentration, core power level and control rod position. Reasonable increments spanning the entire range of values are considered for each of these parameters. A xenon "free oscillation" method is used to create the many and varied axial xenon distributions required for this analysis.

b) Proposed delta-I bands are selected such that the COLR FQ(Z) limit is met for all power shapes within the proposed bands. These power shapes are then analyzed to determine which shapes result in an approach to the LOFA limits.

c) Final normal operation delta-I bands are established such that the LOCA and the LOFA limits are satisfied.

d) Conditions that yield shapes within the final normal operation delta-I bands are used as initial conditions for the bounding Condition II accident simulations.

e) The resultant transient shapes are analyzed and the OPΔT and OTΔT trip function/setpoints are verified to ensure that margin to fuel design limits is maintained.

f) N(Z) functions (non-equilibrium power distribution multiplier) are formulated based on the maximum composite calculated Condition I FQ(Z) x P (i.e., local FQ times total core thermal power) to support the implementation of FQ Technical Specifications surveillance.

All neutronic calculations are performed with NRC-approved codes and methods. All DNBR calculations are performed using NRC-approved thermal-hydraulic code(s), correlation(s), and methods.

3.2.2 Conditions and Limitations

There are no specific conditions or limitations specified in the RPDC SER (Reference 8) for use of this methodology. Commonality amongst the Dominion, Westinghouse, CE, and Exxon versions of this methodology is noted throughout the RPDC SER. The following comments from the RPDC SER are relevant to the assessment of use of the RPDC method for KPS:

a) Approved methods were used for analyses supporting RPDC.
b) Justification was provided for the uncertainties assigned.
c) Impact of cycle specific variations on the delta-I power domain, OPΔT and OTΔT trip setpoints, and other safety analyses will be evaluated on a reload basis.
d) RPDC is an acceptable methodology for application to reload cores that are similar to those of the Surry (SPS) and North Anna (NAPS) reactors.

The following methodology items must be evaluated to determine if the values stated are applicable to KPS cores:

e) The appropriate plant-specific value for the maximum calculated temperature reduction during an EOC cooldown transient is to be determined for Kewaunee. A cooldown of 20°F was shown to be bounding for North Anna.
f) A conservative relationship is established for the Technical Specification/COLR FQ surveillance such that a 1% increase in FQ is mitigated by a 1% narrowing of the delta-I bands or a 1% reduction in core power (with commensurate reductions in trip setpoints).

g) A dilution time of 15 minutes is assumed after the control rods pass the insertion limit for the boron dilution event.
h) The stated combined calculation uncertainty (FQU) for predicted FQ is 1.0815.

3.2.3 Assessment

Conditions a-c cited in Section 3.2.2 are met for use of RPDC at KPS because the same methods, uncertainty parameters, and analyses will be employed as are currently employed for North Anna RPDC analyses. The specific KPS uncertainty for calculated FQ (FQU, conditions b and h) will be developed as part of KPS validation analysis for the CMS methods (see applicability assessment for DOM-NAF-1 methodology in Section 3.3). Condition d is satisfied due to the many similarities between KPS, SPS, and NAPS. All three stations use Westinghouse designs for the nuclear steam supply system (NSSS) and reactor protection system (RPS). In addition, KPS reload cores are similar to those at SPS because 14x14 fuel and 15x15 fuel have nearly identical fuel pin design and pin pitch (see comparison in Table 3.3.1). The applicability of the RPDC methodology to KPS is also supported by the fact that the Westinghouse Relaxed Axial Offset Control (RAOC) method is currently used by KPS. There are a number of similarities between Dominion's RPDC method and the Westinghouse RAOC power distribution control method (Reference 11), including:

a) Technical Specifications/COLR Delta I bands (limits) and variation of delta-I bands versus power level
b) FQ surveillance requirements and FQ limits are in the Technical Specifications and/or COLR
c) Normal (Condition I) and abnormal (Condition II) events are considered
d) Skewed axial xenon distributions are used to generate Condition I axial power shape variations
e) Combinations of key variables are used (control rod insertion, core burnup, and power level)
f) Cycle specific evaluations are performed for delta-I bands vs. power level
g) Cycle specific evaluations are performed for OPΔT, OTΔT setpoint verification
h) RAOC non-equilibrium multiplier W(Z) is analogous to the RPDC N(Z) function

With regard to the applicability of specific values in the RPDC methodology to KPS (conditions e-h), the cooldown transient assumption of 20°F (condition e) is smaller than the 30°F value in the Westinghouse method currently used for KPS. A cooldown limit of 30°F shall be used unless a KPS-specific analysis can demonstrate that a plant trip will occur prior to reaching 30°F. KPS Technical Specification 3.10.b requires a reduction of 1% in the delta-I bands (axial flux distribution-AFD bands) or 1% in power (with a commensurate reduction in setpoints) for each 1% violation of the FQ limit when evaluated including the W(Z) factor (condition f). This is the same relationship specified for Dominion's RPDC method (Reference 8). The at-power 15 minute dilution time prior to operator action (condition g) value is the same as that stated in KPS USAR Section 14.1.4 (CVCS malfunction event description), and the same as that currently used for KPS. A KPS-specific value for FQU (condition h) will be developed using CMS methodology benchmark results. It is expected that the value 1.0815 will be supported by those results.

3.2.4 Summary

The Dominion RPDC method is determined to be applicable to KPS and can be applied to KPS licensing analysis for nuclear core design and reload safety evaluation. Specific values assumed in the Dominion RPDC methodology, except for the maximum assumed cooldown temperature, are appropriate for KPS application or will be determined following CMS core design methods validation analyses (FQU). The License Amendment Request (LAR) to add DOM-NAF-5-A to Section 6.9.a.4 of the KPS Technical Specifications will include Technical Specification changes necessary for conformance with the RPDC methodology.

3.3 Applicability Assessment of Core Management System Methods – DOM-NAF-1, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations"

3.3.1 Description of Methodology

The Dominion reactor physics methods include the Studsvik Core Management System (CMS) core modeling code package. The primary computer codes in the CMS package are CASMO-4 and SIMULATE-3. The CASMO-4 computer code is the fuel assembly lattice code. CASMO-4 is a multi-group, two-dimensional transport theory code used for depletion and branch calculations for a single fuel assembly. The SIMULATE-3 code is a two-group, 3-dimensional nodal code based on the modified coarse mesh (nodal) diffusion theory calculation technique, coupled with thermal hydraulic and Doppler feedback. The general CMS calculation approach is to model the fuel assembly using the CASMO two-dimensional lattice physics code, and then to construct the three-dimensional SIMULATE reactor core model using lattice physics cross section data.

CMS reactor physics codes are used to model the core physics characteristics of the reload core including depletion/isotopic effects, reactivity, reactivity coefficients, power distribution, and shutdown margin. Dominion uses CMS reactor physics models in licensing applications, including calculations for core reload design, core operation, and key core parameters for reload safety analyses. CMS models are applied in the analyses for relaxed power distribution control, for startup physics testing (including control rod worth determination using the boron dilution and rod swap measurement techniques), and to provide physics constants for measurement of core power distributions.

CMS models are used to analyze the reactor core in all modes of operation including refueling shutdown, cold shutdown, 0% to 100% reactor power, and conditions associated with design basis transients. CMS models are applied over the entire fuel cycle from beginning to end of cycle.

3.3.2 Conditions and Limitations

a) The DOM-NAF-1 title and several statements in its SER refer to use of CMS for North Anna (NAPS) and Surry (SPS) Power Stations.
b) Benchmarking data was provided for 15x15 (SPS) and 17x17 (NAPS) fuel designs, while the KPS fuel design is 14x14.
c) In Section 5.0, "Conditions and Limitations," the SER lists two conditions that would require further validation and NRC approval:

   i. Use of mixed oxide fuel
   ii. Introduction of significantly different or new fuel designs

3.3.3 Assessment

Referring to the Conditions and Limitations listed in Section 3.3.2:

Condition a is not a technical limitation. Condition b is not a technical limitation provided the 14x14 design is not "significantly different" than the 15x15 and/or 17x17 designs. For condition c, part i, KPS does not use mixed oxide fuel. The balance of this assessment will focus on condition c, part ii.

The KPS fuel assembly lattice is a 14x14 fuel lattice and the current KPS fuel design is the Westinghouse 422 V+ fuel design. The KPS fuel lattice and fuel design are not significantly different from the SPS (15x15) and NAPS (17x17) fuel designs. Table 3.3.1 demonstrates the similarities of the 14x14 and 15x15 designs.

Table 3.3.1 FUEL ASSEMBLY AND COMPONENT DESIGN PARMETERS		
Component	**KPS**	**SPS**
Fuel Assembly		
Array	14 x 14	15 x 15
Pitch (Assy)	7.803 in.	8.466 in.
Pitch (Rod)	0.556 in.	0.563 in.
No. guide tubes	16	20
No. instrument tubes	1	1
No. spacer grids	7	7
Fuel Rods		
Fuel Pellet Diameter	0.3659 in.	0.3659 in.
Fuel Pellet Material	Sintered UO_2	Sintered UO_2
Fuel/Clad Diametric Gap	0.0075 in.	0.0075 in.
Fuel Cladding		
O.D.	0.422 in.	0.422 in.
I.D.	0.3734 in.	0.3734 in.
Material	ZIRLO	ZIRLO
Spacers (Top & Bottom/Mid)		
Material	Inconel/ZIRLO	Inconel/ZIRLO
Guide Tube		
O.D. (above dashpot)	0.526 in.	0.533 in.
I.D. (above dashpot)	0.492 in.	0.499 in.
Material	ZIRLO	ZIRLO

The most significant difference between the 14x14 and 15x15 designs for core physics calculations is the slight asymmetry of the 14x14 design (off-center instrument thimble). Because the SER restriction related to "significantly different" or "new fuel designs" is not clearly defined, any difference between the KPS fuel design and those specifically addressed in DOM-NAF-1 (Reference 9) could cause the 14x14 fuel to be categorized as a "new fuel design." Although the actual differences are minor, additional validation information will be provided to support application of DOM-NAF-1 methods to KPS.

Using the methods and processes delineated in DOM-NAF-1, the accuracy of the CMS models will be demonstrated through comparisons with reactor measurements and through comparisons with higher order Monte Carlo neutron transport calculations. This demonstration will be consistent with the assessment performed for the Dominion Surry and North Anna units as described in DOM-NAF-1. Where applicable, nuclear reliability factors (NRFs) will be determined for the key reactor physics parameters. The KPS NRFs are expected to be similar to those established for NAPS and SPS. The capability of the CMS models to support the KPS Startup Physics Test Program will also be demonstrated through the reactor measurement comparisons. It is expected that the CMS models are fully compatible with the KPS reactor test program.

As part of the development of the KPS model validation data, Dominion will compare CMS and Monte Carlo code calculations of reactivity worth for soluble boron, control rods, burnable absorber, moderator temperature defect, and fuel temperature. Dominion will compare SIMULATE predictions to reactor measured data and use statistical methods where applicable to determine CMS model uncertainties. Dominion will also use the CMS and Monte Carlo code calculations in combination with normalized reactor flux map reaction rate comparisons to determine appropriate power distribution peaking factor reliability factors.

Dominion will use data from multiple KPS operating cycles to benchmark the CMS models. These cycles cover core design changes including transitions in fuel enrichment, fuel density, spacer grid design and material, fuel vendor, core operating conditions (full-power average moderator temperature and rated thermal power), and burnable poison material and design. Dominion will use critical boron concentration and critical rod position, startup physics testing data, measured power distributions, and operational transient data in the CMS model benchmarking. The agreement between the measured and calculated values will be used to validate the application of CMS to KPS. Dominion will demonstrate that the CMS reactor physics models in conjunction with the indicated nuclear reliability factors, adequately represent the operating characteristics of KPS.

3.3.4 Summary

Results of the applicability assessment for the DOM-NAF-1 reactor physics methods to KPS will be provided in a supplement to DOM-NAF-5 to be included as Attachment A.

3.4 Applicability Assessment of RETRAN Methods – VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code"

3.4.1 Description of Methodology

3.4.1.1 Background

Dominion has developed the capability to perform system transient analyses. This capability, coupled with core thermal/hydraulic analysis capability, encompasses the non-LOCA licensing analyses required for the Condition I, II, III, and IV transients and accidents addressed in the Updated Safety Analysis Report (USAR). In addition, the capability for performing best-estimate analyses for plant operational support applications has also been developed.

The purposes of having transient and accident safety analysis capability are to: 1) maintain in-house cognizance and expertise in the system transient analysis area; 2) support plant operation; and 3) provide a basis for the reload core safety analysis and licensing process. The principal analysis tool is the RETRAN computer code (Reference 12), which determines the time-dependent (transient) thermal-hydraulic response of a Nuclear Steam Supply System (NSSS). The RETRAN computer code calculates: 1) general system parameters as a function of time; and 2) boundary conditions for input into more detailed calculations of Departure from Nucleate Boiling (DNB) or other thermal and fuel performance margins. The theory and numerical algorithms, the programming details, and the user's input information for the RETRAN computer code have been documented by its developers, Energy Incorporated (EI) and the Electric Power Research Institute (EPRI), in Volumes I through IV of Reference 12. Volume IV of Reference 12 provides the results of the extensive verification and qualification of the RETRAN code. The verification activity consisted of qualification of the code by comparison of code results with separate effects experiments, with systems effects tests, and with integrated system responses based on actual plant data or USAR results.

In conjunction with both an analysis tool and system models, the development of a non-LOCA licensing analysis capability requires conservative analysis assumptions and input data. For licensing calculations, the major Dominion analysis assumptions are consistent with those documented in the units' USAR. If a change in analysis assumptions is required by a plant modification, core reload, or a related change, the change will be assessed via the 10 CFR 50.59 process. Depending on the results of that assessment, either the analysis is submitted to the NRC for approval or a normal update of the appropriate section of the USAR is prepared.

3.4.1.2 Licensing Applications

Dominion's system transient analysis capability is intended for both best-estimate (e.g. training simulator validation) and licensing applications (e.g. core reload analysis). Since core reloads are the most common and expected reason for accident reanalysis, Dominion's system transient methodology is discussed in that context.

Transient analyses form an integral part of evaluations performed to verify the acceptability of a reload core design from the standpoints of safety, economics, and operational flexibility. The reload process consists of design initialization, design of the core loading pattern, and detailed characterization of the core loading pattern by the nuclear designer. The latter process determines the values of core physics related key analysis parameters. These key parameters are provided to the safety analyst who uses them in conjunction with current plant operating configurations and limits to evaluate the impact of the core reload on plant safety.

In performing this evaluation, it is necessary to ensure that those key parameters that influence accident response are maintained within the bounds or "limits" established by the parameter values used in the reference analysis (i.e. the currently applicable licensing calculation). The reference analysis (and the associated parameter limits) may be updated from time to time in support of a core reload or to evaluate the impact of some other plant parameter change.

For cases where a parameter is outside of these previously defined limits, an evaluation of the impact of the change on the results for the appropriate transients must be made. This evaluation may be based on known sensitivities to changes in the various parameters in cases where a parameter change is small or the influence on the accident results is weak. For cases where larger parameter variations occur, or for parameters that have a strong influence on accident results, explicit reanalysis of the affected transients is required and performed. Past analytical experience has allowed the correlation of the various accidents with those parameters that have a significant impact on them.

If a reanalysis is performed, the results are compared to the appropriate analysis acceptance criteria. The reload evaluation process is complete if the acceptance criteria are met, and internal documentation of the reload evaluation is provided for the appropriate Dominion safety review. If the analysis acceptance criteria are not met, more detailed analyses and/or Technical Specifications changes may be required to meet the acceptance criteria. Analysis changes are evaluated in accordance with the requirements of 10 CFR 50.59.

3.4.1.3 System Model Application

The production of a conservative, reliable safety analysis of a given anticipated or postulated transient is accomplished by combining a system transient model with appropriate transient-specific input. A system transient model is designed to provide an accurate representation of the reactor plant and those associated systems and components that significantly affect the course of the transient. Transient-specific input ensures that the dynamic response of the system to the postulated abnormality is predicted in a conservative manner, and includes: a) initial conditions; b) core reactivity parameters such as Doppler and moderator temperature coefficients, and control rod insertion and reactivity characteristics; and c) assumptions concerning overall systems performance. Important system performance assumptions include the availability of certain system components (such as pressurizer spray or relief valves) and control and protection system characteristics (setpoints, instrument errors, and delay times).

RETRAN affords the modeling flexibility to develop an infinite number of representations for a given nuclear plant. At Dominion, several standard plant models are assembled and maintained for performance of the entire spectrum of system transient analyses. RETRAN makes use of an input structure that allows modification of the base deck input for specific cases by use of override cards. Thus, specific transient cases may be executed without altering the base plant models.

The base models are designed to provide a basic system description comprised of those parameters that would not ordinarily change from cycle to cycle. Thus, such parameters as system volumes and flow areas, characteristics of various relief and safety valves, and primary coolant pump characteristics form part of the base models.

Dominion's RETRAN Topical Report VEP-FRD-41, (Reference 1) describes Dominion's history with the use and application of RETRAN, dating from the early days of code development. The report also provides a detailed description of the three-loop base models that have been developed for Dominion's Surry and North Anna plants.

3.4.1.4 Evolution of the Kewaunee Models

A KPS RETRAN-02 model was developed in the mid-1980's. Prior to development of the KPS RETRAN-02 input, the DYNODE-P code was used to address plant operation, licensing, and design change issues, and to support the reload core design and safety evaluation. The KPS RETRAN-02 model was developed as the RETRAN code was becoming an industry standard for transient safety analysis.

In 2000, the RETRAN-02 input model was converted to a RETRAN-3D input model so it could be used to support steam generators replacement (SGR) and the need to reanalyze the design basis transients for the SGR project. As part of the conversion to RETRAN-3D, a detailed steam generator model was incorporated to replace the RETRAN-02 model single-node steam generator model. The RETRAN-3D model was also upgraded to include the replacement steam generator geometry and associated plant changes.

To support changing from the DYNODE-P code to the RETRAN-3D code for safety analysis and plant support, the RETRAN-3D model was qualified by benchmarking the RETRAN-3D KPS model results to DYNODE results for several USAR Chapter 14 transients.

In the fall of 2005, the KPS RETRAN-3D model was used to address an issue regarding primary- and secondary-side thermal-hydraulic response to a feedwater line break in conjunction with a reactor coolant pump seal failure, and to determine whether the core would remain covered. As part of this project, the RETRAN-3D model was modified to reflect a power uprate from 1683 to 1772 MWt, including the reactor protection system (RPS) and the engineered safety features (ESF).

After the acquisition of KPS by Dominion, the KPS RETRAN-3D model was modified to conform to the Dominion RETRAN Topical Report for the RETRAN-02 models of the Surry and North Anna plants, and overlay decks were prepared to be consistent with the Surry and North Anna overlay decks.

The KPS RETRAN model changes and conversion from RETRAN-3D to RETRAN-02 were performed based on the following governing principles:

a) The KPS RETRAN model is consistent with the NRC-approved methods used in VEP-FRD-41 (Reference 1). There are a few minor differences:

   i. The KPS model explicitly models the safety injection accumulators.
   ii. The KPS model has separate volumes for the steam generator inlet and outlet plenums, where the Reference 1 models lump these volumes with the hot leg and pump suction leg respectively.
   iii. The KPS model treats the pressurizer spray line with explicit volumes and junctions. In the Reference 1 models, the spray is modeled as a pair of fill junctions with boundary conditions controlled by the control system model.
   iv. The KPS models have an explicit volume representation of the main feedwater piping from the point where the auxiliary feedwater piping ties in. In the Reference 1 models, the time to purge residual hot main feedwater from the feedwater lines following auxiliary feedwater initiation is modeled using control blocks.

b) The KPS RETRAN model does not violate any restrictions and limitations of use presented in the RETRAN-02 Safety Evaluation Report (SER) (Reference 13).

c) The model is as consistent as possible with the input parameters and methods used for the KPS USAR Chapter 14 transient analysis.

d) Wherever possible, the KPS RETRAN model is consistent with the other Dominion RETRAN models (similar nodalization and node numbers). Adopting a consistent noding and numbering scheme facilitates use of the models by analysts who are familiar with the Surry and North Anna plant models. As noted in (a) above, there are a few minor differences.

3.4.2 Conditions and Limitations

Appendix 7 of Reference 1 provides a detailed discussion of the conformance of Dominion's Surry and North Anna RETRAN models to the restrictions, limitations and conditions of use imposed by NRC staff in the generic RETRAN code Safety Evaluation Reports (SERs) issued for the RETRAN code Topical Report. Based on the principles cited above for development of the KPS model, the Reference 1 assessment is applicable to the KPS model. Table 3.4.1 lists the KPS-specific evaluations of the NRC RETRAN code restrictions and limitations for which further explanation was warranted for application of VEP-FRD-41 to KPS.

Table 3.4.1
KPS-Specific Evaluation of the USNRC Generic RETRAN Code Restrictions & Limitations

RETRAN02 Mod 002 Restrictions	VEP-FRD-41 Evaluation	Kewaunee Disposition
a) Conservative usage of 1-D kinetics must be demonstrated	Rod ejection performed with point kinetics per Dominion Topical Report VEP-NFE-2.	Dominion does not propose to apply VEP-NFE-2 methods to KPS at this time. Rod ejection analyses will continue to be done with approved Westinghouse methods.
e) Metal-water reaction will have to be justified for specific analyses	Model only used for rod ejection hot pin model. Justification/ benchmarking provided in Dominion Topical Report VEP-NFE-2.	Dominion will apply hot pin model with metal-water reaction to KPS analyses of rod withdrawal from subcritical and locked reactor coolant pump rotor events. Justification/benchmarking is provided in Dominion Topical Report VEP-NFE-2.
RETRAN02 MOD003/004 Restrictions		No differences between VEP-FRD-41 and KPS models have been identified.
RETRAN02 MOD005.0 Restrictions		No differences between VEP-FRD-41 and KPS models have been identified.

3.4.3 Assessment

RETRAN is approved for application to KPS with the Westinghouse 422 V+ fuel design and is used for the current KPS safety analyses of record (References 15, 16, 17 and 18). RETRAN was also approved for application to KPS prior to the recent implementation of the Westinghouse fuel design (Reference 19).

KPS and other Dominion units (Surry, North Anna and Millstone Unit 3) use Westinghouse designs for nuclear steam supply system (NSSS) and reactor protection system (RPS). SPS, NAPS and KPS have many design and operating similarities.

The design basis transients and accidents for KPS are similar to the design basis transients and accidents for the other Dominion Westinghouse units. The range of KPS transients and accidents is bounded by those transients and accidents addressed in Reference 1.

The reactor thermal hydraulic conditions of the design basis transients and accidents are similar between KPS and the other Dominion units, and are within the qualification and capability of the RETRAN code, as demonstrated in References 15, 16, 17, 18 and 19.

Validation of the Dominion KPS RETRAN model involves comparison of RETRAN calculations to the KPS analysis of record for selected transients using the following strategy:

a) Identify unique classes of events (RCS heatup, RCS cooldown/depressurization, reactivity excursion, loss of RCS flow, and loss of secondary heat sink).

b) Select transients that represent the range of transient responses generated by these events.

c) Perform demonstration analyses of selected events to validate the capability to model key phenomena.

d) Verify that applicability assessment criteria are met:

   i. Key phenomena are appropriately modeled and predicted
   ii. Predicted results are technically sound and are in reasonable agreement with the KPS USAR analyses of record (or differences are understood and assessed as acceptable)
   iii. General trends in key parameters are consistent with USAR analyses of record

3.4.4 Summary

Dominion's RETRAN methods (Reference 1) are determined to be applicable to KPS and can be applied to KPS licensing analysis for reload core design and safety analysis. The applicability of these methods is to be further demonstrated in a supplement to DOM-NAF-5 (to be included as Attachment B) by providing:

a) A base model noding diagram and region descriptions
b) Results of benchmarking comparisons to the analyses of record for selected transients

3.5 Applicability Assessment of Statistical DNBR Evaluation Methods – VEP-NE-2, "Statistical DNBR Evaluation Methodology"

3.5.1 Description of Methodology

Topical Report VEP-NE-2, "Statistical DNBR Evaluation Methodology," (Reference 3) describes Dominion's methodology for statistically treating several of the important uncertainties in Departure from Nucleate Boiling Ratio (DNBR) analysis. Previously, these uncertainties were treated in a conservative deterministic fashion, with each parameter assumed to be simultaneously and continuously at the worst point in its uncertainty range with respect to DNBR. The Statistical DNBR Evaluation Methodology is used to determine a plant-specific and fuel-specific statistical DNBR limit. This limit combines the correlation uncertainty with the DNBR sensitivities to uncertainties in key DNBR analysis input parameters. The statistical combination of some of these uncertainties permits a more realistic combination of the independent uncertainties and thus provides a more realistic evaluation of DNBR margin. Even though the statistical DNBR limit (the Statistical Design Limit or SDL) is larger than the deterministic DNBR limit (the Deterministic Design Limit or DDL), its use is advantageous. The Statistical DNBR Evaluation Methodology allows thermal hydraulic evaluations to be performed using nominal operating conditions as opposed to deterministic initial conditions (nominal conditions plus evaluated uncertainty).

In the performance of in-house DNB thermal-hydraulic evaluations, design limits and safety analysis limits are used to define the available retained DNBR margin for each application. The difference between the safety analysis (self-imposed) limit and the design limit is the available retained margin. For deterministic DNB analyses, the DDL is set equal to the applicable code/correlation limit (see Section 3.6). For statistical DNB analyses, the design DNBR limit is set equal to the applicable statistical design limit (SDL).

The Statistical DNBR Evaluation Methodology will be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWSC), and to the Loss of Flow analysis, the Locked Rotor Accident and the Single Rod Cluster Control Assembly Withdrawal at Power, SRWAP. The events modeled statistically (see Table 3.5.1) are limited by the statistical design limits (SDLs) evaluated in the implementation of the Statistical DNBR Evaluation Methodology for KPS, which will be submitted for NRC review and approval. In addition, there are events that will be evaluated with deterministic models. 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 $F_{\Delta H}^{E}$ (engineering uncertainties component), etc. The events modeled deterministically are limited by the deterministic design limits (DDLs) stated in DOM-NAF-2 (Reference 20).

Table 3.5.1 USAR Transients Analyzed with VIPRE-D/WRB-1 for KPS		
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

3.5.2 Conditions and Limitations

Topical Report VEP-NE-2 was reviewed and generically approved by the NRC in May 1987. The fuel-specific and plant-specific implementation of the VEP-NE-2 methodology must be submitted to the NRC for review and approval. Therefore, a plant-specific implementation of the Statistical DNBR Evaluation Methodology for Kewaunee Power Station will be submitted.

The NRC SER for VEP-NE-2 listed the following conditions that must be met by any plant-specific implementation of this generic methodology:

a) The selection and justification of the Nominal Statepoints used to perform the plant-specific implementation must be included in the submittal.

b) The justification of the distribution, mean and standard deviation for all the statistically treated parameters must be included in the submittal.

c) The justification of the value of model uncertainty must be included in the plant-specific submittal.

d) For the relevant critical heat flux (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 submittal, unless there is an approved Topical Report documenting them.

3.5.3 Assessment

A DNBR evaluation method involving the statistical treatment of uncertainties is currently approved for application to KPS with the Westinghouse 422 V+ fuel design using the Westinghouse WRB-1 correlation (References 15, 17 and 22). This DNB evaluation method (called the Revised Thermal Design Procedure, RTDP) is used for the current safety analyses of record for KPS. The RTDP is similar to Dominion's Statistical DNBR Evaluation Methodology.

The plant-specific, fuel-specific implementation of the Dominion Statistical DNBR Evaluation Methodology to KPS cores will be submitted for NRC review and approval. This submittal will provide the specific justification for Conditions (a), (b) and (c) cited in Section 3.5.2 above. Although no specific justification is provided herein for (a), (b) and (c), the application of the Westinghouse RTDP methodology clearly demonstrates that these conditions can be justified for the application of the Dominion Statistical DNBR Evaluation Methodology. The implementation of this methodology to KPS cores will result in a Statistical Design Limit (SDL) that is plant-specific and fuel-type specific. Since Appendix B to Topical Report DOM-NAF-2 has been approved by NRC (Qualification of the VIPRE-D/WRB-1 code/correlation pair), condition (d) has been met for the Westinghouse 422V+ fuel. Should Dominion elect to load in the KPS core a fuel product that uses a CHF correlation not previously qualified with the VIPRE-D computer code, a new submittal would be provided for the plant-specific and fuel-specific application of the Statistical DNBR Evaluation Methodology.

3.5.4 Summary

Statistical DNB evaluation methods are determined to be applicable to KPS and can be applied to KPS licensing analysis for reload core design and safety analysis. A plant-specific and fuel-specific application for Kewaunee Power Station cores containing Westinghouse 422 V+ fuel assemblies will be submitted to the NRC for review and approval.

3.6 Applicability Assessment of VIPRE-D Methods - DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code"

3.6.1 Description of Methodology

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 fuel rod surface temperature. DNB is a concern in reactor design because of the possibility of fuel rod failure resulting from the increased fuel 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. If the MDNBR is greater than the design limit, there is adequate thermal hydraulic 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.

VIPRE-D is the Dominion version of the computer code VIPRE (Versatile Internals and Components Program for Reactors), 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. VIPRE-D, which is based upon VIPRE-01, MOD-02.1, was adapted by Dominion for the specific analysis needs of the various Dominion nuclear power stations and their different fuel designs. 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 into Dominion's thermal hydraulic methodologies.

Topical Report DOM-NAF-2, "Reactor Thermal Hydraulics using the VIPRE-D Computer Code," (Reference 2) describes Dominion's use of the VIPRE-D computer code and the justification of all input, default parameters, and the specific modeling choices selected by Dominion. DOM-NAF-2 demonstrates that the VIPRE-D core thermal-hydraulics methodology is appropriate for pressurized water reactor (PWR) licensing applications. In addition, the various appendices to Topical Report DOM-NAF-2, document the qualification of several CHF correlations with the Dominion VIPRE-D computer code, as well as their associated code/correlation deterministic design limits. Topical Report DOM-NAF-2, including several appendices, received generic NRC approval (Reference 21) and, as such, the VIPRE-D core thermal hydraulics methodology can be used for any of Dominion's nuclear facilities.

3.6.2 Conditions and Limitations

The conditions and limitations associated with the implementation of Topical Report DOM-NAF-2 can be split into three groups. The first group of conditions and limitations is related to the general use of the VIPRE code, and its maintenance, and were imposed by the NRC in the SER for VIPRE-01 (References 24 and 25).

1.A. The application of VIPRE-D is limited to PWR licensing calculations modeling heat transfer regimes up to CHF. VIPRE-D will not be used for post-CHF calculations or for BWR calculations.

1.B. VIPRE-D analyses will only use DNB correlations that have been reviewed and approved by the NRC. The VIPRE-D DNBR calculations will be within the NRC-approved parameter ranges of the DNB correlations, including fuel assembly geometry and grid spacers. The correlation DNBR design limits

associated with these approved CHF correlations will be derived or verified using fluid conditions predicted by the VIPRE-D code. Each CHF correlation used will be qualified or verified in the appropriate appendixes to Topical Report DOM-NAF-2.

1.C. Any plant-specific, fuel-specific application of the DOM-NAF-2 methodology will strictly use the modeling choices approved in Topical Report DOM-NAF-2, which describes the intended uses of VIPRE-D for PWR licensing applications, and 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.

1.D. The Courant number, which is based on flow velocity, time step, and axial node size, will be set at greater than 1.0 in VIPRE-D transient calculations whenever a subcooled void model is used to ensure numerical stability and accuracy.

1.E. VIPRE-D is maintained within Dominion's 10CFR50 Appendix B Quality Assurance program.

A second set of conditions and limitations is related to Dominion planned uses and applications for VIPRE-D, and were imposed by the NRC in the SER for DOM-NAF-2 (Reference 21). According to this group of conditions and limitations, VIPRE-D can be used for:

2.A. Analysis of 14x14, 15x1 5 and 17x17 fuel in PWR reactors.

2.B. Analysis of DNBR for statistical and deterministic transients in the Updated Safety Analysis Report (USAR). Additional DNBR transients that are plant-specific may be analyzed in a plant specific application.

2.C. Steady state and transient DNB evaluations.

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

2.E. Providing the basis for reactor protection setpoints.

2.F. 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 the original DOM-NAF-2 Topical Report.

The third and final set of conditions and limitations is related to the plant-specific and fuel-specific application of the VIPRE-D methodology:

3.A. Changes to the Technical Specifications (TS) to add Topical Report DOM-NAF-2 and applicable approved appendixes to the plant Core Operating Limit Report.

3.B. A plant-specific and fuel-specific Statistical Design Limit(s) for the relevant code/correlation pairs, to be used in statistical evaluations, which is evaluated following the Statistical DNBR Evaluation Methodology (see Section 3.5).

3.C. Any TS changes related to Over-Temperature Delta-T (OTΔT), Over-Power Delta-T (OPΔT), or other reactor protection function, as well as revised reactor core safety limits.

3.D. Changes to the list of USAR transients for which the code/correlations and limits apply.

3.6.3 Assessment

KPS is a standard 2-loop PWR that uses 14x14 fuel. This is one of the approved applications of the DOM-NAF-2 methodology according to conditions 1.A and 2.A.

Conditions 1.B and 2.F are met by the use of the WRB-1/W-3 CHF correlations (which are NRC-approved), with the design limits and ranges of applicability listed in Reference 20. The WRB-1 correlation has been qualified with Dominion's VIPRE-D computer code in Appendix B to Topical report DOM-NAF-2 (Reference 20). A DNBR design limit of 1.17 was obtained for VIPRE-D/WRB-1 that yields a 95% non-DNB probability at a 95% confidence level. The range of validity for VIPRE-D/WRB-1 is also listed in

Reference 20. The Westinghouse W-3 correlation will be used when the local conditions fall outside the range of applicability of the WRB-1 correlation. Specifically, the W-3 correlation will be applied to the lower portion of the fuel assemblies (below the first mixing vane grid) and in low-pressure events, such as main steam line break (MSLB). The DNBR design limit for W-3 is 1.45 for pressures between 500 to 1000 psia and 1.3 for pressures above 1000 psia. This application was specifically approved in Reference 21.

The application of DOM-NAF-2 is approved for all NRC-approved PWR fuel types (Reference 21). Should Dominion elect to load in the KPS core a fuel product that uses a CHF correlation not previously qualified with VIPRE-D, a submittal would be made to the NRC in accordance with DOM-NAF-2 to qualify the new CHF correlation with the VIPRE-D computer code and provide the associated code/correlation deterministic design limit.

Dominion has developed VIPRE-D models for Kewaunee cores containing Westinghouse 422 V+ fuel. These models use all the modeling inputs approved in Topical Report DOM-NAF-2, including two-phase flow models and correlations, heat transfer correlations and turbulent mixing models, thus meeting conditions 1.C and 1.D. Should Dominion elect to load in the KPS core a different fuel product, Dominion would develop new VIPRE-D models for Kewaunee cores containing the new fuel product, and these models would strictly follow all the modeling guidelines specified in Topical report DOM-NAF-2, thus meeting conditions 1.C and 1.D.

These models are used to evaluate the DNB-related design basis transients and accidents (Table 3.5.1). The KPS transients and accidents are a subset of the ones listed in Table 2.1-1 of Topical report DOM-NAF-2. The reactor thermal hydraulic conditions of the design basis transients and accidents are similar between KPS and the other Dominion units and within the qualification and capability of the VIPRE-D code. Therefore, conditions 2.B, 2.C, 2.D and 2.E are met.

These models are also used to evaluate the plant-specific and fuel-specific Statistical Design Limit (SDL) within the context of the Statistical DNBR Evaluation Methodology, which will be submitted for NRC review and approval (see Section 3.5). These models will also be used to verify the Kewaunee setpoint functions, core thermal limit lines and USAR statepoint and transient analyses. Any changes to the Kewaunee reactor protection system setpoints and core thermal limit lines will be evaluated per the provisions of 10CFR50.59. Conditions 3.A, 3.B, 3.C, and 3.D are thus met for application of DOM-NAF-2 methods to KPS.

3.6.4 Summary

The DOM-NAF-2 core thermal hydraulic analysis methodology, including the applicable appendices, can be used for the thermal hydraulic evaluation of Kewaunee power station cores containing NRC-approved PWR fuel (currently Westinghouse 422 V+ fuel). The methods therein are determined to be applicable to KPS and can be applied to KPS licensing analysis for reload core design and safety analysis.

4.0 Conclusions and Implementation

4.1 Conclusions

Dominion nuclear core design and safety analysis methods were assessed for applicability to KPS. The Dominion reload nuclear design methods, as documented in the Dominion Topical Reports below, were determined to be applicable to KPS, and can be employed in the licensing design and evaluation of reload cores for KPS. The bases for this conclusion are provided in the Section 3.0 methodology applicability assessments.

- VEP-FRD-42, "Reload Nuclear Design Methodology" (Reference 6)
- VEP-NE-1, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications" (Reference 8)
- DOM-NAF-1, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations" (Reference 9)
- VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code" (Reference 1)
- VEP-NE-2, "Statistical DNBR Evaluation Methodology" (Reference 3)
- DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code" (Reference 2)

The applicability of the RETRAN and CMS methods to KPS will be further demonstrated through detailed validation analyses that will be documented in a supplement to DOM-NAF-5 (see Sections 4.2.3 and 4.2.4). In addition, the plant-specific and fuel-specific application analysis to define the DNBR Statistical Design Limit (SDL) will be completed to support the applicability of the Statistical DNBR methods to KPS (see Section 4.2.2).

KPS and other Dominion units (Surry, North Anna and Millstone Unit 3) use Westinghouse designs for nuclear steam supply system (NSSS) and reactor protection system (RPS). KPS has many design and operating similarities with the other Dominion Westinghouse units. KPS plant-specific considerations and features were evaluated and the differences from the methods as described in the Dominion Topical Report required for the application to KPS were identified in Section 3.0. The identified differences do not affect the conclusions on applicability of the methods to KPS.

Dominion analysis methods will be applied to KPS consistent with the conditions and limitations described in the Dominion Topical Reports and in applicable NRC Safety Evaluation Reports (SER). The conditions and limitations for each method were addressed in Section 3.0. The conditions and limitations will be met when the method is applied to KPS.

4.2 Steps for DOM-NAF-5 Implementation

The following steps are necessary to fulfill the applicability assessment for Dominion nuclear core design and safety analysis methods to KPS. Accomplishing these steps enables DOM-NAF-5 to be cited as the methodology reference in the KPS Technical Specification 6.9.a.4 and COLR.

4.2.1 Submit a license amendment request (LAR) to add DOM-NAF-5-A to Section 6.9.a.4 of the KPS Technical Specifications. Other conforming Technical Specification changes are to be incorporated into the LAR, as needed, to reflect use of Dominion methods.

4.2.2 Submit, as part of the step 4.2.1 LAR, a plant-specific and fuel-specific application analysis to define a DNBR Statistical Design Limit (SDL) per the provisions of VEP-NE-2 and DOM-NAF-2. The scope of the analysis is defined in Section 3.5 and 3.6 of this report.

4.2.3 Submit, as Attachment A of this report, detailed validation analyses for application of CMS methods (DOM-NAF-1). The scope of the analysis is defined in Section 3.3.

4.2.4 Submit, as Attachment B of this report, detailed validation analyses for application of RETRAN methods (VEP-FRD-41). The scope of the analysis is defined in Section 3.4.

5.0 References

1. Topical Report VEP-FRD-41, Rev. 0.1-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code," June 2004.
2. Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," September 2004.
3. Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.
4. Deleted.
5. Deleted.
6. Topical Report VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.
7. Letter from Scott Moore (NRC) to D. A. Christian (VEPCO), "Acceptance of Topical Report VEP-FRD-42, Revision 2, Reload Nuclear Design Methodology, North Anna and Surry Power Stations, Units 1 and 2," June 11, 2003.
8. Topical Report VEP-NE-1, Rev. 0.1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," August 2003.
9. Topical Report DOM-NAF-1, Rev. 0.0-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," June 2003.
10. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Proprietary), March 1978.
11. WCAP 10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.
12. EPRI NP-1850-CCM-A, "RETRAN-02, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Revision 6," December 1996.
13. Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.
14. Deleted.
15. Letter from J. G. Lamb (NRC) to T. Coutu (NMC) transmitting the NRC SER for Amendment No. 167 to the Operating License, revising TSs for use of Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features, April 4, 2003.
16. Letter from J.G. Lamb (NRC) to T. Coutu (NMC) transmitting the NRC SER for Amendment No. 168 to the Operating License, approving the Measurement Uncertainty Recapture (MUR) Power Uprate (1.4%), Letter No. K-03-094, July 8, 2003.
17. Letter from J.G. Lamb (NRC) to T. Coutu (NMC) transmitting the NRC SER for Amendment No. 172 to the Operating License, approving the 6% Stretch Power Uprate, Letter No. K-035, February 27, 2004.

18. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," (Proprietary), April 1999.

19. Letter from G. Lamb (NRC) to M.E. Reddemann (NMC), transmitting the NRC SER approving WPSRSEM-NP Revision 3, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Evaluation Methods Topical Report," Letter No. K-01-112, September 10, 2001.

20. DOM-NAF-2, Appendix B, "Qualification of the Westinghouse WRB-1 CHF Correlation in the Dominion VIPRE-D Computer Code," December 2004.

21. Letter from C.I. Grimes (NRC) to D. A. Christian (VEPCO) transmitting approval of Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," April 4, 2006 and letter from S. R. Monarque (NRC) to D. A. Christian (VEPCO) transmitting corrected pages for the NRC SER, June 23, 2006.

22. 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.

23. Letter from E. S. Grecheck, Virginia Electric and Power Company, to USNRC, "Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," December 2, 2002.

24. 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.

25. 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.

ATTACHMENT A

CMS Benchmarking Information (TO BE PROVIDED)

ATTACHMENT B

RETRAN Benchmarking Information (TO BE PROVIDED)