VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 17, 2003

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 03-578
NLOS/MM
Docket No.: 50-281
License No.: DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNIT 2
CYCLE 19 CORE OPERATING LIMITS REPORT

Pursuant to Surry Technical Specification 6.2.C, enclosed as Attachment 1 is a copy of the Virginia Electric and Power Company's (Dominion) Core Operating Limits Report (COLR) for Surry Unit 2 Cycle 19 Pattern AM, Revision 0.

Also provided as Attachment 2 for your information is VEP-FRD-42, Rev. 2.1-A "Reload Nuclear Design Methodology," dated August 2003. This Topical Report is Reference 1 of the COLR. Revision 2.1 modifies Revision 2.0 to accommodate use of the Studsvik Core Management System (CMS), which was approved by the NRC in a letter dated March 12, 2003 entitled "Virginia Electric and Power Company – Acceptance of Topical Report DOM-NAF-1, Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations (TAC Nos. MB5433, MB5434, and MB5437)." The modification of VEP-FRD-42, Rev. 2-A to form Rev. 2.1-A was made under the provisions of 10 CFR 50.59 and the analytical model and method approval process identified in VEP-FRD-42, Rev 2-A.

If you have any questions or require additional information, please contact Mr. Gary Miller at (804) 273-2771.

Very truly yours,

C. L. Funderburk, Director
Nuclear Licensing and Operations Support
Dominion Resources Services, Inc.
for Virginia Electric and Power Company

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Attachments:

1. Core Operating Limits Report – Surry 2 Cycle 19 Pattern AM – Rev. 0 – September, 2003.
2. Dominion Topical Report VEP-FRD-42 - Rev. 2.1-A – Reload Nuclear Design Methodology – August, 2003.

Commitment Summary: There are no new commitments as a result of this letter.

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Mr. G. J. McCoy
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Serial No.: 03-578
Docket No.: 50-281
Attachment 1

CORE OPERATING LIMITS REPORT
Surry 2 Cycle 19 Pattern AM
Revision 0

September 2003

Virginia Electric and Power Company (Dominion)

1.0 INTRODUCTION

This Core Operating Limits Report (COLR) for Surry Unit 2 Cycle 19 has been prepared in accordance with the requirements of Technical Specification 6.2.C.

The Technical Specifications affected by this report are:

TS 3.1.E and TS 5.3.A.6.b - Moderator Temperature Coefficient
TS 3.12.A.2 and TS 3.12.A.3 - Control Bank Insertion Limits
TS 3.12.B.1 and TS 3.12.B.2 - Power Distribution Limits

2.0 REFERENCES

1. VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003

   (Methodology for TS 3.1.E and TS 5.3.A.6.b - Moderator Temperature Coefficient; TS 3.12.A.2 and 3.12.A.3 - Control Bank Insertion Limit; TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor)

2a. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2b. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A Computer Code for the Best Estimate Analysis of Reflood Transients-Special Report: Thimble Modeling in W ECCS Evaluation Model," July 1986 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," June 1990 (Westinghouse Proprietary)

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor)

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," July 1990

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor)

3.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.2.C.

3.1 Moderator Temperature Coefficient (TS 3.1.E and TS 5.3.A.6.b)

3.1.1 The Moderator Temperature Coefficient (MTC) limits are:

+6.0 pcm/°F at less than 50 percent of RATED POWER, or

+6.0 pcm/°F at 50% of Rated Power and linearly decreasing to 0 pcm/°F at Rated Power

3.2 Control Bank Insertion Limits (TS 3.12.A.2)

3.2.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

3.3 <u>Heat Flux Hot Channel Factor-FQ(z)</u> (TS 3.12.B.1)

$$FQ(z) \leq \frac{CFQ}{P} K(z) \text{ for } P > 0.5$$

$$FQ(z) \leq \frac{CFQ}{0.5} K(z) \text{ for } P \leq 0.5$$

$$\text{where} : P = \frac{\text{Thermal Power}}{\text{Rated Power}}$$

3.3.1 $CFQ = 2.32$

3.3.2 $K(z)$ is provided in Figure 2.

3.4 <u>Nuclear Enthalpy Rise Hot Channel Factor-FΔH(N)</u> (TS 3.12.B.1)

$$F\Delta H(N) \leq CFDH \times \{1 + PFDH(1-P)\}$$

$$\text{where} : P = \frac{\text{Thermal Power}}{\text{Rated Power}}$$

3.4.1 $CFDH = 1.56$ for Surry Improved Fuel (SIF)

3.4.2 $PFDH = 0.3$

Figure 1

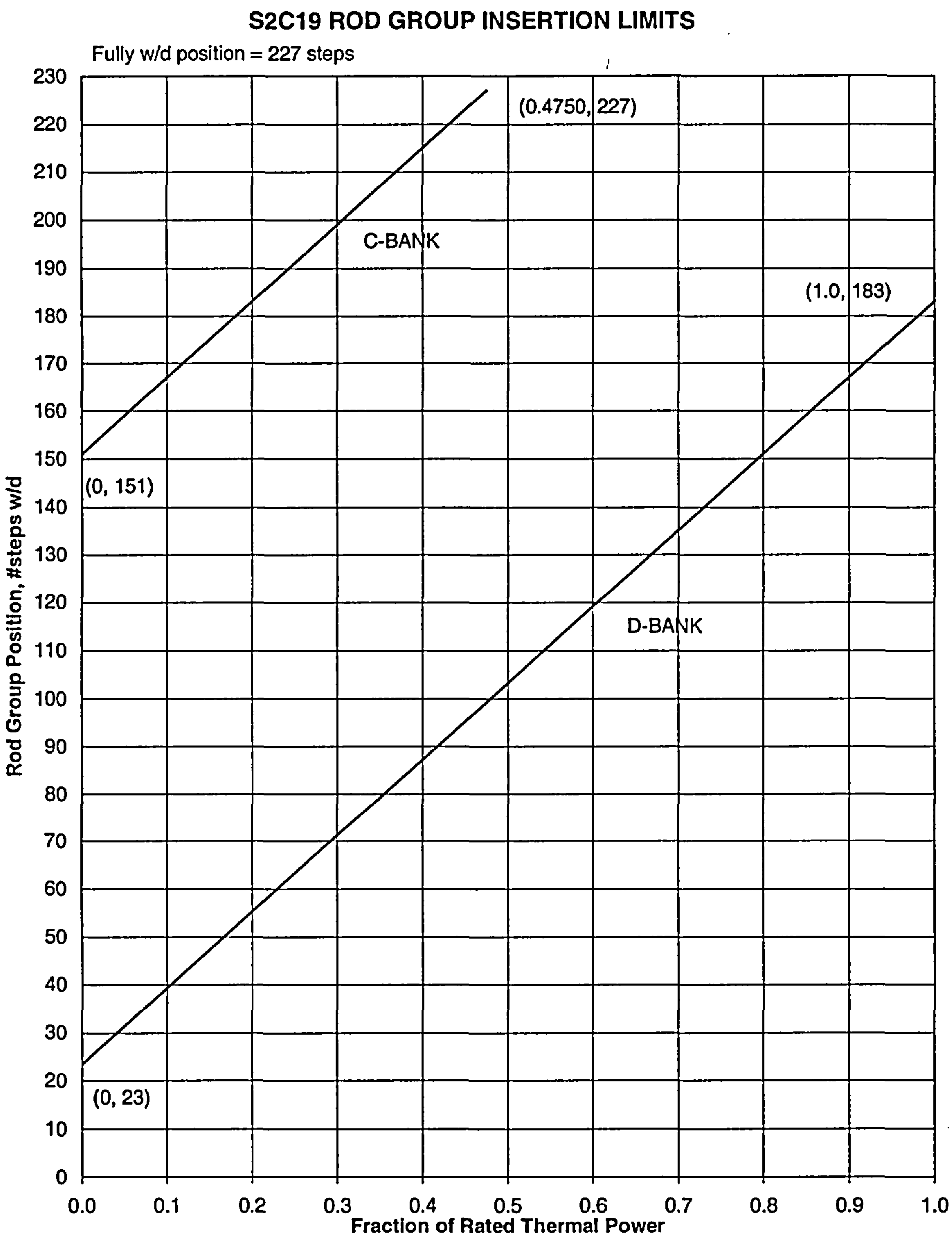
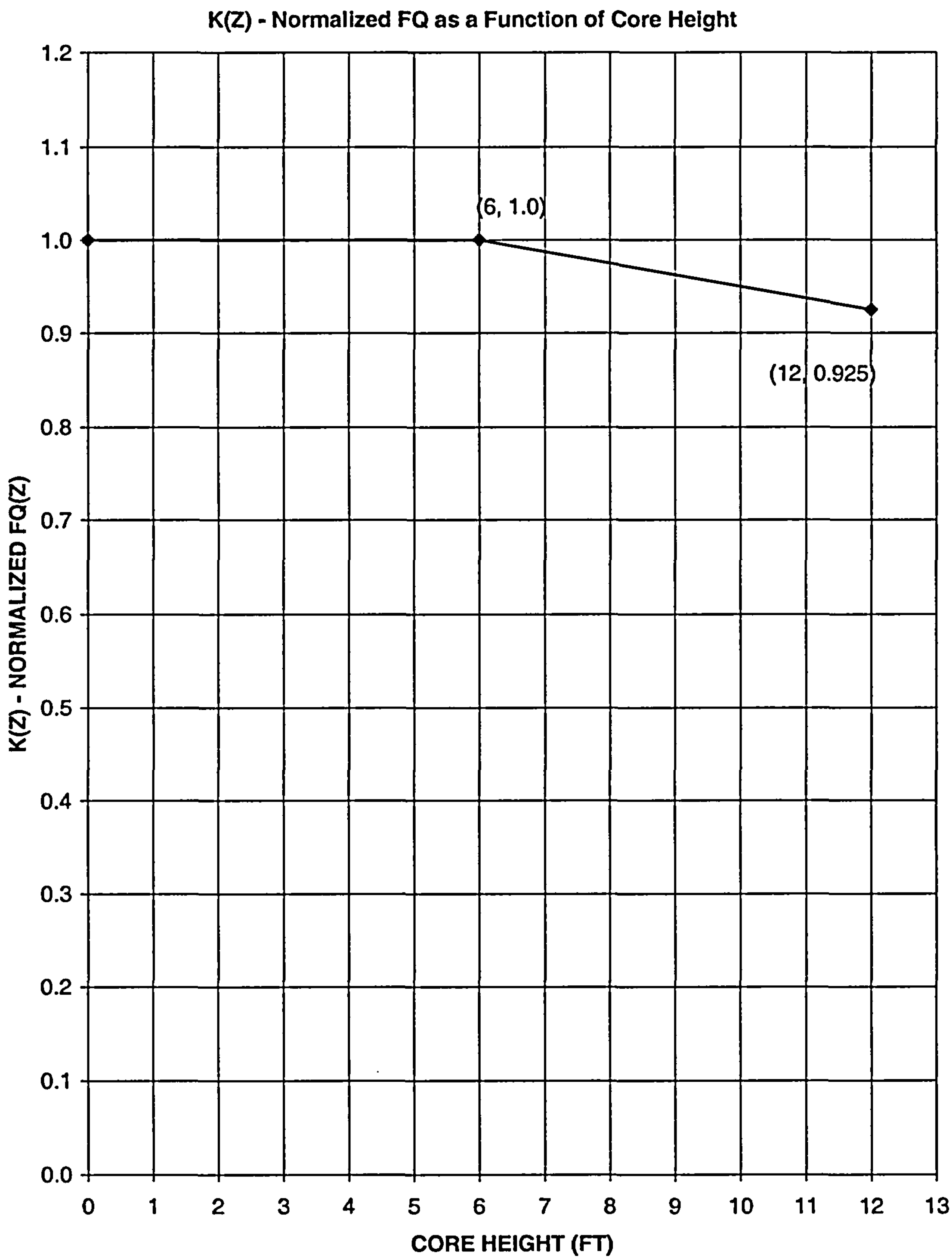

S2C19 ROD GROUP INSERTION LIMITS
Fully w/d position = 227 steps
(0.4750, 227)
C-BANK
(1.0, 183)
(0, 151)
D-BANK
(0, 23)
Rod Group Position, #steps w/d
230
220
210
200
190
180
170
160
150
140
130
120
110
100
90
80
70
60
50
40
30
20
10
0
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
Fraction of Rated Thermal Power

Figure 2


K(Z) - Normalized FQ as a Function of Core Height
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
K(Z) - NORMALIZED FQ(Z)
(6, 1.0)
(12, 0.925)
0 1 2 3 4 5 6 7 8 9 10 11 12 13
CORE HEIGHT (FT)

Serial No.: 03-578
Docket No.: 50-281
Attachment 2

Dominion Topical Report

VEP-FRD-42 - Rev. 2.1-A
Reload Nuclear Design Methodology

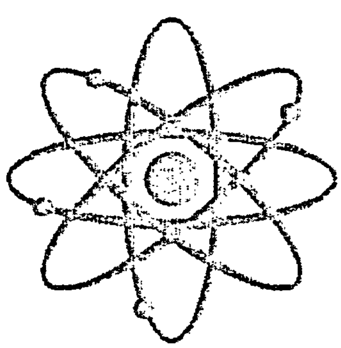
August, 2003

Virginia Electric and Power Company (Dominion)

VEP-FRD-42 Rev. 2.1-A

Dominion[SM]

Reload Nuclear Design Methodology



Nuclear Analysis and Fuel

August, 2003

VEP-FRD-42 Rev. 2.1-A

Reload Nuclear Design Methodology

NUCLEAR ANALYSIS AND FUEL
DOMINION
RICHMOND, VIRGINIA
August 2003

Recommended for Approval:

C. B. LaRoe, Supervisor
Nuclear Core Design

J. R. Harrell, Supervisor
Nuclear Safety Analysis

Approved:

K. L. Basehore, Director
Nuclear Analysis and Fuel

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

June 11, 2003 SERIAL # 03-381

RECD JUN 19 2003

NUCLEAR LICENSING

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
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5000 Dominion Boulevard
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SUBJECT: VIRGINIA ELECTRIC AND POWER COMPANY - ACCEPTANCE OF TOPICAL REPORT VEP-FRD-42, REVISION 2, "RELOAD NUCLEAR DESIGN METHODOLOGY," NORTH ANNA AND SURRY POWER STATIONS, UNITS 1 AND 2 (TAC NOS. MB3141, MB3142, MB3151, AND MB3152)

Dear Mr. Christian:

By letter dated October 8, 2001, as supplemented by letters dated May 13, and December 2, 2002, and March 21, 2003, Virginia Electric and Power Company (VEPCO) requested approval of Topical Report VEP-FRD-42, Revision 2, entitled "Reload Nuclear Design Methodology," for North Anna and Surry Power Stations, Units 1 and 2.

The Nuclear Regulatory Commission (NRC) staff has found that Topical Report VEP-FRD-42, Revision 2, is acceptable for referencing in licensing applications for the North Anna and Surry Power Stations, Units 1 and 2, to the extent specified and under the limitations delineated in the report and in the associated NRC Safety Evaluation (SE). The SE defines the basis for acceptance of the report.

Our acceptance applies only to matters approved in the subject report. We do not intend to repeat our review of the acceptable matters described in the report. When the report appears as a reference in licensing applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this topical report will be subject to a plant-specific review in accordance with applicable review standards.

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

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

Sincerely,

Scott Moore, Acting Director
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosure: Safety Evaluation

cc w/encl: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT VEP-FRD-42, REVISION 2

RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

NORTH ANNA AND SURRY POWER STATIONS, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated October 8, 2001 (Reference 1), as supplemented by letters dated May 13, (Reference 2) and December 2, 2002, (Reference 3) and March 21, 2003, (Reference 4) Virginia Electric and Power Company (VEPCO) requested approval of Topical Report VEP-FRD-42, Revision 2, entitled "Reload Nuclear Design Methodology Topical Report," for North Anna and Surry Power Stations, Units 1 and 2. Topical Report VEP-FRD-42 describes the core reload design methodology for performing a nuclear reload design analysis at North Anna and Surry Power Stations. This includes analytical models and methods, reload design and reload safety analysis, and an overview of analyzed accidents. The Nuclear Regulatory Commission (NRC) staff had previously limited the approval of Topical Report VEP-FRD-42, Revision 1-A, "Reload Nuclear Design Methodology," (Reference 5) to licensing applications involving Westinghouse-supplied fuel reloads. Revision 2 of this topical report extends the VEPCO methodology to Framatome ANP Advanced Mark-BW fuel.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34, "Contents of applications; technical information," requires that safety analysis reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload process, licensees perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, the licensees confirm that key inputs to the safety analyses are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

In an effort to limit cycle-specific Technical Specification (TS) changes, the NRC issued Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," (Reference 6) on October 3, 1988, to provide guidance for relocating cycle-specific parameter limits from the TS to a Core Operating Limits Report (COLR). Specifically, this GL allows a licensee to implement a COLR to include cycle-specific parameter limits that are established using NRC-approved methodology. The NRC staff-approved analytical methods used to

Enclosure

determine the COLR cycle-specific parameters are to be identified in the Administrative Controls section of the TS.

Topical Report VEP-FRD-42 is listed in the COLR Administrative Controls section of the North Anna and Surry TS and describes VEPCO's methodology for designing reload cores and performing reload safety analyses. Because the NRC staff previously approved Topical Report VEP-FRD-42, Revision 1-A, the NRC staff's review of Topical Report VEP-FRD-42, Revision 2, focused on the changes made to the approved version. Specifically, the NRC staff review focused on the extension of the methodology to Framatome ANP Advanced Mark-BW fuel types.

3.0 TECHNICAL EVALUATION

Topical Report VEP-FRD-42, Revision 2, describes the methodology applied in the design of reload cores at both the North Anna and Surry Power Stations. This topical report includes descriptions of analytical models and methods, reload nuclear design, reload safety analyses, and an overview of analyzed accidents and key parameter derivations. The NRC staff reviewed and approved Topical Report VEP-FRD-42, Revision 1-A, on July 29, 1986. VEPCO has submitted Revision 2 of this Topical Report to support the transition to Framatome ANP Advanced Mark-BW fuel at the North Anna and Surry Power Stations. In its Safety Evaluation (SE) for VEP-FRD-42, Revision 1-A, the NRC staff stated, "it is clear that the methodology presented is closely related to the Westinghouse methodology, and is applicable in its present form only to Westinghouse supplied reloads of Westinghouse nuclear plants." To support the transition to Framatome ANP Advanced Mark-BW fuel, VEPCO has revised VEP-FRD-42, Revision 1-A, to address this restriction and to present a revised discussion of the reload core design methodology. The Revision 2 changes address the following types of items:

- Applicability of methodology for analysis of incremental fuel design differences
- Generic methodology items impacted by transition to Framatome-ANP fuel
- Consolidation of prior VEPCO submittals regarding code and model updates
- Responses to original NRC staff review questions
- Miscellaneous editorial changes

By letter dated October 8, 2001, VEPCO proposed to apply the methodology described in Topical Report VEP-FRD-42, Revision 2, to both Framatome ANP Advanced Mark-BW and Westinghouse fuel types. In its submittal dated May 13, 2002, VEPCO stated that although the intended extension of this methodology is for the analysis of Framatome ANP Advanced Mark-BW fuel, the methodology is sufficiently robust for use on any fuel product with similar features. However, prior to the use of the Topical Report VEP-FRD-42, Revision 2, methodology for other fuel types, VEPCO must confirm that the impact of the fuel design and its specific features can be completely and accurately modeled with the VEPCO nuclear design and safety analysis codes and methods, that there is no significant effect upon calculated values of key reload safety parameters, and that the safety analysis codes and methods are applicable for analysis of the alternate fuel product. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42, Revision 2, these proposed changes would be submitted to the NRC staff for review and approval.

3.1 Analytical Models and Methods

The major analytical models described in Topical Report VEP-FRD-42, Revision 2, and currently used by VEPCO for reload design and safety analysis include:

- Virginia Power PDQ Two-Zone model
- Virginia Power NOMAD model
- VEPCO RETRAN model
- Core Thermal-Hydraulics models.

Topical Report VEP-FRD-42, Revision 1-A, listed the applicable computer codes, correlations, and methods used for thermal-hydraulic analyses of reload cores at the North Anna and Surry Power Stations. Topical Report VEP-FRD-42, Revision 2, no longer identifies the specific core thermal-hydraulic methods used; instead it states that the applicable codes and correlations for thermal-hydraulic analyses are listed in the COLR section of the North Anna and Surry TS, respectively. NRC GL 88-16 requires prior NRC staff review and approval of all methodologies used to calculate cycle-specific parameters that are in the COLR, and referenced in the COLR TS section. Thermal-hydraulic methodologies used in designing reload cores are typically fuel specific. The thermal-hydraulic methodologies VEPCO currently applies for the North Anna and Surry Power Stations, for example, the WRB-1 DNB correlation, and the VEPCO COBRA code and a statistical design methodology, are approved for use with the current Westinghouse fuel loaded in the North Anna and Surry cores. As such, in accordance with VEP-FRD-42, Revision 2, methodology, when transitioning to Framatome ANP Advanced Mark-BW fuel, VEPCO must submit a license amendment request to add the applicable and approved thermal-hydraulic methodology references to the COLR TS section. Since NRC GL 88-16 requires prior NRC staff review and approval of the thermal-hydraulic codes, correlations, and methods listed in the COLR section of the TS, the NRC staff finds that generic reference to the thermal-hydraulic methodology listed in the COLR TS section is acceptable.

The NRC staff reviewed and approved all codes used by VEPCO in the physics and thermal-hydraulics analyses of the reload core and described in Topical Report VEP-FRD-42, Revision 1-A. Topical Report VEP-FRD-42, Revision 2, describes the code changes and modifications that have been implemented by VEPCO since the NRC staff approved Topical Report VEP-FRD-42, Revision 1-A, on July 29, 1986. By letters dated October 1, 1990, August 10, 1993, and November 13, 1996, VEPCO formally requested NRC staff approval of these code modifications (References 7 - 9). VEPCO eventually implemented these changes under the provisions of 10 CFR 50.59. Because Topical Report VEP-FRD-42 is listed in the TS COLR section and requires NRC approval, the NRC staff informed VEPCO that the NRC staff must review and approve the analytical methods described within this topical report (Reference 10). Therefore, as part of this review, the NRC staff reviewed the PDQ Two-Zone, NOMAD and RETRAN code modifications described in Topical Report VEP-FRD-42, Revision 2, that were previously implemented under the provisions of 10 CFR 50.59.

PDQ Two-Zone Model

By letter dated October 1, 1990, VEPCO initially requested approval of the PDQ Two-Zone model in order to support the use of axially zoned flux suppression inserts in Surry, Units 1 and 2. The PDQ Two-Zone model is a three-dimensional, coarse mesh model that was developed to replace the PDQ Discrete model described in Topical Report VEP-FRD-42, Revision 1-A. The PDQ Two-Zone model is used to calculate three-dimensional power distributions, delayed neutron data, radial and axial peaking factors, assembly-wise burnup and isotopic concentrations, differential and integral rod worths, differential boron worth and boron endpoints, xenon and samarium worth, and core average reactivity coefficients such as temperature and power coefficients. In addition, PDQ is used to generate predicted power and flux distributions in order to translate thimble flux measurements into measured power distributions.

As part of the review of Topical Report VEP-FRD-42, Revision 2, the NRC staff reviewed the PDQ Two-Zone model as described in Topical Report VEP-NAF-1, "PDQ Two Zone Model," that VEPCO submitted on October 1, 1990. By letter dated December 2,2002, VEPCO verified that this topical report was the latest revision that has not received NRC staff approval and that this report contains an accurate representation of current codes and models with regard to methodology. That is, the theory, sources of input data, solution schemes, geometric mesh structure, energy group structure, and use of the models in the core modeling process have not changed since the October 1, 1990, submittal. Because VEPCO has been using the PDQ Two-Zone model in core designs for some time, the NRC staff review focused on model predictions relative to actual plant data.

VEPCO informed the NRC staff of its intent to implement the PDQ Two-Zone model under the provisions of 10 CFR 50.59 in a letter dated November 25, 1992 (Reference 11). Since that time, the PDQ Two-Zone model has been used in numerous core designs for both the North Anna and Surry Power Stations. The accuracy of the PDQ Two-Zone model has been verified each cycle during startup physics testing and during routine core follow. For each cycle, a Startup Physics Test Report and a Core Performance Report is issued to document the behavior of the core relative to the model predictions. By letter dated March 21, 2003, VEPCO provided additional information that demonstrated the accuracy of the PDQ model. This information includes measured and predicted data for key reactor physics parameters and confirmation that the nuclear reliability factors for these parameters are within the NRC-approved acceptance limits. Based on the accuracy demonstrated by these comparisons to actual plant data, the NRC staff finds the PDQ Two-Zone model to be acceptable for continued use in licensing calculations for the North Anna and Surry Power Stations. VEPCO's use of the PDQ Two-Zone model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in VEPCO's submittal dated March 21, 2003, and with Section 5.0 of this SE.

NOMAD

The VEPCO NOMAD model is a one-dimensional (axial), two energy group, diffusion theory computer code with thermal-hydraulic feedback. The NRC staff approved Topical Report VEP-NFE-1-A, "The VEPCO NOMAD Code and Model," for use of the NOMAD code and model on March 4, 1985. This version of the model is referenced in VEP-FRD-42, Revisions 1 and 2. VEPCO subsequently requested approval of an enhanced version of the NOMAD model on November 13,1996. The most significant enhancement to the NOMAD model is the use of multi-plane data from the three-dimensional (3-D) VEPCO PDQ Two-Zone model as the primary source of input. All model inputs to NOMAD come either directly or indirectly from the PDQ 3-D model calculations. Other enhancements to the model include improvements to the xenon model, the control rod model, the cross-section fit model, and the buckling model. The NOMAD model is used in the calculation of core average axial power distributions, axial offset, axial power peaking factors, differential control rod bank worth, integral control rod worth as a function of bank position, fission product poison worth, and reactivity defects.

As part of the review of Topical Report VEP-FRD-42, Revision 2, the NRC staff reviewed the NOMAD model as described in VEPCO's Topical Report VEP-NFE-1-A, Supplement 1, dated November 13, 1996. By letter dated December 2, 2002, VEPCO verified that this was the latest revision of the topical report that has not received NRC staff approval and that this report contains an accurate representation of current codes and models with regard to methodology. That is, the theory, sources of input data, solution schemes, geometric mesh structure, energy group structure, and use of the models in the core modeling process have not changed since the November 13, 1996, submittal. Because VEPCO has been using this enhanced NOMAD model in core designs for some time, the NRC staff review focused on model predictions relative to actual plant data.

VEPCO informed the NRC staff of its intent to implement the enhanced NOMAD model under the provisions of 10 CFR 50.59 in a letter dated November 13, 1996. Since that time, the NOMAD model has been used in numerous core designs for both the North Anna and Surry Power Stations. The accuracy of the NOMAD model has been verified each cycle during startup physics testing and during routine core follow. For each cycle, a Startup Physics Test Report and a Core Performance Report is issued to document the behavior of the core relative to the model predictions. VEPCO provided additional information on March 21, 2003, that demonstrates the accuracy of the NOMAD model. This information includes measured and predicted data for key reactor physics parameters and confirmation that the nuclear reliability factors for these parameters are within the NRC-approved acceptance limits. The NRC staff reviewed the measured data against the predicted data, and based on the accuracy demonstrated by these comparisons to actual plant data, the NRC staff finds the NOMAD model to be acceptable for continued use in licensing calculations for the North Anna and Surry Power Stations. VEPCO's use of the NOMAD model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in VEPCO's submittal dated March 21, 2003, and with Section 5.0 of this SE.

<u>RETRAN</u>

In the generic RETRAN SE dated September 4, 1984 (Reference 13), the NRC staff generically approved the use of RETRAN-01/MOD003 and RETRAN-02/MOD002 subject to the limitations and restrictions outlined in the SE and its enclosed Technical Evaluation Reports (TERs). The NRC staff reviewed VEPCO's RETRAN models and capabilities and approved the use of RETRAN-01/MOD003 for VEPCO in a letter dated April 11, 1985 (Reference 12). The NRC staff's SE stated that VEPCO had not provided information to address the restrictions stated in the NRC staff's SE for the generic RETRAN computer code and that VEPCO had not provided an input deck to the NRC staff as was required by the NRC staff's SE for the generic RETRAN code. The input deck submittal was required from VEPCO as a condition of the approval to use RETRAN. The NRC staff has verified VEPCO submission of the RETRAN input decks on August 21, 1985 (Reference 16), but could not verify that VEPCO submitted the RETRAN code limitations and restrictions.

In a letter dated August 10, 1993, VEPCO informed the NRC staff of various modifications and updates to its RETRAN model, and that these changes were to be implemented under the provisions of 10 CFR 50.59. This letter described several changes to the VEPCO RETRAN models, including expansion to a three-loop Reactor Coolant System and multi-node steam generator secondary side. Although this letter was submitted for the North Anna Power Station, VEPCO provided additional information on December 2, 2002, and March 21, 2003, justifying the applicability of the RETRAN model to both the Surry and North Anna Power Stations. By letter dated December 2, 2002, VEPCO provided additional information regarding its capability to make modifications to the RETRAN model. The NRC staff's SE dated April 11, 1985, for the VEPCO RETRAN model recognized that model maintenance activities would be performed under the utility's 10 CFR 50, Appendix B, Quality Assurance program, and stated, "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures." The NRC staff has determined that VEPCO has followed the requirements specified in the NRC staff's SE in updating the RETRAN models. Additionally, the NRC staff has also determined the qualification, documentation and implementation of the new models was performed in a manner that meets the programmatic elements of NRC GL 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," dated June 24, 1999 (Reference 17).

VEPCO is currently using RETRAN 02/MOD005.2. As such, the NRC staff requested additional information describing how each of the limitations, restrictions, and items identified as requiring additional user justification in the generic NRC staff's SEs, through the currently used version, are satisfied. This includes RETRAN02/MOD002 (Reference 13), RETRAN02/MOD003 and MOD004 (Reference 14) and RETRAN02/MOD005 (Reference 15). By letter dated March 21, 2003, VEPCO provided detailed information describing how each limitation (approximately 48 total) is treated in the North Anna and Surry RETRAN models. The NRC staff has reviewed VEPCO's responses and finds that the limitations, restrictions, and items identified as requiring additional user justification are satisfactorily addressed.

Based on the above discussions, the NRC staff finds that the VEPCO RETRAN models and the use of RETRAN continue to be acceptable for use in licensing calculations for the North Anna and Surry Power Stations.

Core Thermal-Hydraulics and Nuclear Design Models

In its submittal dated May 13, 2002, VEPCO provided information to demonstrate that the Framatome ANP Advanced Mark-BW fuel features affecting the safety analysis design inputs were within the modeling capability of the analytical models used as part of the reload design process and were identified in Topical Report VEP-FRD-42, Revision 2. From a core design perspective, the differences in modeling Framatome ANP Advanced Mark-BW fuel relative to Westinghouse fuel are small and are accommodated using model input parameters. These differences between the fuel types are similar in magnitude to incremental changes in Westinghouse fuel over time, which VEPCO has successfully modeled. Some of these minor changes include spacer grid differences, a slight increase in fuel density, a slight difference in the position of the fuel stack, and use of the advanced M5 alloy cladding. VEPCO has performed comparisons of measured and predicted Framatome ANP Advanced Mark-BW lead test assembly axial and integral power distributions over three cycles of operation in North Anna, Unit 1. The results of these comparisons provide direct confirmation of the accuracy with which VEPCO's reload analytical models can model Framatome ANP Advanced Mark-BW fuel. VEPCO has also performed several benchmark calculations to support use of these analytical models. In addition, in its submittal dated May 13, 2002, VEPCO also stated that the modeling changes associated with the Framatome ANP Advanced Mark-BW fuel are within the restrictions and limitations of the VEPCO core design and safety analysis codes. The NRC staff has reviewed this information provided by VEPCO and agrees that the Framatome ANP Advanced Mark-BW fuel features are within the modeling capability of the VEPCO core design analytical models. As such, the NRC staff finds that this modeling capability is applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

Analytical Methods

Topical Report VEP-FRD-42, Revision 2, Section 2.2, "Analytical Methods," provides a description of the various analytical methods used in the cycle design and evaluation. These methods are classified into three types of calculations: core depletions, core reactivity parameters and coefficients, and core reactivity control. Topical Report VEP-FRD-42, Revision 2, provides a very general description of the methods used to calculate these types of core physics parameters. These methods are consistent with those approved by the NRC staff in Topical Report VEP-FRD-42, Revision 1-A. VEPCO has incorporated some very minor changes. For example, the temperature increment and decrement range used in calculating reactivity coefficients can now be ±5°F or ±10°F about the nominal temperature, rather than only ±5°F as in Topical Report VEP-FRD-42, Revision 1-A. VEPCO added the range of ±10°F to minimize 3-D model convergence tolerance on the coefficients. The NRC staff does not consider these types of minor input changes as changes to the reload methodology. Additionally, the NRC staff agrees with VEPCO and finds that the analytical methods discussed in this section of Topical Report VEP-FRD-42, Revision 2, are not inherently dependent upon a specific fuel design or manufacturer. As such, the NRC staff finds that these methods are applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types because the analytical models used to implement these methods have been shown to be applicable for both Westinghouse and Framatome ANP Advanced Mark-BW fuel.

Analytical Model and Method Approval Process

Topical Report VEP-FRD-42, Revision 2, Section 2.3, "Analytical Model and Method Approval Process," is a new section in the topical report that describes acceptable means by which analytical models and methods can achieve approved status for use in the reload methodology. These acceptable means include: Implementation in accordance with the provisions of 10 CFR 50.59, independent review and approval by NRC, incorporation as a reference in the COLR section of the plant TS, and incorporation as a reference tool under VEPCO's GL 83-11, Supplement 1, Program. In its submittal dated May 13, 2002, VEPCO provided clarification regarding the types of changes that would be allowed under the provisions of 10 CFR 50.59, and the NRC staff has determined that VEPCO's interpretation is consistent with the intent of 10 CFR 50.59. Each of these means of achieving approved status either requires prior NRC approval or is a mechanism already acceptable to the NRC staff. Therefore, the NRC staff finds the addition of this new section to be acceptable. Additionally, these methods of achieving approved status are not fuel-specific and apply to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

3.2 Reload Design

The overall objective of core reload design is to determine fuel enrichment, feed batch size, and a core loading pattern that fulfills cycle energy requirements while satisfying the constraints of the plant design basis and safety analysis limits. Topical Report VEP-FRD-42, Revision 2, provides a general description of the reload design methodology used for the North Anna and Surry Power Stations, and is largely consistent with the NRC-approved methodology of Topical Report VEP-FRD-42, Revision 1-A. This VEPCO methodology divides the reload design process into three phases: 1) core loading pattern design and optimization, 2) determination of core physics related key analysis parameters for reload safety analysis, and 3) design report, operator curve, and core follow predictions.

In the reload safety analysis process, VEPCO uses a bounding analysis concept. This approach employs a list of key analysis parameters and limiting directions of the key analysis parameters for various transients and accidents. For a proposed core reload design, if all key analysis parameters are conservatively bounded, then the reference safety analysis is assumed to apply, and no further analysis is necessary. If one or more key analysis parameters is not bounded, then further analysis or evaluation of the transient or accident in question is performed. Topical Report VEP-FRD-42, Revision 2, Table 2 lists the key analysis parameters considered in reload design. To account for Framatome ANP Advanced Mark-BW fuel types, VEPCO determined that one additional key analysis parameter is required. This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. By letter dated May 13, 2002, VEPCO stated it calculates this key analysis parameter using the existing nuclear design codes PDQ Two-Zone and NOMAD.

The methods VEPCO used to determine the key parameters were consistent with the methods documented in Topical Report VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated F_Q Surveillance Technical Specifications," dated March 1986 (Reference 18), Topical Report WCAP-9272, "Westinghouse Reload Safety Evaluation," dated March 1978 (Reference 19), and Topical Report WCAP-8385, "Topical Report Power Distribution Control and Load Following Procedures," dated September 1974 (Reference 20). Topical Reports WCAP-9272 and WCAP-8385 are Westinghouse WCAP methodologies used for reload safety evaluations, and power distribution control and load following procedures. Topical Report VEP-NE-1-A documents VEPCO's NRC-approved Relaxed Power Distribution Control methodology. As part of the Topical Report VEP-FRD-42, Revision 2, review, the NRC staff questioned the applicability of these methodologies to Framatome ANP Advanced Mark-BW fuel types. By letter dated May 13, 2002, VEPCO provided additional information to the NRC staff, including the justification for the application of these methods for analyzing Framatome ANP Advanced Mark-BW fuel. Topical Reports VEP-NE-1-A and WCAP-8385 describe methodologies involving the simulation of a number of perturbed core states and power distributions using detailed nuclear core design codes and models. These analyses depend upon defining proper design inputs that characterize the reactor core. As discussed in Section 3.1, "Analytical Models and Methods," of this SE, VEPCO has demonstrated that the Framatome ANP Advanced Mark-BW fuel features are within the existing capability and range of applicability of the nuclear core design and safety analysis tools. Topical Report WCAP-9272 describes the Westinghouse reload methodology and forms the basis for VEPCO's reload methodology as described in Topical Report VEP-FRD-42, Revision 2. This Westinghouse methodology defines the specific key parameters for use in accident analyses and provides limiting directions for consideration in reload evaluations. VEPCO evaluated the use of an alternative fuel type and concluded that none of the physical design features invalidate the key parameter definitions or usage as cited in Topical Reports WCAP-9272 or VEP-FRD-42, Revision 1-A.

Topical Report VEP-FRD-42, Revision 2, incorporated Westinghouse's methodology for the analysis of the dropped rod event described in Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990 (Reference 21). This Westinghouse methodology requires that analyses be performed to determine: 1) statepoints (reactor power, temperature and pressure), 2) radial power peaking factors, and 3) DNB analysis at the conditions determined by items 1 and 2. This methodology incorporated data that is both plant-specific and cycle-specific. As part of the Topical Report VEP-FRD-42, Revision 2, review, the NRC staff questioned the applicability of this methodology to Framatome ANP Advanced Mark-BW fuel types. In its submittal dated May 13, 2002, VEPCO provided additional information to the NRC staff justifying the application of this methodology. VEPCO stated that the core physics characteristics of the Framatome ANP Advanced Mark-BW fuel are nearly identical to the Westinghouse fuel it will replace. There is no change in loading pattern strategy associated with the Framatome ANP Advanced Mark-BW fuel that would cause a change in the range of dropped rod worth or in the relationship between dropped rod worth and peaking factor increase. Reload cores, therefore, will not respond in a fundamentally different way to the dropped rod event due to the use of Framatome ANP Advanced Mark-BW fuel. Based on VEPCO's response and a review of the Westinghouse methodology, the NRC staff finds that this methodology would be applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

The NRC staff has reviewed the information provided by VEPCO and finds that the reload nuclear design methodology described in Topical Report VEP-FRD-42, Revision 2, is applicable to Framatome ANP Advanced Mark-BW fuel in addition to Westinghouse fuel types. This methodology incorporates several key elements, none of which is inherently dependent upon a specific fuel design or manufacturer. These key attributes of the methodology include:

- analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits,
- use of bounding key parameter values in reference safety analyses,
- recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data,
- representation of key fuel features via detailed inputs in core design and safety analysis models, and
- fuel is modeled 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.

4.0 CONCLUSIONS

The NRC staff has reviewed VEPCO's submittals and supporting documentation. Based on the considerations above, the NRC staff has concluded that the proposed Topical Report VEP-FRD-42, Revision 2, is acceptable for use in licensing applications at the North Anna and Surry Power Stations involving Westinghouse and Framatome ANP Advanced Mark-BW fuel types. Additionally, the NRC staff finds the continued use of PDQ Two-Zone, NOMAD, and RETRAN acceptable for licensing applications at the North Anna and Surry Power Stations involving Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) use of this topical report will not be inimical to the common defense and security nor to the health and safety of the public.

5.0 CONDITIONS AND LIMITATIONS

Prior to the use of the Topical Report VEP-FRD-42, Revision 2, methodology for fuel types other than Westinghouse and Framatome ANP Advanced Mark-BW fuel, VEPCO must 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 as discussed in its submittal dated May 13, 2002. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42, Revision 2, these proposed changes are required to be submitted for prior NRC review and approval.

In accordance with the Topical Report VEP-FRD-42, Revision 2, methodology, when transitioning to Framatome ANP Advanced Mark-BW fuel, VEPCO must submit a license amendment request to add the applicable and approved thermal-hydraulic methodology references to the COLR TS section. In addition, NRC GL 88-16 requires prior NRC staff review and approval of the thermal-hydraulic codes, correlations, and methods listed in the COLR section of the TS.

VEPCO's use of the PDQ Two-Zone model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in Attachment 2 of VEPCO's submittal dated March 21, 2003.

VEPCO's use of the NOMAD model for the North Anna and Surry core designs shall be in accordance with the restrictions and limitations listed in Attachment 3 of VEPCO's submittal dated March 21, 2003.

6.0 REFERENCES

1. Letter from L. N. Hartz, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Dominion's Reload Nuclear Design Methodology Topical Report," Docket Nos. 50-338/339 and 50-280/281, dated October 8, 2001.

2. Letter from L. N. Hartz, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," Docket Nos. 50-338/339 and 50-280/281, dated May 13, 2002.

3. Letter from E. S. Grecheck, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," Docket Nos. 50-338/339 and 50-280/281, dated December 2, 2002.

4. Letter from L. N. Hartz, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Request for Additional Information on Topical Report VEP-FRD-42, Reload Nuclear Design Methodology," Docket Nos. 50-338/339 and 50-280/281, dated March 21, 2003.

5. Letter from C. E. Rossi, USNRC, to W. L. Stewart, VEPCO, "Acceptance for Referencing of Licensing Topical Report VEP-FRD-42, Revision 1-A, Reload Nuclear Design Methodology," dated July 29, 1986.

6. USNRC GL 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 3, 1988.

7. Letter from W. L. Stewart, VEPCO, to USNRC, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Topical Report - PDQ Two Zone Model," Docket Nos. 50-280/281 and 50-338/339, dated October 1, 1990.

8. Letter from S. P. Sarver, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Supplemental Information for the NOMAD Code and Model, Reload Nuclear Design Methodology, and Relaxed Power Distribution Control Methodology Topical Reports," Docket Nos. 50-338/339 and 50-280/281, dated November 13, 1996.

9. Letter from M. L. Bowling, VEPCO, to USNRC, "North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Docket Nos. 50-338/339, dated August 10, 1993.

10. Letter from S. R. Monarque and G. E. Edison, USNRC, to D. A. Christian, VEPCO, "North Anna Power Station Units 1 and 2, and Surry Power Station Units 1 and 2 - Request for Additional Information on Virginia Electric and Power Company's Reload Nuclear Design Methodology Topical Report VEP-FRD-42 (TAC NOS. MB3141, MB3142, MB3151, and MB3152)," dated October 25, 2002.

11. Letter from W. L. Stewart, VEPCO, to USNRC, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Topical Report Use Pursuant to 10CFR50.59," Docket Nos. 50-280/281 and 50-338/339, dated November 25, 1992.

12. Letter from C. O. Martin, USNRC, to W. L. Stewart, VEPCO, "Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," dated April 11, 1985.

13. Letter from C. O. Thomas (USNRC) 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," dated September 4, 1984.

14. Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," dated October 19, 1988.

15. Letter from A. C. Thadani (USNRC) to W. J. Boatwright (RETRAN02 Maintenance Group), "Acceptance for Use of RETRAN02/MOD005.0," dated November 1, 1991.

16. Letter from W. L. Stewart, VEPCO, to H. R. Denton, USNRC, "Virginia Power, Surry and North Anna Power Stations, Reactor System Transient Analyses," Docket Nos. 50-280/281 and 50-338/339, dated August 21, 1985.

17. USNRC GL 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," dated June 24, 1999.

18. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated F_Q Surveillance Technical Specifications," dated March 1986.

19. WCAP-9272, "Westinghouse Reload Safety Evaluation," dated March 1978.

20. WCAP-8385, "Topical Report Power Distribution Control and Load Following Procedures," dated September 1974.

21. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

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Date: June 11, 2003

CLASSIFICATION/DISCLAIMER

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PREFACE

Revision 2 of this topical presents revised discussion of the Dominion reload core design methodology. The changes address several types of items that are listed here:

- Applicability of methodology for analysis of incremental fuel design differences
- Generic methodology items impacted by transition to Framatome-ANP fuel
- Consolidation of prior Dominion submittals regarding code and model updates
- Responses to original NRC Staff review questions
- Miscellaneous editorial changes

Although the intent of these changes is to qualify the methodology for use with Framatome-ANP fuel, the methodology is sufficiently robust that it can be applied to other fuel types with similar features.

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SECTION 1.0 - INTRODUCTION

The Dominion methodology for designing a reload core at its nuclear units is an iterative process. The process involves determining a fuel 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 not to be met for operation within the current operating requirements, one of two remedies is selected. Either the loading pattern is revised or changes are made in the operating requirements (Technical Specifications or Core Operating Limits Report (COLR), as applicable). Such changes ensure that plant operation will satisfy the applicable safety analysis criteria for the proposed loading pattern.

This report presents the methodology employed by Dominion for performing a nuclear reload design analysis at the North Anna and Surry Power Stations. It covers analytical models and methods, reload nuclear design, reload safety analysis, and an overview of analyzed accidents and key parameter derivations. This revision also incorporates generic reference to approved methodologies that are applicable to core thermal-hydraulic analyses. The COLR section of the plant Technical Specifications provides a listing of such applicable methodologies. The generic citation of these methodologies is intended to minimize duplicate NRC Staff review effort, since review and approval of any such methodologies would precede their listing in the COLR.

Detailed in this report are: (1) design bases, assumptions, design limits and constraints which are considered as part of the design process, (2) the determination and fulfillment of cycle energy requirements, (3) loading pattern determination, (4) the reload safety evaluation and (5) preparation of the cycle design report and related documents.

Dominion was formerly known as Virginia Power or (prior to January 15, 1985) as Virginia Electric and Power Company (VEPCO) and the topicals referenced were submitted using the former names in their titles. The current report introduces the Dominion designation but retains the prior nomenclature for citation of historical references.

SECTION 2.0 - ANALYTICAL MODELS AND METHODS

2.1 Analytical Models

The major analytical models currently used by Dominion for reload design and safety analysis are:

1. Virginia Power PDQ Two Zone and NOMAD models
2. Studsvik Core Management System (CMS)
3. VEPCO RETRAN model
4. Core Thermal-Hydraulics models

The PDQ Discrete model was originally approved for reference in licensing applications by the NRC (Reference 1). The PDQ Two Zone model was subsequently developed to replace the PDQ Discrete model. A Topical Report for the PDQ Two Zone model (Reference 2) was provided to the NRC prior to implementation under the provisions of 10CFR50.59 (Reference 3). The NOMAD model was originally approved for reference in licensing applications by the NRC (Reference 4) and has subsequently undergone significant enhancements. The updated model was implemented under the provisions of 10CFR50.59 (Reference 5).

As indicated in the SER for this Topical Report, the PDQ Two Zone model and the NOMAD model have been reviewed and approved by the NRC for continued use in licensing calculations for North Anna and Surry core designs within the limitations listed in Appendix C.

These models have been used to model the entire range of cores at the Surry and North Anna power stations, including evolutionary changes in fuel enrichment, fuel density, loading pattern strategy, spacer grid design and material, fuel clad alloy, and burnable poison material and design. Some of these changes were implemented as part of various Lead Test Assembly programs, and have included fuel assemblies from both Westinghouse and Framatome-ANP. The predictive accuracy of the models throughout these changes demonstrates that incremental design variations in fuel similar to the Westinghouse design are well within the applicable range of the core design models. Each model has sufficient flexibility such that minor fuel assembly design differences similar to those noted can be adequately accounted for using model design input variables.

Dominion has updated its capability to perform nuclear utility reactor analyses in support of the Surry and North Anna nuclear power stations by obtaining NRC approval to use the Studsvik Core Management System (CMS) core modeling code package (Ref. 16 – CMS Topical "A" version). The CMS package consists primarily of the CASMO-4 and SIMULATE-3 codes. CMS core models were extensively validated by comparing calculated data to higher order calculations and to measurements from Surry and North Anna Units 1 and 2.

Use of the RETRAN Code and models, as originally approved for reference in licensing applications is documented in Reference 6. Supplemental details concerning the models used with the RETRAN Code were provided to the USNRC in an informational letter, Reference 7. These models were implemented under the provisions of 10CFR50.59. The applicable thermal-hydraulic codes and models are listed in the COLR section of the plant Technical Specifications. These analysis models have been used successfully to model plant transient response for core reloads, as well as various changes to plant configuration including core uprate and steam generator replacement.

2.1.1 Virginia Power PDQ Two Zone and NOMAD Models

The Virginia Power PDQ Two Zone Model performs three-dimensional geometry diffusion-depletion calculations for two neutron energy groups. The model uses the CELL2 code (Reference 8) and several auxiliary codes to generate and format the cross section input, perform shuffles, and other operations. The model employs a non-uniform mesh structure (25 X-Y mesh and 26 axial mesh) to represent each fuel assembly. Quarter core symmetric or full core geometry may be specified. The effects of non-uniform moderator density and fuel temperatures are accounted for with thermal-hydraulic feedback. More complete descriptions of the model and the auxiliary codes may be found in Reference 2.

The PDQ Two Zone model is used to calculate three-dimensional power distributions (including steamline break statepoints), delayed neutron data, radial and axial peaking factors, assembly-wise burnup and isotopic concentrations, differential and integral rod worths, differential boron worth and boron endpoints, xenon and samarium worth and core average reactivity coefficients such as temperature and power coefficients. In addition, PDQ is used to generate predicted power and flux distributions in order to translate thimble flux measurements into measured power distributions.

The Virginia Power NOMAD Model performs one-dimensional axial diffusion-depletion calculations (with thermal-hydraulic feedback) for two neutron energy groups. The NOMAD Model makes use of data from the PDQ Two Zone model for two group cross sections and for normalization. The NOMAD model and its auxiliary codes are described in detail in Reference 5. The NOMAD model is used in the calculation of core average axial power distributions, axial offset, axial peaking factors, differential control rod bank worth, integral control rod worth as a function of bank position, fission product poison worth, and reactivity defects. In addition, NOMAD has the capability to perform criticality searches on boron concentration, control rod position, core power level, and inlet enthalpy. Simulation of load follow maneuvers, performance of Final Acceptance Criteria (FAC) analysis, and Relaxed Power Distribution Control (RPDC, Reference 9) may also be performed with the NOMAD model.

2.1.2 Studsvik Core Management System (CMS)

The principal computer codes in CMS are CASMO-4 and SIMULATE-3. The CMS-LINK code provides the coupling between these two codes. More complete descriptions of the model and the auxiliary codes may be found in Reference 16.

CASMO-4 (CASMO) is a multigroup two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies or simple pin cells. The code handles geometry types consisting of cylindrical fuel rods of varying composition in a square pitch array. Fuel rods may be loaded with integral poisons such as gadolinium or boron. The fuel assembly model may contain burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, boron steel curtains, and cruciform control rods in the regions separating fuel assemblies.

In order to generate a neutronic data library for SIMULATE-3 a series of CASMO depletions and branch cases is required. This series of calculations is defined within CMS as the "SIMULATE-3 Case Matrix." This case matrix consists of a series of depletions and instantaneous branch cases vs. exposure as a function of varied boron concentration, moderator temperature, fuel temperature, and shutdown cooling time, as well as cases with control rods and without removable burnable poison in guide tube locations. CMS-LINK is a linking code that processes CASMO card image files into a binary formatted nuclear data library for use by SIMULATE-3.

SIMULATE-3 (SIMULATE) is an advanced two-group nodal code for the analysis of both PWRs and BWRs. The code employs fourth-order polynomial representations of the intranodal flux distributions in both the fast and thermal groups. Key features of SIMULATE are:

- Pin power reconstruction
- No normalization required against higher order calculations
- Explicit representation of the reflector region
- Coupled neutronics/thermal-hydraulics
- Internal calculation of the effect of spacer grids on axial power distributions
- Calculation of intra-nodal axial power distribution effect on FQ

SIMULATE is capable of performing all of the calculations performed by PDQ and NOMAD. Due to the greatly reduced run times for SIMULATE, 1D/3D synthesis techniques are not required when using SIMULATE. All SIMULATE analyses are performed directly using 3D geometry.

2.1.3 VEPCO RETRAN Models

The VEPCO RETRAN models (Reference 6 and 7) are used to perform reactor coolant system (RCS) transient analyses. As part of the reload methodology, these models are used to confirm that reload cores continue to meet the safety analysis criteria for those instances when a key analysis parameter is not bounded for the reload. Such reanalysis begins with the plant base model with the transient specific input modifications necessary to reflect the reload core characteristics in the revised licensing analysis.

The VEPCO RETRAN Models include appropriate representations of core power (via point kinetics), forced and natural circulation fluid flow and heat transfer. Plant specific models of components such as pumps, relief and safety valves, protection and control systems are also included.

2.1.4 Core Thermal-Hydraulics Models

The applicable code(s) and correlation(s) for thermal-hydraulic analyses are listed in the COLR section of the plant Technical Specifications. The code(s) solve the governing conservation and state equations to resolve the flow and energy fields within the reactor core geometry. These results are used in turn to calculate the departure from nucleate boiling ratio (DNBR) with the appropriate CHF correlation. Such models are used to perform either steady state DNBR calculations or transient DNBR analyses with forcing functions which have been supplied by the RETRAN code. Steady state applications include thermal limit generation, DNBR statepoint analyses and reload axial shape verification. Examples of transient applications are loss of flow and locked rotor DNBR analysis.

The COLR section of the plant Technical Specifications lists the applicable methodology for statistically treating several of the important uncertainties in DNBR analysis. Previously, these uncertainties were treated in a conservative deterministic fashion, with each parameter assumed to be simultaneously and continuously at a bounding value within its uncertainty range with respect to effect upon the calculated DNBR. The statistical methodology uses a statistical combination of these uncertainties, permitting a more realistic evaluation of DNBR margin.

2.2 Analytical Methods

This section presents a description of the various analytical methods used in the cycle design and evaluation. These methods may be classified into three types of calculations: core depletions; core reactivity parameters and coefficients; and core reactivity control.

2.2.1 Core Depletions

During the preliminary fuel loading and loading pattern search, depletions of the reload core are performed based on the low and high estimates of the end-of-cycle (EOC) burnup (the burnup window) for the previous cycle. The reload core loading pattern is depleted at hot full power (HFP), all rods out (ARO) conditions, typically in quarter-core geometry. During the depletion, criticality is maintained by varying the boron concentration. These calculations provide relative power distributions, burnup predictions and an estimate of the cycle's full power capability.

For the reload safety evaluation of a loading pattern, burnup window depletions allow the sensitivities of the predicted reload cycle parameters to be examined as a function of the previous EOC burnup. The calculation of reload design parameters required for startup physics testing and core follow are made as near to the actual operating conditions of the reload as possible.

2.2.2 Core Reactivity Parameters and Coefficients

The core reactivity parameters and coefficients describe the kinetic characteristics of the core. These parameters and coefficients quantify the changes in core reactivity due to varying plant conditions such as changes in the moderator temperature, fuel temperature, or core power level. The reactivity coefficients and parameters are calculated on a core-wide basis for a representative range of core conditions at the beginning, middle and end of the reload cycle. These include zero power, part power, and full power operation; at various rodded core configurations; and for equilibrium xenon or no xenon conditions. These parameters are used as input to the safety analysis for modeling the reactor's response during accidents and transients. In addition, they may be used to calculate reactivity defects (integral of the coefficient over a specific range of temperature or power) to determine the reactor's response to a change in temperature or power. A description of each type of calculation follows.

2.2.2.1 Reactivity Coefficients and Defects

The Doppler temperature coefficient (DTC) is defined as the change in reactivity per degree change in the fuel temperature. The moderator temperature coefficient (MTC) is defined as the change in reactivity per degree change in the moderator temperature. The isothermal temperature coefficient (ITC) is defined as the change in reactivity per degree change in the moderator and fuel temperatures with the moderator and fuel temperatures changing uniformly. Isothermal temperature coefficients are of particular interest at hot zero power (HZP) when the whole core is at approximately a single temperature, allowing reactivity changes due to temperature variation to be readily measured and compared to predicted values. Temperature coefficients are typically calculated using two cases at ±5°F or ±10°F about the nominal temperature, with all other core parameters held constant. The Doppler temperature change can result from a change in core power or from a change in moderator temperature.

The total power coefficient (TPC) is defined in terms of core reactivity per percent change in core power due to the combined effect of the moderator and fuel temperature changes associated with core power level changes. The Doppler power coefficient (DPC) is the portion of the TPC that is related to the change in fuel temperature. Power coefficients typically include the effect of flux redistribution caused by the core power change and are typically calculated using two cases at ±5% power or ±10% power about the nominal power.

Temperature and power defects are the integrals of the coefficients over a desired range and are calculated using two cases at the upper and lower endpoint of the desired range. The method of calculating temperature and power coefficients depends on whether the parameter is desired at HZP (or no thermal-hydraulic feedback) conditions or at-power conditions. At-power calculations typically include the effects of thermal-hydraulic feedback.

2.2.2.2 Differential Boron Worth

The differential boron worth is defined as the change in reactivity due to a unit change in boron concentration. Differential boron worths are calculated by noting the change in core average reactivity due to a change in the core-wide boron concentration, (typically ± 20 ppm about the target value), with all other core parameters being held constant.

2.2.2.3 Delayed Neutron Data

Delayed neutron data are used in evaluating the dynamic response of the core. The delayed neutrons are emitted from precursor fission products a short time after the fission event. The delayed neutron fraction and decay constant for six delayed neutron groups at various core conditions are found by weighting the delayed neutron fraction for each fissionable isotope in each group by the core integrated fission rate of that isotope.

The SIMULATE model also includes an importance weighting using the cell average adjoint flux in each energy group.

2.2.2.4 Fission Product Poison Worths

The buildup and decay of certain fission products (such as Xe^{135} and Sm^{149}) and actinides (such as Np^{239}, Pu^{239}, Pu^{241}, and Am^{241}) result in reactivity changes that are important during core conditions including plant startups, power ramp maneuvers, reactor trips, and extended outages. The effect of Xe^{135} is most important for maneuvers occurring over a few hours or days. The most important time scale for changes in the other significant nuclides is days or months, and the reactivity effect is typically calculated as a combined net effect.

2.2.3 Core Reactivity Control

The full length control rods control relatively rapid reactivity variations in the core. The control rods are divided into four control banks (designated D, C, B, and A) and two shutdown banks (designated SB and SA). The control banks D, C, B, and A are used to compensate for core reactivity changes associated with changes in operating conditions such as temperature and power level and are moved in a fixed sequential pattern to control the reactor over the power range of operation. The shutdown banks are used to provide shutdown reactivity.

Changes in reactivity which occur over relatively long periods of time are compensated for by changing the soluble boron concentration in the coolant. Significant parameters governing core reactivity control characteristics are calculated as follows.

2.2.3.1 Integral and Differential Rod Worths

Integral rod worths are calculated by determining the change in reactivity due to the control rod being out of the core versus being inserted into the core with all other conditions being held constant. Differential and integral rod worths are calculated as a function of axial position. The change in core average reactivity is evaluated as a function of the axial position of the rod or rods in the core to obtain the differential rod worth.

2.2.3.2 Soluble Boron Concentrations

Boron in the form of boric acid is used as the soluble absorber in the reactor coolant. At HFP, soluble boron is used to compensate for the reactivity changes caused by variations in the concentration of xenon, samarium and other fission product poisons, the depletion of uranium and the buildup of plutonium, and the depletion of burnable poisons. Predictions of the soluble boron concentration necessary to maintain criticality or subcriticality are performed.

2.3 Analytical Model and Method Approval Processes

The Dominion reload evaluation methodology defines an approach for the design of reload cores and the evaluation of key characteristics of reload cores that have an impact upon plant safety. It is a general methodology consisting of the tools and a process that has been demonstrated to adequately consider the relevant factors and assess their impact. The methodology is robust enough to allow incorporation of alternate analytical models and methods, subject to the provision that such models and methods are demonstrated to be acceptable.

Demonstration of acceptability for potential alternative tools is a necessary precondition for their use in the Dominion reload methodology. However, such demonstration is separate from the reload methodology itself. There are several acceptable means by which either analytical models or methods can achieve approved status for use in the reload methodology. These are listed below.

- implemented in accordance with the provisions of 10CFR50.59
- independent review and approval by NRC
- incorporated as a reference in the COLR section of the plant Technical Specifications
- incorporated as a reference tool under Dominion Generic Letter 83-11, Supplement 1 program

SECTION 3.0 - RELOAD DESIGN

3.1 Introduction

The overall objective in the design of a reload core is to determine the enrichment and number of new fuel assemblies and a core loading pattern which will fulfill the energy requirements for the cycle while satisfying the design basis and all applicable safety analysis limits. The nuclear design effort to accomplish these objectives can be divided into three phases. These phases, in the chronological order of performance, are:

I. Core loading pattern design and optimization.
II. Determination of core physics related key analysis parameters for reload safety analysis.
III. Design report, operator curve, and core follow predictions.

These phases hereafter will be referred to as design Phases I, II and III, respectively.

The objective of Phase I design is to produce a core loading pattern which meets the constraints outlined in the design initialization (see Section 3.2.1). These constraints are general items such as energy requirements, plant operational changes and physical changes planned during the cycle. In addition, some preliminary calculations are performed to verify that parameters considered integral for an acceptable core loading pattern are met.

The objective of Phase II of the design process is to verify that all core physics related limits are met for the core loading pattern. Once the final loading pattern for the reload cycle has been optimized under Phase I, the core physics related key analysis parameters for the reload cycle are verified to determine if they are bounded by the limiting values for these parameters assumed in the reference safety analyses. These Phase II parameters are calculated using conservative assumptions to ensure the results adequately bound the reload. If a key analysis parameter for the reload cycle exceeds the limiting value, the corresponding transient is evaluated or reanalyzed using the reload value. Should the reload value for a key parameter cause a safety criterion not to be met, the reload design may be altered or new operating limits may be specified in the COLR or Technical Specifications.

Physics design predictions for the support of station operations are calculated in Phase III using analysis techniques consistent with those of Phase II, except their calculation is performed on a best-estimate basis. These predictions are compared with measurements during startup physics testing and core follow to verify the design calculations, insure that the core is properly loaded, and verify that the core is operating properly.

3.2 Core Loading Pattern Design and Optimization

3.2.1 Design Initialization

Before any nuclear design calculations are performed for a reload core, a design initialization is performed. The design initialization marks the formal beginning of the design and safety evaluation effort for a reload core and identifies the objectives, requirements, schedules, and constraints for the cycle being designed. It includes the collection and review of design basis information to be used in initiating design work. This review is to insure that the designer is aware of all information which is pertinent to the design and that the subsequent safety evaluation will be based on the actual fuel and core components that are available, the actual plant operating history, and any plant system changes projected for the next cycle.

The design basis information to be reviewed includes:

1. Unit operational requirements.
2. Applicable core design parameter data.
3. Safety criteria and related constraints on fuel and core components as specified in the Final Safety Analysis Report (FSAR) as updated (UFSAR).
4. Specific operating limitations on the plant as contained in the Technical Specifications and COLR.

5. Plant or Technical Specification changes implemented since the last reload or expected to be implemented during the upcoming cycle.

6. Reload safety analysis parameters (mechanical, nuclear, and thermal-hydraulic) used in the current safety analyses.

This review will establish or define:

1. The nominal end of cycle (EOC) burnup window for the previous cycle.

2. The length, operational requirements, and license limit on cycle burnup for the reload cycle.

3. Reload design schedules.

4. The available reload fuel for use in the core.

5. Any constraints on the fuel to be used in the reload design.

6. Restrictions on the use and location of core insert components.

7. Expected plant operating conditions.

3.2.2 Fuel Loading and Pattern Determination

The determination of the fuel loading consists of finding a combination of enrichment and number of fresh fuel assemblies which meets the reload cycle energy and operational requirements established during the design initialization. Based on design experience from previous cycles, enrichment limits and economic calculations, the enrichment and number of feed assemblies is chosen. These assemblies along with the assemblies to be reinserted will be arranged in a preliminary loading pattern. This loading pattern is modeled and depleted to determine the cycle's energy output and power distributions. This is repeated with different numbers of feed assemblies and/or enrichments until the cycle energy requirements are met. During this time, shuffling of the assemblies to different locations to improve the power distribution may also be performed. Once a fuel loading is determined, the rearrangement of the fuel assemblies continues until the following conditions are satisfied:

1. The radial peaking factor values for the all rods out (ARO) and D bank inserted to the HFP insertion limits core configurations at hot full power (HFP), equilibrium xenon conditions, including uncertainties, do not exceed the COLR limits.

2. The moderator temperature coefficient at operating conditions meets the COLR limits.

3. Sufficient rod worth is available to meet the shutdown margin requirements with the most reactive control rod fully withdrawn.

4. Other key parameters considered integral to the confirmation of the loading pattern are acceptable.

When a loading pattern meets the above conditions, the fresh fuel enrichment, the number of fresh fuel assemblies, and the burnable poison requirements are set. The pattern is further evaluated to verify that other core physics related limits are likely to be met. Modification of the loading pattern is performed if specific limits are not met.

3.3 Nuclear Design Aspects of Reload Safety Analysis

3.3.1 Introduction

This section discusses the derivation of the core physics related key analysis parameters (hereafter referred to as key parameters) and the relationship of these parameters to the reload safety analysis. For each reload cycle, the effects of reload core physics related or plant related changes is evaluated to determine if the existing safety analysis is valid for the reload.

Mechanisms and procedures used to determine the validity of the current safety analysis are detailed in Sections 3.3.3 and 3.3.4. A conceptual discussion of all accidents of concern for the UFSAR and subsequent licensing submittals, and an outline of procedures used to derive each of the reload nuclear parameters important to the safety analysis are given in Section 3.3.4.

3.3.2 Safety Analysis Philosophy

To receive and retain an operating license from the NRC, it must be demonstrated that the public will be safe from consequences of plant operation. In addition, it is important to show that the plant itself will suffer, at most, only limited damage from all but the most incredible transients.

Plant safety is demonstrated by accident analysis, which is the study of nuclear reactor behavior under accident conditions. Accident analyses are usually performed in the initial stages of plant design and documented in the FSAR. The accident analyses for North Anna and Surry are typical in that the NSSS vendor performed the complete FSAR analysis. The four categories of plant conditions based on their anticipated frequency of occurrence and potential for public harm are described in References 10 and 11. The accident analyses consider all relevant aspects of the plant and core including the operating procedures and limits on controllable plant parameters (Technical Specifications) and the engineered safety, shutdown, and containment systems.

There are two stages in the typical safety analysis process, and these stages are applicable to either initial plant design analyses or analyses that may be initiated during reload core design. 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, which have a significant impact on the events under consideration.

During the original FSAR analysis, the NSSS vendor determined the key nuclear parameter values which had a high probability of being bounding over plant life. FSAR accident analyses were performed using these bounding values of the key parameters.

Subsequent to initial plant design, Dominion has verified the key parameters for Condition I, II, III, and IV UFSAR events and analyses (excluding LOCA) and the safety of its plants using its own analysis capability (References 6 and 13). The UFSAR documents acceptable plant safety via detailed results of accident analyses performed with the bounding values of key nuclear parameters. Plant safety is demonstrated if accident analysis results meet the applicable acceptance criteria. However, an unbounded key analysis parameter could occur in a reload cycle. For this reason, all key analysis parameters are re-evaluated for each reload.

Plant changes may take place between cycles or during a cycle. Examples are changes in operating temperatures and pressures, and setpoint changes. These changes may affect the key analysis parameters. If a key parameter value for a reload exceeds the current limit, an evaluation is performed using the reload value of the key parameter. This evaluation uses sensitivities for the impact of the parameter involved that have been demonstrated to be applicable to the reference analysis. Such an evaluation may indicate that a transient reanalysis is warranted if the unbounded parameter value exceeds the value in the reference safety analysis by a sufficient amount, or if the parameter impact is otherwise difficult to quantify. The general philosophy followed in performing an accident evaluation as opposed to a reanalysis is that the analyst must be able to clearly demonstrate that the results of an analysis performed with cycle-specific input would be less severe than the results of the reference analysis.

The reload evaluation process is complete if the acceptance criteria delineated in the UFSAR are met, and internal documentation of the reload evaluation is provided for the appropriate Dominion safety review. If, however, an accident reanalysis is necessary, more detailed analysis methods and/or Technical Specifications changes may be required to meet the acceptance criteria. Such changes will be processed in accordance with the relevant regulations (e.g., 10CFR50.59).

Therefore, the overall process is as follows:

1) Determine expected bounding key analysis parameters ("current limits").
2) Perform accident analysis using the bounding key analysis parameters and conservative assumptions.
3) Determine, for each reload, the value of each key analysis parameter.
4) Compare reload key analysis parameters to current limits.
5) Evaluate whether an accident reanalysis is needed based on the effect the reload key analysis parameters may have.
6) Perform reanalysis, change operating limits, or revise loading pattern as necessary.

This reload analysis philosophy has been used for the past reload cores for Dominion Surry Units 1 and 2 and North Anna Units 1 and 2 and will be used by Dominion in the future.

The accidents analyzed for the UFSAR and evaluated for each reload cycle are listed in Table 1. The key parameters to be determined for each reload cycle are listed in Table 2. The non-specific parameters (designated '(NS)' in Table 2) are generated by evaluating general core characteristics, while the specific parameters (designated '(S)' in Table 2) are generated by statically simulating an accident. The third type of key parameters are fuel performance and thermal-hydraulic related parameters (designated '(F)' in Table 2). The methods that will be employed by Dominion to determine these key parameters will be consistent with the methods documented in References 9, 12 and 14.

3.3.3 Non-Specific Key Parameters

Non-specific key parameters are derived by evaluating core characteristics for conditions bounding those expected to occur during the reload cycle to ensure that sufficiently limiting values of the parameter are determined. These conditions include conservative assumptions for such core parameters as xenon distributions, power level, control rod position, operating history, and burnup. These parameters are designated with '(NS)' in Table 2. Each non-specific key parameter generally serves as safety analysis input to several accidents including the accidents that also require specific key parameters, such as rod ejection. In addition, numerical uncertainty factors that are appropriate to the models being used are applied to the calculated parameter.

3.3.3.1 Rod Insertion Limits

Control rod insertion limits (RIL) define the maximum allowable control bank insertion as a function of power level. Rod insertion limits (RIL) are required in order to: maintain an acceptable power distribution during normal operation, obtain acceptable consequences following postulated accidents, and to insure that the minimum shutdown margin (SDM) assumed in the safety analyses is available. The current RILs for the unit are given in the plant COLR.

The rod insertion allowance (RIA) is the maximum amount of control bank reactivity which is allowed to be inserted in the core at HFP, and is selected to conservatively bound the amount of rod worth not available for shutdown margin over a range of power levels from HFP to HZP.

The relationship between the RIA and the RIL is such that insertion limits determined purely from RIA considerations are usually shallow enough that other bases for rod insertion limits such as acceptable power distributions and acceptable postulated rod ejection consequences are satisfied. The determination of the RIL is made by simulation of the control banks moving into the core with normal overlap while assuring the minimum shutdown margin is maintained over a range of power levels and insertions from HFP to HZP. The calculation is performed at the limiting times in cycle life (typically EOC), and for conservatism, the model is depleted in such a way that the burnup and xenon distribution force the power to the top of the core. This maximizes the worth of the inserted portion of the control banks which is not available for shutdown margin.

When tentative RIL lines have been selected by the method just outlined, they are then checked to see that they satisfy all of the other evaluation requirements. If any basis is not satisfied by the tentative insertion limits, the insertion limits are raised until the most limiting basis is satisfied. These limits are then checked against the COLR. If these RIL lines exceed those in the COLR, the COLR is revised accordingly.

3.3.3.2 Shutdown Margin

The shutdown margin (SDM) is the amount of negative reactivity by which a reactor is maintained in a subcritical state at HZP conditions after a reactor trip. Shutdown margin is calculated by determining the amount of negative reactivity available (control and shutdown bank worth) and finding the excess available once the positive reactivity associated with going from HFP to HZP conditions has been overcome.

The amount of rod worth available is calculated in two parts. First, calculations are performed to determine the highest worth single control rod or most reactive rod (MRR) for the loading pattern. Next, the total control rod worth assuming the MRR is stuck out of the core (N-1 rod worth) is determined and reduced an additional amount for conservatism. The N-1 rod worth is then reduced by the amount of rod insertion allowance to account for rods being inserted to the insertion limits.

Once the available shutdown reactivity is determined, calculations are performed to determine the amount of reactivity to be overcome to maintain the core in a subcritical state. The power defect is conservatively calculated by increasing the total moderator temperature change above that seen from HFP to HZP conditions. The effect of flux redistribution is included in the shutdown margin calculations. In addition, subcooled void collapse may occur when going from HFP to HZP, causing a positive reactivity insertion. A generic estimate of void collapse reactivity is typically used in the shutdown margin calculations.

The shutdown margin is the amount by which the available negative reactivity (rod worth) exceeds the positive reactivity to be overcome. This calculation is performed at the limiting times in cycle life (typically BOC and EOC).

3.3.3.3 Trip Reactivity Shape

The trip reactivity shape is a measure of the amount of negative reactivity entering the core (in the form of control rods) after a trip as a function of trip bank insertion. For conservatism in the accident analysis a minimum amount of trip worth based on near full power conditions is assumed to be available. This minimum trip worth is confirmed to be conservative by calculating the available trip worth for near full power conditions on a reload basis.

The actual parameter of interest to the accident analysis is reactivity insertion versus time. To determine this parameter, rod insertion versus time information is combined with the trip reactivity shape. The conservatism of the rod insertion versus time information used for the analysis is verified by rod drop measurements taken during the startup tests for each cycle.

The trip reactivity shape is generated and evaluated at the limiting times in cycle life (typically the depletion step with the most bottom peaked axial power distribution and the HFP end of reactivity depletion step). Control banks and/or xenon distributions are used to conservatively skew the power distributions prior to inserting the trip reactivity worth. The calculated total minimum trip reactivity worth is inserted in discrete steps and the integral worth corresponding to each step is determined. The calculated trip reactivity shape is then compared to the shape assumed in the safety analysis. The safety analysis curve is established to be a conservative representation of the reload values generated using the methodology above. A conservative trip reactivity comparison is confirmed if the safety analysis value shows less negative reactivity insertion for the major part of the rod insertion (i.e., except for the endpoints which are always equal), than the values calculated for the reload core.

3.3.3.4 Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedbacks, in particular the moderator temperature (density) coefficient and the Doppler power and temperature coefficients. The reactivity coefficient generation for the reload design was discussed in Section 2.2.2.

For each core there is a range of assumed values for the reactivity coefficients. The coefficients used as key analysis parameters are derived using the appropriate techniques and at the appropriate conditions to obtain the limiting (maxima and minima) values.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, a small reactivity coefficient value would be conservative. Some accidents and their analyses are not affected by reactivity feedback effects. Where reactivity effects are important to the analysis of an event, the use of conservatively large versus small reactivity coefficient values is treated on an event by event basis.

3.3.3.5 Neutron Data

Delayed neutrons are emitted from fission products. They are normally separated into six groups, each characterized by an individual decay constant and yield fraction. The delayed neutron fractions are calculated using the appropriate cross-section data. The total delayed neutron fraction (total β) is the sum of the delayed neutron fractions for the six groups.

The key analysis parameter is the β_{eff}, which is the product of the total β and the importance factor. The importance factor reflects the relative effectiveness of the delayed neutrons for causing fission. For some transients, it is conservative to use the minimum expected value of β_{eff}, while for others, the maximum expected value is more conservative. The use of conservatively large versus small β_{eff} values is treated on an event by event basis. β_{eff} is calculated at the times in cycle life that would produce the bounding values for the cycle (typically BOC and EOC).

The prompt neutron lifetime is the time from neutron generation to absorption. It is calculated by core averaging a region-wise power weighted prompt neutron lifetime calculated by a fuel lattice physics code for each region in the core.

This calculation is performed internally in SIMULATE. The key analysis parameter used for transients is the maximum prompt neutron lifetime, which is calculated at the limiting time in cycle life (typically EOC).

3.3.3.6 Power Density, Peaking Factors

The thermal margins of the reactor system are dependent on the initial power distribution. The power distribution is typically characterized by the radial peaking factor, $F_{\Delta H}$, and the total peaking factor, F_Q. The COLR specifies the peaking factor limits that apply to each cycle. Two key mechanisms are employed to constrain the peaking factors to be within the COLR limits: 1) the nuclear design of the core, by judicious placement of new and depleted fuel and by the use of burnable poisons, and 2) operational constraints, such as the axial power distribution control procedures and the rod insertion limits. Together, these mechanisms protect the core from power distributions more adverse than those allowed by the COLR.

For transients which may be DNB limited, the radial peaking factor, $F_{\Delta H}$, is of importance. The allowable radial peaking factor increases with decreasing power level. For transients which may be overpower limited, the total peaking factor, F_Q, is of importance. Above 50% power the allowable value of F_Q increases with decreasing power level such that the full power hot spot heat flux is not exceeded, i.e., F_Q * Power = design hot spot heat flux. For a reload, peaking factors are checked for various power levels, rod positions, and cycle burnups assuming conservative power distributions to verify the limits are not exceeded.

3.3.4 Specific Key Parameters

Specific key parameters are generated by statically simulating an accident. These parameters are designated with '(S)' in Table 2. The parameters are (or are directly related to) rod worths, reactivity insertion rates, or peaking factors. The static conditions are selected to be conservative for the accident and to account for variations in such parameters as initial power level, rod position, xenon distribution, previous cycle burnup, and current cycle burnup. In addition, numerical uncertainty factors which are appropriate to the models being used are applied to the calculated parameter.

3.3.4.1 Uncontrolled Control Rod Bank Withdrawal

The rod withdrawal accident occurs when control banks are withdrawn from the core due to some control system malfunction with a resulting reactivity insertion. The accident is assumed to be able to occur over a range of core powers. For rod withdrawal from subcritical (HZP), the parameter of interest is the maximum differential worth of two sequential control banks (D and C, C and B, etc.) moving together at HZP with 100% overlap. The rod withdrawal at power accident differs from the rod withdrawal from subcritical in that it occurs at–power and assumes that banks D and C are moving with the normal overlap. The parameter of interest is the maximum differential rod worth.

The following assumptions and conservatisms are used:

1) The axial xenon distribution is conservatively calculated at conditions that tend to maximize peak differential rod worth.

2) Calculations are performed at cycle burnups that tend to maximize the peak differential rod worth.

3.3.4.2 Rod Misalignment

Rod misalignment accidents result from the malfunctioning of the control rod positioning mechanisms, and include:

1) static misalignment of an RCCA (Rod Cluster Control Assembly, i.e., control rod).
2) single RCCA withdrawal.
3) dropped RCCA / dropped bank.

The key acceptance criterion for rod misalignment accidents is the minimum DNBR. The DNBR in the case of a rod misalignment accident is primarily a function of radial peaking factors ($F_{\Delta H}$). For conservatism, all of the rod misalignment cases are performed at the cycle burnups that maximize the radial peaking factors. Typically, a search is made to determine worst case rods for each type of rod misalignment. Uncertainty factors appropriate to the models used are applied. The maximum $F_{\Delta H}$ calculated for each of these types of rod misalignments are used to confirm that the DNB acceptance criterion has been met.

In the static misalignment accident, an RCCA is misaligned by being a number of steps above or below the rest of its bank. The RCCA misalignment below its bank is bounded by the dropped RCCA analyses for Surry and North Anna as described below. Note that the $F_{\Delta H}$ calculated for the RCCA misalignment upward analysis bounds the $F_{\Delta H}$ for the single RCCA withdrawal accident. However the single RCCA withdrawal accident is a condition III event and therefore a small percentage of fuel rods may be expected to fail. The event is analyzed to ensure that only a small percentage (<5%) of the fuel rods could exceed the fuel thermal limits and enter into DNB. The percentage of rods in DNB is determined through the use of a fuel rod census where the peak power for each rod in the core is tabulated.

The dropped RCCA(s) event (dropped rod or dropped bank) is conservatively evaluated using the methodology described in WCAP-11394-P-A (Reference 15). Dominion acquired the transient databases and methodology information necessary to perform the dropped rod analyses of Reference 15 from Westinghouse. Dominion has performed evaluations which demonstrated the applicability of the methodology, the correlations, and the transient database for the analysis of the dropped rod event for the North Anna and Surry Power Stations. This methodology for the evaluation of the dropped rod(s) event has been implemented for both the North Anna and Surry Power Stations pursuant to the provisions of 10CFR50.59.

The dropped RCCA(s) event evaluation consists of three analyses: system transient, nuclear, and thermal-hydraulic. The transient response is calculated using a system code which simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. Nuclear models are used to obtain hot channel factors consistent with the primary system conditions at the statepoints generated by the transient simulation. These analyses are performed using a parametric approach so that cycle specific conditions may be evaluated using the data generated from the three analyses above. Specifically, these analyses provide: 1) statepoints, i.e., the reactor power, pressure, and temperature at the most limiting time in the transient and 2) the radial peaking factor at the most limiting conditions in the transient. The DNB design basis is shown to be met using a core thermal-hydraulics code by combining the conditions associated with 1 and 2.

The reload evaluation of the dropped rod(s) event involves an analysis using two cycle-specific, key parameters: the rod worth available for withdrawal and the moderator temperature coefficient. These parameters are used to determine the radial peaking factor prior to the dropped RCCA(s) event which would produce conditions at the DNBR limit during the transient for a range of dropped RCCA(s) worths. These predrop radial peaking factors are compared to the reload design predictions to confirm that the limiting predrop conditions for DNB do not occur during the cycle.

3.3.4.3 Rod Ejection

The rod ejection accident results from the postulated mechanical failure of a control rod mechanism pressure housing such that the coolant system pressure ejects the control rod and drive shaft to the fully withdrawn position. This results in rapid reactivity insertion and high peaking factors. Rod ejections are analyzed at the beginning and end of the cycle at hot zero power and hot full power.

The following scenario describes the rod ejection. With the core critical (at either HZP or HFP) and the control rods inserted to the appropriate insertion limit, the pressure housing of the most limiting ejected rod fails. The rod is ejected completely from the core resulting in a large positive reactivity insertion and a high F_Q in the vicinity of the ejected rod. The most limiting ejected rod is that rod that gives the highest worth (or positive reactivity addition) and/or the highest F_Q when ejected from the core.

The rod ejection accident produces a brief power excursion which is limited by Doppler feedback. The rod ejection accident is a Condition IV event that has a potential for fuel damage and some limited radioactivity releases. A more detailed discussion of the rod ejection accident scenario and analysis may be found in Reference 13.

The key parameters for the rod ejection accident are the ejected rod worth and total peaking factor, F_Q. The rod ejection key analysis parameters for the bounding power levels and burnups are derived for each reload core. The models used for the calculation of axial powers are depleted in such a way as to insure that, at EOC, the top part of the core has less burnup than would be expected from a best estimate calculation based on operational history. The depletion is performed with D Bank partially inserted, which insures higher worths and peaking factors, for both HZP and HFP, as compared to the best estimate axial burnup shape.

The rod ejection parameter derivation is performed in a conservative manner. Although the rod ejection accident is limited by Doppler feedback, the key analysis parameters are derived with all feedback frozen. Conservatism is ensured by calculating all physics parameters at steady state conditions using the "adiabatic assumption." This assumption asserts that any fuel damage which might occur during the transient takes place in a small time interval immediately following the ejection of the rod and before the thermal-hydraulic feedback effects of the core become important. This freezing of the core feedback effects leads to larger values of the total power peaking factor and ejected rod worth than would otherwise be expected in the transient.

3.3.4.4 Steamline Break

The steamline break (or steambreak) accident is an inadvertent depressurization of the main steam system or a rupture of a main steamline. The first type of event is referred to as a "credible break" and is a Condition II event. The second type is called a "hypothetical break" and is a Condition IV event.

The credible steambreak accident can occur when any one steam dump, relief, or safety valve fails to close. The hypothetical steambreak is a rupture or break in a main steamline. For the credible break the safety analysis must show that no DNB and subsequent clad damage occurs. For the hypothetical break, DNB or clad damage may occur, but the safety analysis must show that the 10CFR100 limits are not exceeded.

The steamline depressurization caused by this accident results in a temperature decrease in the reactor coolant which in the presence of a negative moderator temperature coefficient results in a positive reactivity insertion. The reactivity insertion and a possible return to critical are more limiting when the MTC is most negative (typically at EOC).

The starting point for both analyses is a reference safety analysis using RETRAN. The input parameters for the RETRAN model include nuclear parameters which are considered conservative for the reload core being analyzed. RETRAN predicts, for various shutdown margins and secondary break sizes, the system trends as a function of time. The nature of the analysis is such that although the plant volumes, temperatures and flows are reasonably detailed, more specific core DNB determinations must be made using more detailed methods.

First, a detailed nuclear calculation is performed at the limiting time in cycle, HZP power conditions with all rods fully inserted, except the highest reactivity worth stuck rod. These conditions are conservative initial assumptions for steambreak (see References 10 and 11). Next, conditions including power, non-uniform inlet temperature distribution, pressure, and flow (derived from the RETRAN code output data at the point where the minimum DNBR may occur) are input, and peaking factors and axial power distributions are generated. The stuck rod is assumed to occur in the coldest quadrant to maximize reactivity insertion.

Several limiting statepoints are chosen from RETRAN for minimum DNBR analysis. The temperature and pressure information from these statepoints along with peaking factor information from the detailed nuclear calculation are input to the thermal-hydraulic code to conservatively determine the minimum DNBR for the steambreak transient.

3.3.4.5 LOCA Peaking Factor Evaluation

A loss of coolant accident (LOCA) is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system. The LOCA reload evaluation methodology that is employed by Dominion is consistent with the fuel vendor methodology used for establishing and validating the operational limits for allowable core power distributions. A description of the reload validation methodology can be found in References 5, and 14.

The primary LOCA key analysis parameter is $F_Q(z) * P$, where $F_Q(z)$ is total peaking factor as a function of core height and P is core average power (fraction of rated). This key parameter is compared to a COLR limit which is based on the total peaking factor assumed in the applicable LOCA analysis. The LOCA operational limits for core power distribution are intended to accommodate a range of core operating conditions that tend to maximize the peak linear heat generation rate and axial power distribution. The LOCA limit envelope is conservative with respect to the power shapes assumed for large and small break LOCA analyses. The specific form of the limit expression is dependent upon LOCA evaluation model methodologies that are generally specific to individual fuel type. The limit envelope is expressed in terms of $F_Q(z) * P$, multiplied by one or more normalization factors, which may be functions of core height or burnup.

To determine these parameters Dominion uses one of two reload analysis methods: 1) a standard CAOC FAC analysis as described in Reference 5 or 2) the Relaxed Power Distribution Control (RPDC) methodology as described in Reference 9.

The key parameters are determined analytically for RPDC in much the same manner as under the CAOC methodology. Each methodology involves calculational verification that the maximum F_Q will not exceed the LOCA limit for operation within the established ΔI bands. The ΔI parameter is defined as the difference in power in the top and bottom halves of the core, expressed as a percentage of core power. The two methodologies can be contrasted as follows. The CAOC analysis determines that the F_Q limit is met when the unit is operated within a narrow ΔI band which is constant over the range of 50% to hot full power. The RPDC analysis determines an allowable ΔI band that is a function of power, within which the unit may operate and meet the F_Q limit. The allowable ΔI band from the RPDC analysis is generally larger than the ΔI band assumed in the CAOC analysis.

To summarize, the procedure for insuring LOCA safety analysis coverage for the reload cycle consists of: 1) determining the applicable LOCA F_Q limit envelope; 2) determining the reload core maximum $F_Q(z) * P$ values for all normal operational modes; and 3) specifying the appropriate COLR changes to ensure that the reload $F_Q(z) * P$ values are bounded by the LOCA F_Q envelope.

3.3.4.6 Boron Dilution

Reactivity can be added to the reactor core by feeding primary grade (unborated) water into the Reactor Coolant System (RCS) through the Chemical and Volume Control System (CVCS). This addition of reactivity by boron dilution is intended to be controlled by the operator. The CVCS is designed to limit the rate of dilution even under various postulated failure modes. Alarms and instrumentation provide the operator sufficient time to correct an uncontrolled dilution if it occurs. Boron dilution accidents are Condition II events and are evaluated for all phases of plant operation.

The core boron concentrations and the minimum shutdown margins to be maintained for the different phases of plant operation are specified in the plant Technical Specifications, the COLR and plant procedures. The minimum shutdown margins for credible cases are specified in order to provide the required operator response time. For each reload, calculations are performed to demonstrate that the minimum shutdown margins are met at the core conditions and boron concentrations specified.

3.3.4.7 Overpower Evaluations

An overpower condition occurs in a reactor when the 100% power level is inadvertently exceeded due to incidents such as an uncontrolled boron dilution or an uncontrolled rod withdrawal. The overpower evaluation key analysis parameter for both of these accidents is the maximum linear heat generation rate (LHGR), in kw/ft. The methodology used to derive the key analysis parameter for CAOC is described in Reference 14. The analogous methodology for RPDC is described in Reference 9.

3.3.5 Non-Nuclear Design Key Parameters

Non-nuclear design key parameters are safety analysis inputs from non-nuclear areas such as core fuel performance and thermal-hydraulics. These parameters are designated with '(F)' in Table 2. Changes to these parameters are infrequently made and are typically linked to changes in either the plant operating conditions or fuel products. These inputs are reviewed for each reload cycle to ensure that the safety analysis assumptions continue to bound the key parameter values for the current plant configuration.

3.4 Reload Safety Evaluation Process

As has been discussed in previous sections, past analytical experience has allowed the correlation of the various accidents with those key safety parameters which have a significant impact on them. When a key safety analysis parameter exceeds its previously defined safety analysis limit, the particular transient(s) in question must be evaluated. This evaluation may be based on known sensitivities to changes in the various parameters in cases where the change is expected to be minimal and the effects are well understood. In cases where the impact is less certain or the effects of the parameter on the results is of a more complicated nature, then the transient will be reanalyzed. The majority of these reanalyses are performed with the Virginia Power RETRAN models described in References 6, 7, and 13.

Each transient reanalysis method and assumption will be based on a conservative representation of the system and its response. This includes appropriate initial conditions, conservative reactivity feedback assumptions, conservative reactor trip functions and setpoints, and assumptions concerning systems performance. More discussion of these items can be found in References 6, 7 and 13.

Transients requiring core minimum DNBR analyses are analyzed using the applicable thermal-hydraulic code(s) and model(s) and applicable statistical DNB methodology that are listed in the COLR section of the plant Technical Specifications. The necessary core operating condition inputs are determined from the RETRAN code. Peaking factor inputs are determined from the appropriate nuclear design code.

TABLE 1

EVALUATED ACCIDENTS

CONDITION II EVENTS

a) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition
b) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
c) Rod Cluster Control Assembly Misalignment
d) Uncontrolled Boron Dilution
e) Partial Loss of Forced Reactor Coolant Flow
f) Startup of an Inactive Reactor Coolant Loop
g) Loss of External Electrical Load and/or Turbine Trip
h) Loss of Normal Feedwater
i) Loss of all Off-Site Power to the Station Auxiliaries (Station Blackout)
j) Excessive Heat Removal Due to Feedwater System Malfunctions
k) Excessive Load Increase Incident
l) Accidental Depressurization of the Reactor Coolant System
m) Accidental Depressurization of the Main Steam System

CONDITION III EVENTS

a) Complete Loss of Forced Reactor Coolant Flow
b) Single Rod Cluster Control Assembly Withdrawal at Power
c) Small Break Loss of Coolant Accident

CONDITION IV EVENTS

a) Rupture of a Steam Pipe
b) Rupture of a Feedline
c) Single Reactor Coolant Pump Locked Rotor
d) Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)
e) Large Break Loss of Coolant Accident

TABLE 2

KEY ANALYSIS PARAMETERS

1) Core Thermal Limits (F)
2) Moderator Temperature (Density) Coefficient (NS)
3) Doppler Temperature Coefficient (NS)
4) Doppler Power Coefficient (NS)
5) Delayed Neutron Fraction (NS)

6) Prompt Neutron Lifetime (NS)
7) Boron Worth (NS)
8) Control Bank Worth (NS)
9) Rod Worth Available for Withdrawal (S)
10) Ejected Rod Worth (S)

11) Shutdown Margin (NS)
12) Boron Concentration for Required Shutdown Margin (NS)
13) Reactivity Insertion Rate due to Rod Withdrawal (S)
14) Trip Reactivity Shape and Magnitude (NS)
15) Power Peaking Factors (S)

16) Maximum F_Q * P (S)
17) Radial Peaking Factor (S)
18) Ejected Rod Hot Channel Factor (S)
19) Initial Fuel Temperature (F)
20) Initial Hot Spot Fuel Temperature (F)

21) Fuel Power Census (NS)
22) Densification Power Spike (F)
23) Axial Fuel Rod Shrinkage (F)
24) Fuel Rod Internal Gas Pressure (F)
25) Fuel Stored Energy (F)

26) Decay Heat (F)
27) Maximum Linear Heat Generation Rate (LHGR) (S)
28) Maximum LHGR Vs. Burnup (F)

<u>Parameter Designation</u>
NS: Non-Specific
S: Specific
F: Fuel Performance and Thermal-Hydraulics Related

3.5 Nuclear Design Report, Operator Curves, and Core Follow Data

Before the operation of the cycle, a Nuclear Design Report which documents the nuclear design calculations performed in support of the cycle operation is issued. In addition, operator curves and core follow data (e.g., startup physics testing data, shutdown margin data, nuclear instrumentation data, etc.) are also generated for specific core configurations based on the calculations for the nuclear design report. The nuclear design report, operator curves, and core follow data are for use by station personnel in the operation of the cycle.

The parameters calculated for the reload safety evaluation are calculated for the most conservative conditions and in addition have uncertainty factors applied to them. This same practice is used in the derivation of the shutdown margin data and some of the nuclear instrumentation and operator curve data. The remaining nuclear instrumentation and operator curve data, startup physics testing data, and nuclear design report data are best estimate calculations for conditions which the plant may see and be anticipated to operate under. For the most part these parameters are calculated for actual previous end-of-cycle conditions. However, where a parameter shows little or predictable variation for different previous end-of-cycle burnups the calculations may be made for the nominal end of the burnup window if values are needed prior to shutdown of the previous cycle.

The parameters calculated on a reload basis for a design report include:

1) Boron endpoints and boron worths at various core configurations;

2) Reactivity coefficients and defects (Isothermal temperature coefficients, Doppler temperature coefficients, isothermal temperature defects, total power defects, etc.) at various core conditions;

3) Integral and differential bank worths at various core conditions;

4) Delayed neutron data and prompt neutron lifetime;

5) Relative power distributions at various core conditions;

6) Iodine and Xenon concentrations and worths at various core conditions;

7) Reactivity due to isotopic decay (excluding xenon) at various core conditions;

8) Assembly-wise burnup as a function of cycle burnup;

9) Most reactive stuck rod worths at various core conditions;

10) Miscellaneous calculations to support operator curve generation or core follow input.

Core physics measurements taken during the cycle startup and operation are compared to the physics design predictions documented in the Nuclear Design Report to insure that the plant is being operated within safety limits. Results of the measurements and the comparisons to predictions are published as a Startup Physics Test Report and a Core Performance Report for each reload cycle.

SECTION 4.0 - SUMMARY AND CONCLUSIONS

The in-house fuel management and reload design capability developed by Dominion utilizes models and techniques developed in-house and licensed by the NRC. These models have been shown to accurately predict the necessary core parameters and simulate the core behavior necessary to perform the reload design process outlined in this report.

The first step in the reload safety analysis of a core is the preparation of a listing of the current limits for core physics related key analysis parameters. Appropriate calculations are performed for generation of the reload values of the key parameters (generally static nuclear calculations) based on this list. Evaluation and, if necessary reanalysis of any accidents (using transient methods) is performed as required by the results of the key parameter calculations. A Reload Safety Evaluation (RSE) report is then issued documenting the results of the safety analysis for the reload cycle. For the typical reload, the derived key analysis parameters are bounded by the current limit key analysis parameters.

If the current limits are exceeded, that event may be handled in a number of ways. If the parameter only slightly exceeds its limits, or the affected transients are relatively insensitive to that parameter, a simple quantitative evaluation may be made which conservatively estimates the magnitude of the effect and explains why an actual reanalysis does not have to be made. The current limit is not changed.

If the deviation is large and/or expected to have a more significant or not easily quantifiable effect on the accident, the accident is reanalyzed following standard procedures (such as those used in the FSAR analyses or other NRC approved methods). After the reanalysis is performed, and if the results of the reanalysis meet all applicable licensing criteria the reload evaluation is complete upon completion of the appropriate internal documentation and review.

Sometimes reanalysis will produce unsatisfactory results and other steps may have to be taken. Technical Specifications changes, COLR changes, or core loading pattern changes are typical adjustments that may be required. Raising the rod insertion limits, in order to reduce the ejected rod Fq and worth, is an example of a COLR change. If Technical Specifications changes are necessary to keep key parameters bounded, these changes must be approved by the NRC in accordance with 10CFR50.59 prior to implementation at the plant. In addition, loading pattern adjustments may be required to bring some key parameters within the current limits or reduce the size of the deviation.

SECTION 5.0 - REFERENCES

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4. S. M. Bowman, "The Vepco NOMAD Code and Model," VEP-NFE-1-A (May 1985).

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8. G. R. Poetschat, et al., "EPRI-PRESS, Volume 2: User's Manual," Part II, Chapter 5, ARMP-02, EPRI NP-4574, (August 1986).

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SECTION 5.0 - REFERENCES (CONTINUED)

14. T. Morita, D. M. Lucoff, et al., "Topical Report Power Distribution Control and Load Following Procedures," WCAP-8385, (September 1974).

15. R. L. Haessler, D. B. Lancaster, et al., "Methodology for the Analysis of the Dropped Rod Event," WCAP-11394-P-A, (January 1990).

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APPENDIX A

RAI Questions and Responses Set # 1

(Sent by Dominion letter Serial No. 02-280 to NRC dated May 13, 2002)

Total Pages : 27

Attachment

REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT
VEP-FRD-42, Revision 2

North Anna Power Station Units 1 and 2
Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

In April 15 and 16, 2002 discussions with the NRC staff, regarding Dominion's Topical Report, VEP-FRD-42, Revision 2, "Reload Nuclear Design Methodology," the following additional information was requested.

Question 1:

Is the Dominion reload methodology discussed in Topical Report VEP-FRD-42, Revision 2, intended to be applicable only for Westinghouse and Framatome ANP fuel types? If the intent is for other fuel types, please provide a discussion regarding how applicability determinations will be made and the process for determining the need for prior NRC approval.

Response:

The methodology discussed in VEP-FRD-42, Revision 2 is supported by extensive nuclear design predictions that encompass various evolutionary changes in fuel design features for Westinghouse fuel. Such predictions have been made for more than 40 reload cores, loaded in both North Anna and Surry reactors. Although the intended extension of this methodology is for the analysis of Framatome ANP fuel, the methodology is sufficiently robust for use on any fuel product with similar features. The methodology has several key elements, none of which are inherently dependent upon a specific fuel design or manufacturer. These key attributes of the methodology are:

- Analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits
- Use of bounding key parameter values in reference safety analyses
- Recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data
- Representation of key fuel features via detailed inputs in core design and safety analysis models
- Fuel is modeled using approved critical heat flux (CHF) correlations demonstrated to be applicable and within the range of qualification

The Dominion reload design methodology focuses upon determining appropriately conservative values for two types of parameters: 1) the bounding value for key parameters assumed in the safety analyses and 2) the values for these same key parameters calculated for each reload core. The first parameter set constitutes the allowable limits for which the existing safety analyses remain valid. The reload values are determined for each specific core with the objective of confirming that they remain within the limit values. Application of this methodology to alternate fuel types would be accomplished in a fashion that preserves this fundamental approach. Prior to the use of the Dominion nuclear reload methodology for other fuel types, it is necessary to confirm that the impact of the fuel design and its specific features can be adequately modeled with the Dominion nuclear design and safety analysis codes. This includes comparison with appropriate benchmark data to confirm the capability to model the specific fuel features and to determine the inherent accuracy of such predictions. Results of these comparisons would also be used to determine whether any changes are needed in uncertainties that are applied to the nuclear calculations. If the features of an alternate fuel design can be modeled with comparable accuracy to the existing models and fuel design and require no change in the applied uncertainty factors, the applicability of the nuclear design portion of the methodology is established. This approach confirms that there should be no significant effect upon calculated values of reload key parameters. To determine applicability of safety analysis codes for analysis of alternate fuel products, a similar modeling capability assessment would be performed. This assessment would involve incorporating the appropriate detailed fuel design inputs into safety analysis code calculations and verifying that existing codes and models conservatively model the fuel behavior. This would be accomplished either by direct evaluation of the key phenomena or comparison to available vendor calculation results. The need to obtain prior NRC approval for these changes is governed by the requirements of 10 CFR 50.59, which in Sections (a)(2) and (c)(2)(viii) includes provisions that are relevant to methodology changes. If the changes necessary to accommodate another fuel product required changes to the reload methodology of VEP-FRD-42, Revision 2, these would be submitted for prior NRC review and approval.

Question 2:

The licensee states that the minor changes in Framatome ANP fuel features that could affect safety analysis design inputs are within the modeling capability of Dominion safety and core design analysis codes. Please verify that Framatome ANP fuel features are within all restrictions and limitations of Dominion safety and core design analysis codes.

Response:

Core Design Models

From a core design perspective, the differences in modeling Framatome ANP fuel relative to Westinghouse fuel are small and are accommodated using model input parameters. These differences are similar in magnitude to incremental changes in Westinghouse fuel over time, which have been successfully modeled. Minor changes include spacer grid differences, a slight increase in fuel density, and a slight difference in the position of the fuel stack. The grid differences are primarily due to the presence of intermediate flow mixer grids. In the PDQ and NOMAD models, grids are not explicitly modeled, but are homogenized over the entire length of the fuel stack. The effect of more grid material (primarily zirconium) is directly modeled in PDQ via input parameters (treated as nuclides) representing grid material and moderator displacement. The macroscopic cross section effect is transferred to the NOMAD model from PDQ. Similarly, cross sections in the PDQ model are a function of fresh fuel isotopic content; therefore, the density effects are also directly modeled.

Minor changes in fuel alignment have occurred in the past due to evolutionary changes in Westinghouse fuel products, such as the incorporation of protective lower grids. If there is a significant shift in the relative alignment of the burnable poison (BP) and the fuel, the burnable poison position is directly modeled by axially volume weighting the BP input in the axial nodes where the BP/fuel boundary changes. Comparison of measured and predicted Framatome ANP lead test assembly (LTA) axial and integral power distributions over three cycles of operation provides direct confirmation of the accuracy of the axial weighting, grid modeling, and fuel density modeling techniques.

RETRAN Models

In preparation for application of the Dominion RETRAN model to Framatome ANP fuel, specific card (record) overlays to the RETRAN input cards were developed. These overlays were developed such that appending them to the end of the current, Westinghouse fuel based model creates a Framatome ANP-specific RETRAN model.

<u>Fuel properties</u>

The Framatome ANP overlays were developed from fuel and clad properties data supplied by Framatome ANP which are consistent with those used in the approved Framatome ANP safety analysis models. Formal documents developed under the Framatome QA program were developed to transmit this data. Fuel properties covered included:

- Material properties of the three conductor materials (the fuel pellet, the pellet-cladding helium gap, and the M5 cladding)
  - Thermal conductivity
  - Volumetric heat capacity
  - Thermal linear expansion coefficient

These data were converted into the RETRAN input structure. Plots of the data, the analytical equations used to develop the data, and graphical and numerical comparisons were presented of the Framatome ANP data to the corresponding data in:

- the existing W fuel based model
- The International Nuclear Safety Center (INSC) Material Database, Argonne National Laboratory for the US Department of Energy
- NUREG/CR-6150 (MATPRO)

Generally, only minor differences in the data were observed. The most significant property differences are those associated with the M5 versus ZIRLO cladding.

Core Geometry Input

The Framatome ANP overlays were developed from Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. All dimensional data were transmitted via documentation that was formally prepared and reviewed under Framatome ANP's 10 CFR 50 Appendix B QA program. Input changes were developed in the following areas:

- Core bypass geometry
  - Volume
  - Flow area
  - Flow diameter
- Active core geometry
  - Volume
  - Flow area
  - Flow diameter
- Reactor vessel flow path length and area
- Reactor vessel form loss coefficients
- Reactor core target pressure drops
- Active core inlet mass flow rate
- Geometry of the active core heat conductors

The calculation of each RETRAN input was documented in a reviewed engineering calculation and prepared in accordance with Dominion's 10 CFR 50 Appendix B Quality Assurance Program. The engineering calculation presents detailed comparisons of the Framatome ANP overlay parameters to the base model parameters in tabular format. The parameter changes represented minor adjustments with respect to the existing inputs.

Steady-state initializations were run with and without the Framatome ANP overlays to ensure adequate convergence of the new models. Detailed comparisons of the steady-state initialization results were presented in the engineering calculation in tabular format. Review of these results showed that there are only minor differences in the Westinghouse Fuel Based and Framatome ANP Fuel based models.

The modeling changes associated with Framatome ANP fuel fall within the restrictions and limitations of the Dominion core design and safety analysis codes.

Question 3:

Use of Framatome ANP fuel will require changes to various computer model inputs. Please discuss how the practices of NRC Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses", are applied in making these model changes.

Response:

General comment

The scope and applicability of GL 83-11 Supplement 1 is discussed in Attachment 1 to GL 83-11. An excerpt relevant to this discussion is as follows:

> "This attachment presents a simplified approach for qualifying licensees to use NRC-approved analysis methods. Typically, these methods are developed by fuel vendors, utilities, national laboratories, or organizations such as the Electric Power Research Institute, Incorporated (EPRI). To use these approved methods, the licensee would institute a program (e.g., training, procedures) that follows the guidelines below and notify the NRC that it has done so.
>
> The words 'code' and 'method' are used interchangeably within this document, i.e., a computer program. In many cases, however, an approved method may refer not only to a set of codes, an algorithm within a code, a means of analysis, a measurement technique, a statistical technique, etc., but also to selected input parameters which were specified in the methodology to ensure conservative results. In some cases, due to limitations or lack of appropriate data in the model, the code or method may be limited to certain applications. In these cases, the NRC safety evaluation report (SER) specifies the applicability of the methodology."

Dominion is proposing to apply the existing methodology of VEP-FRD-42 to the analysis of Framatome ANP fuel. Therefore GL 83-11, which involves code and methodology changes, is not directly applicable. However, the principles outlined in Attachment 1 to the GL have been followed in the development of Framatome ANP specific models (input changes) for use with existing, approved codes and methods. The process of Framatome ANP specific model development will be discussed in that context.

Dominion has established and uses a formal GL 83-11 program. Dominion notified the NRC of the establishment of this program in Reference 3.1. This program addresses all of the elements of GL 83-11, Supplement 1, Attachment 1 identified below:

- Application Procedures
- Training and Qualification of Licensee Personnel
- Comparison Calculations
- Quality Assurance and Change Control
- Error/Problem Reporting

Dominion's reload analysis methodology as set forth in VEP-FRD-42 has been developed and qualified in accordance with these principles. For example:

Application Procedures

Specific analytical steps for performing a reload analysis are outlined in the Nuclear Core Design (NCD) Manual and the Safety Analysis Manual (SAM). The NCD Manual is structured such that the calculational process is transparent to fuel type. Specific NCD code input varies according to fuel type as necessary (i.e., grid size differences, grid material difference, etc.). Detailed techniques for determining model input are provided in the NCD Manual and are supplemented by model setup calculations for previous fuel types, and by evaluation of proposed fuel changes in an operational impact assessment. The operational impact assessment is mandated by a departmental Implementing Procedure, which requires evaluations of proposed core changes in light of SOER 96-02.

The Safety Analysis Manual provides detailed calculational instructions for providing reload-specific thermal hydraulic evaluations as well as a chapter of guidance for the performance of analyses of the specific accidents presented in Chapters 14 and 15 of the Surry and North Anna UFSARs, respectively. Typically, accident reanalyses are not performed for core reloads, in that the key analysis parameters are found to be bounded by the assumptions in the accident analyses.

Quality Assurance/Change Control

Core Physics Models – The answer to Question 2 deals with the Framatome ANP changes of importance to the core design models. The changes were identified and evaluated in an operational impact assessment, and specific input changes were determined for Framatome ANP Lead Test Assembly (LTA) modeling using the same techniques used for other fuel types.

RETRAN Models - In preparation for application of the Dominion RETRAN model to Framatome ANP fuel, specific card (record) overlays to the RETRAN input cards were developed. These overlays were developed such that appending them to the end of the current, Westinghouse fuel based model creates a Framatome ANP-specific RETRAN model.

Specific changes modeled were discussed in detail in the Response to Question 2.

The Framatome ANP overlays were developed from the following data:

- Framatome ANP supplied fuel and clad properties data that are consistent with those used in the approved Framatome ANP safety analysis models. Formal documents developed under the Framatome QA program were developed to transmit this data.

- Framatome ANP supplied dimensional data for the Framatome ANP fuel assemblies. All dimensional data was transmitted via documentation that was formally prepared and reviewed under Framatome ANP's 10 CFR 50 Appendix B QA program.

Comparison Calculations

Previously submitted topical reports for PDQ Two Zone Models, NOMAD, and TIP/CECOR contain extensive model benchmarking information. In addition, the accuracy of power distribution predictions for Framatome ANP LTA fuel has been documented for three cycles of operation.

Dominion's RETRAN model has been benchmarked against the following items:

- Westinghouse analyses of record as published in the Surry and North Anna FSAR's in the 1970's and 1980's - see Section 5.2 of VEP-FRD-41A.
- Plant transient data, including:
  - Surry and North Anna pump coastdown tests - see Section 5.3 of VEP-FRD-41A
  - North Anna Unit 1's cooldown and safety injection transient September 25, 1979- See Section 5.3.3 of VEP-FRD-41A.
  - North Anna Unit 1's July 1987 Steam Generator Tube Rupture-see Section 3.2 of Attachment 1 to Letter 93-505, Supplemental Information on the RETRAN NSSS Model, August 10, 1993.
  - Westinghouse LOFTRAN calculations for the following:
    - Reactor trip with turbine trip
    - Turbine trip without direct reactor trip
    - Simultaneous loss of 3 reactor coolant pumps
    - See VEPCO Letter No. 376A, August 24, 1984.

These benchmark calculations have been studied and understood and support the conclusion that the Dominion RETRAN model provides a realistic representation of the Surry and North Anna reactor plants. Conservative results are ensured when the RETRAN model is used for licensing basis analyses through the use of appropriate input assumptions governing availability and performance of systems and components, core reactivity coefficients, and uncertainties in initial conditions.

Reference:

3.1 Virginia Power Letter to the NRC (Serial No. 00-087), dated March 15, 2000, Qualifications for Performing Safety Analyses, Generic Letter 83-11, Supplement 1.

Question 4:

The Dominion Topical Report on Reload Methodology (VEP-FRD-42, Revision 2) includes four computer codes or code modifications which have been implemented for use under the provisions of 10 CFR 50.59:

- PDQ Two Zone - replaced PDQ Discrete Model and the FLAME Model (Transmitted via Ref. 2 and 3 in VEP-FRD-42)
- NOMAD - was significantly modified (transmitted in Ref. 5 in VEP-FRD-42)
- TIP/CECOR - (Transmitted via Ref. 3 in VEP-FRD-42)
- RETRAN - code modifications (Transmitted via Ref. 7 in VEP-FRD-42)

References 2, 3 and 5 in VEP-FRD-42, Revision 2, and an additional letter not referenced in this topical (dated March 1, 1993) requested NRC review and approval of the associated topical reports for the first three codes listed. Dominion (VEPCO at the time) also recognized that these would need NRC approval because North Anna and Surry are COLR plants. For RETRAN, no review was requested, and the transmittal letter was for NRC information only. As such,

a. Have those topical reports/codes and code modifications been reviewed and approved for use by the NRC staff? If so, please provide a reference to the staff SERs. If not, then codes and models will need to be reviewed and approved to permit use in the COLR.

b. Have they been used by Dominion as part of the Reload Design Methodology? If so, why is their use acceptable and not a violation of the requirements for implementing a COLR? Generic Letter 88-16 requires that NRC approved methodology be referenced in the COLR, and VEP-FRD-42, Revision 1 is referenced in the COLR. VEP-FRD-42, Revision 1, and therefore the COLR does not reflect what Dominion is currently using as part of its Reload Methodology.

c. Please submit Technical Specification changes to incorporate references to actual methodology being used.

d. What procedures and controls do you use on the application of computer codes and models for core design and safety analysis? In other words, how does the core designer or safety analyst know he or she is using the right tools?

Response to 4a:

PDQ Two-Zone Model

The PDQ Two-Zone Model was transmitted via References 4.1 and 4.2:

Reference 4.1 requested approval of the 3-D coarse mesh PDQ model (the two-zone model) by the end of the 1st Quarter, 1991 to support the use of axially zoned flux suppression inserts (FSI's) in Surry Unit 1 Cycle 12.

Reference 4.2 reiterated the need for the 3D capability, to support FSI's, although first use had shifted to Cycle 13. We noted that to support the planned use of FSI's in Cycle 13 would require approval of the topical by the end of the 1st Quarter, 1993. Since the NRC review schedule would not support this, we proposed implementation of the methodology via 10 CFR 50.59 in advance of formal NRC approval of the reports. As noted in Reference 4.2, telephone conversations were held with the Staff on October 7 and 14, 1992 to discuss the 10 CFR 50.59 approach. Although the NRC could not concur with the specific application without formal review, the staff agreed with the use of 10 CFR 50.59 evaluations where applicable. Reference 4.2 documented these discussions. Dominion's request for formal review of the topicals was not withdrawn, although these changes were implemented via 10 CFR 50 .59.

On March 1, 1993 Dominion submitted Topical Report VEP-NAF-1, Supplement 1, entitled, "The PDQ Two-Zone Model," again for review and approval. The Supplement describes a coarse mesh 2-D model that is closely related to and used in conjunction with the 3-D model. We again stated our intent to implement the code via 10 CFR 50.59 prior to NRC review and approval, but requested concurrent review of the VEP-NAF-1 and Supplement 1.

The 10 CFR 50.59 approach to changing "elements of a methodology" as defined in NEI 96-07, Rev. 1 and endorsed by USNRC Regulatory Guide 1.187 is applicable in the case of the PDQ Two-Zone models. We refer specifically to NEI 96-07 Section 4.3.8, entitled, "Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Bases or in the Safety Analyses?"

The relevant discussion is as follows:

"... The following changes are not considered departures from a method of evaluation described in the UFSAR:

- Departures from methods of evaluation that are not described, outlined or summarized in the UFSAR (such changes may have been screened out as discussed in Section 4.2.1.3).
- Use of a new NRC-approved methodology (e.g., new or upgraded computer code) to reduce uncertainty, provide more precise results or other reason, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable SER. The basis for this determination should be documented in the licensee evaluation.
- Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER".

Subsection 4.3.8.1 of NEI 96-07 provides guidance for making changes to one or more elements of an existing method of evaluation used to establish the design bases or in the safety analyses. Specifically,

"4.3.8.1 Guidance for Changing One or More Elements of a Method of Evaluation

The definition of 'departure ...' provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are 'conservative' or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods.

Conservative vs. Nonconservative Results

Gaining margin by changing one or more elements of a method of evaluation is considered to be a nonconservative change and thus a departure from a method of evaluation for purposes of 10 CFR 50.59. Such departures require prior NRC approval of the revised method. Analytical results obtained by changing any element of a method are 'conservative' relative to the previous results, if they are closer to design bases limits or safety analyses limits (e.g., applicable acceptance guidelines). For example, a change from 45 psig to 48 psig in the result of a containment peak pressure analysis (with design basis limit of 50 psig) using a revised method of evaluation would be considered a conservative change when applying this criterion. In other words, the revised method is more conservative if it predicts more severe conditions given the same set of inputs. This is because results closer to limiting values are considered conservative in the sense that the new analysis result provides less margin to applicable limits for making potential physical or procedure changes without a license amendment.

In contrast, if the use of a modified method of evaluation resulted in a change in calculated containment peak pressure from 45 psig to 40 psig, this would be a nonconservative change. That is because the change would result in more margin being available (to the design basis limit of 50 psig) for the licensee to make more significant changes to the physical facility or procedures.

Essentially the Same

Licensees may change one or more elements of a method of evaluation such that results move in the nonconservative direction without prior NRC approval, provided the revised result is 'essentially the same' as the previous result. Results are 'essentially the same' if they are within the margin of error for the type of analysis being performed. Variation in results due to routine analysis sensitivities or calculational differences (e.g., rounding errors and use of different computational platforms) would typically be within the analysis margin of error and; thus, considered 'essentially the same.' For example, when a method is applied using a different computational platform (mainframe vs. workstation), results of cases run on the two platforms differed by less than 1%, which is the margin of error for this type of calculation. Thus, the results are essentially the same, and do not constitute a departure from a method that requires prior NRC approval.

The determination of whether a new analysis result would be considered 'essentially the same' as the previous result can be made through benchmarking the revised method to the existing one, or may be apparent from the nature of the differences between the methods. When benchmarking a revised method to determine how it compares to the previous one, the analyses that are done must be for the same set of plant conditions to ensure that the results are comparable. Comparison of analysis methods should consider both the peak values and time behavior of results, and engineering judgment should be applied in determining whether two methods yield results that are essentially the same."

In the case of the PDQ Two-Zone models, the governing topical report documents extensive comparisons of these models to measured data and demonstrates that the Nuclear Reliability Factors (NRFs) documented in Topical Report VEP-FRD-45-A, "Nuclear Design Reliability Factors" remain bounding. Therefore, from a reload analysis perspective, the results with these new tools (elements of the VEP-FRD-42 methodology) are "essentially the same" and implementation via 10 CFR 50.59 is permissible.

NOMAD

Dominion uses the NOMAD 1-D core physics code to perform both reload design analyses and core operation evaluations. Use of this code and its associated model was approved by the NRC on March 4, 1985, with its issuance of Acceptance for Referencing of Licensing Topical Report VEP-NFE-1-A, "The VEPCO NOMAD Code and Model." As stated in VEP-NFE-1-A, verification of and improvements to the NOMAD code and model would continue to be made as more experience was gained in the application of the model to the units at the Surry and North Anna Power Stations. The primary reload safety analysis use of NOMAD is as one of the analytical tools (elements) of the Relaxed Power Distribution Control and Constant Axial Offset Control Methodologies. Use of NOMAD within the framework of those methodologies was not altered by the model update.

Letter 96-319 (Reference 4.4) documented the NOMAD code and model update. These changes were necessitated by the transition to 3-D PDQ (see discussion above). The NOMAD flux solution and axial nodalization were not altered. The updated NOMAD model was qualified against plant data and its fidelity to the data was found to be as good as or better than that of the original code and model. The Nuclear Reliability Factors currently applied in reload analyses were shown to remain appropriate and reload results obtained with the updated model are essentially the same as those obtained with the previous version. As such, the code and model updates do not constitute a change in the approved methodology of VEP-FRD-42 or the Code as described in VEP-NFE-1-A (see the discussion of NEI 96-07, Section 4.3.8, above).

TIP/CECOR

The CECOR code was reviewed and approved generically by the NRC and is documented in CENDP-153-P, Rev. 1-P-A. TIP-CECOR uses the same solution algorithm as CECOR, but is adapted to accept input from movable incore detectors as opposed to fixed detectors. Comparisons with experiments and development of uncertainties for TIP-CECOR are consistent with the CECOR topical report and with VEP-FRD-45-A, the Nuclear Design Reliability Factor topical report.

Additionally, comparisons between TIP/CECOR predictions and those from the previously approved INCORE code revealed that the two codes produce essentially the same results. Therefore, the adoption of TIP/CECOR as a replacement for INCORE represented a change to an element of the reload methodology that can be implemented via 10 CFR 50.59 under the guidance of NEI 96-07. Additionally, qualification of TIP/CECOR for Dominion use met the intent of the programmatic elements of Generic Letter 83-11, Supplement 1, Attachment 1.

RETRAN

Dominion's reload methodology incorporates the RETRAN-02 code. RETRAN-02 was generically approved by the NRC in a 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.

Dominion's RETRAN models and capability were approved by the NRC in a letter from C. O. Thomas (NRC) to W. L. Stewart, Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, "Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

The RETRAN Topical SER recognized that model maintenance activities would be performed under the control of the utility 10 CFR 50 Appendix B QA program. The VEP-FRD-41 SER emphasized that the NRC viewed the primary objective of the report was to demonstrate Dominion's general capability for performing non-LOCA accident analyses:

- "The VEPCO topical report VEP-FRD-41, 'Reactor System Transient Analysis Using the RETRAN Computer Code,' was submitted to demonstrate the capability which VEPCO has developed for performing transient analysis using the RETRAN 01/M0D03 computer code."

- "The staff has reviewed the... VEPCO model descriptions and finds them acceptable for demonstrating understanding of the RETRAN code."

- "Based on the VEPCO RETRAN model and the qualification comparisons ...., the staff concludes that VEPCO has demonstrated their capability to analyze non-LOCA initiated transients and accidents using the RETRAN computer code."

Dominion has demonstrated that use of our models with RETRAN-02 versus RETRAN01 is an equivalent methodology. In a letter (Serial No. 85-753) dated November 19, 1985, Dominion showed that results with RETRAN-02 versus RETRAN-01 were essentially identical except for nonequilibrium pressurizer pressure behavior, where significant improvements were made in the RETRAN-02 solution scheme. This letter requested approval to use RETRAN-02 by February 1986 to support upcoming licensing applications; however, no formal NRC Staff review has been performed to date.

The VEP-FRD-41 SER further stated:

> "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures."

Dominion followed these requirements in updating our RETRAN models. Updated models and the qualification results were documented consistent with our 10 CFR 50 Appendix B, QA program and provided to the NRC for information in letter (Serial No. 93-505) dated August 10, 1993.

It should be noted that the new model results were very similar to those obtained with the old models. No margins in key analysis results were gained. The new models have improved, more mechanistic Doppler reactivity feedback models and more detailed main steam system modeling. This resulted in some changes which were documented and well understood (see Letter 93-505).

While this model upgrade was not a code change, the qualification, documentation and implementation of these new models was done in a manner that meet the programmatic elements of Generic Letter 83-11, Supplement 1.

RETRAN models are code input, and represent an element of Dominion's RETRAN methodology as discussed in NEI 96-07. Because the results obtained with the new models met the "essentially the same" test, we believe that these model upgrades do not represent a change to a method of analysis as defined in 10 CFR 50.59 (c)(2)(viii).

Therefore, VEP-FRD-41A remains the applicable reference for Dominion's approved RETRAN capability.

Response to 4b:

Dominion has used these codes as part of its reload design methodology. However, with respect to the COLR, Dominion notes that the codes above are not listed in the COLR methods reference list in the Technical Specifications, because they do not represent analytical methods that determine core-operating limits. Dominion considers this treatment to be consistent with the guidance in Generic Letter 88-16, which discusses "methodology for determining cycle-specific parameter limits." PDQ and NOMAD represent tools that predict core performance and core parameter values, which are then compared to core operating limits. Similarly, TIP/CECOR processes core surveillance data to confirm that core parameters are behaving as predicted by PDQ and NOMAD and that the operating limits are continuously met. RETRAN provides transient system thermal hydraulic responses that are used in conjunction with the COBRA and LYNXT codes to perform transient DNB calculations for Chapter 15 accidents. The Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) limit in the COLR is established using COBRA and LYNXT in conjunction with the Reactor Core Safety Limits, and not by RETRAN. Similarly the total peaking factor limit (F_Q) in the COLR is established by the referenced, approved LOCA methodology, not by the neutronics codes.

Although VEP-FRD-42, Rev. 1 was not formally revised to reflect changes to these codes and models, it was updated via supplements sent with references 4.3 and 4.4. In neither case was there any NRC request or directive given to revise the topical to incorporate these changes. In particular, Reference 4.3 summarizes several changes relevant to VEP-FRD-42, Rev. 1-A and states:

> "These changes have effectively superseded portions of VEP-FRD-42, Rev. 1-A. Supplement 1 to VEP-FRD-42, Rev. 1-A (enclosed) consolidates and summarizes these changes for your information."

Dominion therefore, considers that these supplements are part of VEP-FRD-42, Rev. 1 and that VEP-FRD-42, Rev. 1 continues to represent Dominion's reload methodology for Westinghouse fuel. It is not Dominion's intention to change our reload methodology as outlined in VEP-FRD-42, Rev. 2 under the provisions of 10 CFR 50.59. However, there are analytical tools, which form elements of the methodology, which can be changed under the provisions of 10 CFR 50.59(c)(2)(viii) as discussed in NEI 96-07 Section 4.3.8.

It is Dominion's intent to apply this guidance of NEI 96-07, Rev. 1, as endorsed by Regulatory Guide 1.187, in determining the applicability of 10 CFR 50.59 to proposed changes to analytical tools which support our reload methodology. The qualification and benchmarking of new elements of the methodology for making this determination will be performed and documented in accordance with the provisions of our quality assurance program.

Response 4c:

The code/model updates discussed in the response to 4a and 4b, above, have been incorporated into VEP-FRD-42, Rev. 2 by referencing the appropriate documentation. Since VEP-FRD-42 is currently referenced in the Technical Specifications no additional changes are necessary.

Response 4d:

A. Production Codes

Core designers and safety analysts have access to a controlled Production Code List.

The Production Code List includes the code version, the effective date, a reference to the applicable code file (which contains the software development, qualification and release documentation), the Code Manager and applicable references documenting the qualification and implementation of the code. This documentation is prepared and peer reviewed in accordance with applicable quality assurance procedures. (The Code Manager is an individual designated by the Department Manager to ensure the required code documentation is completed for new codes and changes to existing codes).

Engineers refer to the List when referencing the name and version of a computer code used to perform design calculations. This procedure ensures that any computer code referenced in a Calculation is available for production work and that the appropriate version of the code is used.

The code version and release date is printed on the output header of all computer calculations. Computer code versions are required to be included as formal references in the engineering calculations which document production applications (e.g., reload calculations).

Dominion software control procedures require that qualified code users be notified when modifications to a code are made.

B. Models

A procedure governs the development and control of Nuclear Analysis and Fuel models. A model is defined as a standardized, controlled set of plant specific input to a computer code. The physical model consists of one or more electronic input files. Models are treated as controlled documents.

Production model input files are write-protected with only authorized personnel given change authority, or monitored in such a way that the Model Manager can determine whether the files have been modified. Model users are responsible for ensuring that the appropriate model is used correctly in an analysis.

Recent changes to applicable production codes and models are discussed as part of the reload design initialization process (see VEP-FRD-42, Rev. 2 Section 3.2.1).

References:

4.1 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report–PDQ Two Zone Model," Serial No. 90-562, October 1, 1990.

4.2 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report Use Pursuant to 10 CFR 50.59," Serial No. 92-713, November 25, 1992.

4.3 Letter from M. L. Bowling (Virginia Electric and Power Company) to U. S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2, Supplement 1 to VEP-FRD-42 Revision 1-A, Reload Nuclear Design Methodology Modifications," Serial No. 93-723, December 3, 1993.

4.4 Letter from S. P. Sarver (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, North Anna Power Station Units 1 & 2, Surry Power Station Units 1 and 2, Supplemental Information for the NOMAD Code and Model, Reload Nuclear Design Methodology, and Relaxed Power Distribution Control Methodology Topical Reports," Serial No. 96-319, November 13, 1996.

Question 5:

VEP-FRD-42, Revision 1 included the code or model used to calculate each of the Key Analysis Parameters within the sections of the report, which discussed each parameter. This is not done in Revision 2. Please provide a listing of the code or model used to calculate each Key Analysis Parameter used in the reload analysis methodology. Does the use of Framatome ANP fuel introduce any new Key Analysis Parameters?

Response:

The models currently used to calculate each parameter are provided below, in terms of the key parameter list from Table 2 of VEP-FRD-42, Revision 2. It was determined that the Framatome ANP fuel required the addition of one key parameter (item 28 below). This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. The code or model currently used to calculate each parameter is listed in the following table. The name PDQ refers to the PDQ two-zone 3D model.

KEY ANALYSIS PARAMETER	CODE OR MODEL
1) Core Thermal Limits (F)	COBRA/LYNXT
2) Moderator Temperature (Density) Coefficient (NS)	PDQ
3) Doppler Temperature Coefficient (NS)	PDQ
4) Doppler Power Coefficient (NS)	PDQ
5) Delayed Neutron Fraction (NS)	PDQ
6) Prompt Neutron Lifetime (NS)	NULIF
7) Boron Worth (NS)	PDQ
8) Control Bank Worth (NS)	PDQ/NOMAD
9) Rod Worth Available for Withdrawal (S)	PDQ/NOMAD
10) Ejected Rod Worth (S)	PDQ/NOMAD
11) Shutdown Margin (NS)	PDQ/NOMAD
12) Boron Concentration for Required Shutdown Margin (NS)	PDQ
13) Reactivity Insertion Rate due to Rod Withdrawal (S)	PDQ/NOMAD
14) Trip Reactivity Shape and Magnitude (NS)	PDQ/NOMAD
15) Power Peaking Factors (S)	PDQ/NOMAD
16) Maximum F_Q * P (S)	PDQ/NOMAD
17) Radial Peaking Factor (S)	PDQ
18) Ejected Rod Hot Channel Factor (S)	PDQ/NOMAD
19) Initial Fuel Temperature (F)	PAD /TACO3
20) Initial Hot Spot Fuel Temperature (F)	PAD /TACO3
21) Fuel Power Census (NS)	PDQ/NOMAD
22) Densification Power Spike (F)	PAD /TACO3
23) Axial Fuel Rod Shrinkage (F)	PAD /TACO3
24) Fuel Rod Internal Gas Pressure (F)	PAD /TACO3
25) Fuel Stored Energy (F)	PAD /TACO3
26) Decay Heat (F)	ANSI ANS-1979 ANSI ANS-1971
27) Maximum Linear Heat Generation Rate (LHGR) (S)	PDQ/NOMAD
28) Maximum LHGR Vs. Burnup (F)	PDQ/NOMAD

Parameter Designation
S: Specific
NS: Non-specific
F: Fuel Performance and Thermal-Hydraulics Related

Question 6:

Regarding Section 2.2.2.1 - Reactivity Coefficients and Defects:

a. Revision 1 discussed a set of four calculations performed to determine temperature and power coefficients at HZP, and an additional four cases to determine the coefficients at power. The Revision 2 methodology includes two cases at ±5°F or ±10°F about the nominal temperature for the temperature coefficients, and two cases at ±5% or ±10% about the nominal power for the power coefficients. Please provide the technical basis supporting this change in methodology.
b. The cases at ±10°F or ±10% were not included in Revision 1 methodology. Please provide the technical basis for these cases.
c. Please discuss the procedures or processes by which the Dominion analyst determines whether to use ±5 or ±10.

Response:

Parts a and b:

Two cases are used for each coefficient. Four cases are still required to determine all three coefficients (ITC, DTC, and MTC). The discussion of HZP coefficients simply reflects the calculation of individual coefficients because all three coefficients are not required at all conditions.

The choice of ±5°F or ±10°F does not have a significant effect on most coefficients (particularly the DTC) because they behave nearly linearly versus temperature over this small a temperature range. Mathematically, as long as the defect is no more complex than a quadratic function of temperature, there is no effect at all in the choice of temperature difference, provided that a centered difference is used. In general, ±5°F is used for all but the DTC. The DTC is always small in magnitude and, therefore, is more susceptible to K-effective convergence tolerance. A range of ±10°F reduces the influence of convergence tolerance. The defining methodology features in the calculation of coefficients are:

1) changing only the variable(s) of interest (fuel temperature, moderator temperature or both, or core power), and
2) the use of a centered difference about the desired point over a range large enough to get a significant change but small enough that the answer still represents the derivative.

As indicated, valid technical reasons may arise which lead to a change in the exact choice of temperature difference or the specific input used to calculate a coefficient. The above discussion also applies to the at-power ITC, DTC, and MTC cases. As in the case of the temperature coefficients, the use of ±10% power for power coefficients does not represent a significant change due to the nearly linear nature of the power coefficients versus power. The primary reason for using ±10% is to minimize 3D-model

THF convergence tolerance on the coefficients. We do not view these specific input changes as changes to the reload methodology.

Part c:

The analyst uses standard techniques described in the core design procedures. These techniques, including the choice of temperature or power change are not changed unless a valid new technical reason arises. A change to the standard technique requires peer review and management approval.

Question 7:

Section 2.3 - Analytical Model and Method Approval Process was added in Revision 2 and discusses the acceptable means by which either analytical models or methods can achieve approved status for use in reload methodology. The first method listed allows reload methodology changes to be implemented in accordance with the provisions of 10 CFR 50.59. The NRC staff does not accept this option as a means to change reload methodology. Implementation under 10 CFR 50.59 would require that new or different methods have already been reviewed and approved by the NRC for the intended application.

Response:

Dominion did not and does not change the reload methodology as outlined in VEP-FRD-42, Rev. 2 under the provisions of 10 CFR 50.59. However, there are analytical tools, which form elements of the methodology, which can be and have been changed under the provisions of 10 CFR 50.59(c)(2)(viii) as discussed in NEI 96-07, Section 4.3.8 (see our response to Question 4, above for further discussion).

The qualification and benchmarking of new or revised inputs or elements of the methodology are performed and documented in accordance with the provisions of our quality assurance program. Dominion then applies the guidance of NEI 96-07, Rev. 1, as endorsed by Regulatory Guide 1.187, in determining the applicability of 10 CFR 50.59 to the proposed changes.

This practice is analogous to that used for previous model updates prior to the issuance of NEI 96-07. For example, application of the 50.59 process to the PDQ model changes (and later the NOMAD and TIP/CECOR changes) was focused on the key issues of whether the change created an unreviewed safety question (USQ), maintaining the "margin of safety," and whether the change involved a change to a Technical Specification. The SER for prior model approvals were reviewed to ascertain the NRC basis for previous approval. In particular, the PDQ Two Zone model was found to be an equivalent replacement of the previous models used for the same purposes inside the existing reload methodology framework and hence the change was determined not to be a USQ. The validation process was at least as broad as for the earlier models, with far more available data. Although the data supported reductions in some uncertainty factors, the existing uncertainty factors were maintained (no reduction in margin of safety). The process used is functionally equivalent to changing elements of the method under the current 50.59 process. This was an internal review process using the same criteria as the original review as described in associated NRC SERs and using appropriate screening techniques under 50.59. Finally, since PDQ was not directly referenced in the COLR, implementation of the model upgrades did not require a change to the Technical Specifications. As discussed in the response to Question 4b, PDQ is not listed among the analytical methods supporting the COLR in Technical Specifications since it is not used to determine values for core operating limits.

The process for qualifying the new RETRAN models was analogous. The qualification tests performed included comparisons between the new and old models as well as to plant transient data. The qualification supported the conclusion that the new models were an equivalent replacement of the transient analysis element of Dominion's reload methodology.

Question 8:

Regarding Section 3.3.2 - Safety Analysis Philosophy, please discuss the procedural or process type of guidance available to the Dominion analyst for determining whether to evaluate or reanalyze a particular transient. This would be important if a key reload parameter value exceeds the current limit in the reference safety analysis, or if the parameter impact is difficult to quantify.

Response:

Quantitative evaluation of a small departure from a parameter limit of parameter limits may be made in one of several ways. First, if the interplay between the various key safety parameters in determining accident response is well defined, margin in one parameter may be used to offset a small departure in another parameter. A second method of quantitative evaluation involves using tradeoffs of known sensitivities. This process is best defined by presenting some examples:

- Studies performed by Dominion and others have shown that a key parameter in determining the severity of the core power response to a rod ejection event is the ejected rod worth in units of dollars (delta k/k ejected rod worth/delayed neutron fraction). For the case of a cycle-specific departure from the minimum delayed neutron fraction, the safety analyst can take advantage of available cycle-specific margin in ejected rod worth by showing that the ejected worth in dollars is less than the worth assumed in the safety analysis.

- For some reload cycles where small departures (a few percent) from an accident specific limit occur, these studies can be used to show that margin in another key parameter that influences the same accident offsets the departure. For example, the end of cycle (EOC) least negative moderator temperature coefficient is a key safety parameter for the rod ejection accident, although its influence is relatively weak. For one recent cycle, a small departure from the limit for this parameter was shown to be offset by large margins in the calculated ejected rod worth, which strongly influences the accident analysis results. These sensitivities are documented in VEP-NFE-2-A.

The general philosophy followed in performing an accident evaluation as opposed to a reanalysis is that the analyst must be able to clearly demonstrate that the results of an analysis performed with cycle-specific input would be less severe than the results of the reference analysis. In other words, in performing the evaluation, no credit is taken for margin between the reference analysis results and the design basis criteria, even though this margin may be substantial. In some cases the analyst and/or reviewer may determine that a cycle specific transient analysis should be performed to verify that the reference analysis remains bounding. No specific quantitative criteria have been established for making this determination, but every instance in which an evaluation (as opposed to a reanalysis) of a key parameter departure is performed must be documented. In the documentation the analyst presents the exact numerical values pertaining to the departure from a limit and a detailed discussion of the reasoning and approach used in reaching a conclusion regarding the parameter in question. This documentation is subject to peer review and approval. The results of these cycle specific evaluations are summarized in the Reload Safety Evaluation (RSE) report.

Question 9:

In Section 3.3.2 - Safety Analysis Philosophy, it is stated that, "The methods that will be employed by Dominion to determine these key parameters will be consistent with the methods documented in References 9, 12, and 14" [of VEP-FRD-42, Revision 2]. References 12 and 14 are Westinghouse WCAP methodologies for reload safety evaluations, and power distribution control and load following procedures. Please discuss the evaluations performed to verify that these methodologies are also applicable for Framatome ANP fuel.

Response:

This section of VEP-FRD-42, Revision 2 defines 3 types of key parameters used to characterize the behavior of reload cores to various postulated accidents. The detailed calculation of specific key parameter values for a reload core is performed using the applicable core design or fuel design tools, dependent upon the parameter involved. The reload safety analysis framework involves evaluating the key parameter values determined for each reload to verify that margin exists between the reload value and the limiting value assumed in the reference safety analysis. This bounding value approach requires the existence of certain predefined relationships that identify the relevant key parameters for a given postulated accident, and their sensitivities (i.e., direction of most limiting effect).

References 9 and 14 of VEP-FRD-42, Revision 2 describe the detailed methodology for defining achievable core power distributions and associated operating limits for two different control schemes employed in Dominion analyses. Reference 9 defines the Dominion-developed Relaxed Power Distribution Control (RPDC) methodology and Reference 14 defines the Westinghouse-developed Constant Axial Offset Control (CAOC) methodology. Each of these methodologies involves the simulation, using detailed nuclear core design codes and models, of a defined number of perturbed core states and the corresponding power distributions. Each of these methodologies is used to determine the limits of normal core operation that will ensure that localized core power distributions remain within the values assumed as initial conditions in the accident analyses. Both methodologies are dependent upon defining proper design input details that characterize the core neutronic behavior. The required design input items involve detailed inputs such as nuclear cross-sections, geometry (fuel pellet, fuel rod and fuel assembly) and enrichment and reactor system inputs such as power, temperature and flowrate. There are several features of the Framatome ANP fuel that differ from the existing fuel design, including: theoretical density, use of Mid-Span Mixing Grids and use of alloy M5. The evaluation of these changes has concluded that each represents alteration of a detailed design input, but not a change that affects the reload methodology. Each of these features of the Advanced Mark-BW fuel was reviewed and found to be within the existing capability and range of applicability of the nuclear core design and safety analysis tools. It was thus concluded that the existing methodologies documented in References 9 and 14 could be used for analysis of the Advanced Mark-BW fuel with its slightly different features.

Reference 12 of VEP-FRD-42, Revision 2 documents the Westinghouse-developed reload evaluation methodology that supports the generic basis for the Dominion reload methodology. The Westinghouse methodology defines specific key parameters for use in accident analyses and their limiting directions for consideration in reload evaluations. Reference 12 is referenced in this sense, in that it defines part of the overall framework that constitutes the Dominion methodology. The changes associated with an alternate fuel design may be of two types: 1) changes that reflect physical fuel design features and 2) changes that reflect licensed analysis approaches or requirements. The Advanced Mark-BW fuel design was assessed for both types of change with respect to applicability of the Reference 12 methodology. It was concluded that none of the physical design features invalidate the key parameter definitions or usage as cited in Reference 12 and VEP-FRD-42, Revision 1. The review associated with potential licensed analysis approaches determined that the Framatome ANP fuel required an additional key parameter, which is reflected in Table 2 of VEP-FRD-42, Revision 2. This parameter, maximum linear heat generation rate versus burnup, is used in the NRC-approved Framatome ANP methodology for cladding stress evaluations. This parameter can be calculated with existing nuclear design codes. This review has demonstrated that the citation of Reference 12 as used within the reload methodology of VEP-FRD-42, Revision 2 is valid for reload evaluation of the Framatome ANP fuel.

Question 10:

Please identify and provide a reference for the fuel lattice physics code used to calculate the prompt neutron lifetime key analysis parameter (Section 3.3.3.5). Include a reference to the NRC staff SER approving this code. Please verify and provide the technical basis for the application of this code to expected fuel designs.

Response:

The lattice code referred to in Section 3.3.3.5 is NULIF, which is the same code used in VEP-FRD-42, Rev. 1. NULIF was originally reviewed as part of VEP-FRD-19A (Ref. 10.1) and the prompt neutron lifetime reliability factor was approved in VEP-FRD-45A (Ref. 10.2). NULIF is a pin cell neutron spectrum / isotopic depletion code. The input to NULIF (i.e., fuel density, fuel enrichment, clad material, fuel pin geometry, soluble boron concentration, depletion power, depletion interval, etc.) for Framatome ANP fuel is not significantly different than for Westinghouse fuel. NULIF is used for both Surry (15x15 lattice) and North Anna (17x17 lattice), and the differences between 15x15 and 17x17 fuel are more significant than the differences between Framatome ANP and Westinghouse fuel.

Reference:

10.1 M. L. Smith, "The PDQ07 Discrete Model," VEP-FRD-19A (July 1981).

10.2 Letter from United States Nuclear Regulatory Commission to Mr. W. N. Thomas, Virginia Electric and Power Company, "Acceptance for Referencing of Topical Report VEP-FRD-45 'Nuclear Design Reliability Factors,' " August 5, 1982.

Question 11:

The dropped RCCA(s) event (dropped rod or dropped bank) is evaluated using the methodology described in Westinghouse WCAP-11394-P-A (Reference 15 of this topical report). Please discuss the evaluation performed to verify that this methodology is also applicable for Framatome ANP fuel.

Response:

The dropped rod methodology of WCAP-11394 requires that three analyses be performed in order to perform an evaluation of the dropped rod event. These analyses, referred to as transient, nuclear, and thermal-hydraulic analyses, provide (1) the statepoints (reactor power, temperature, and pressure), (2) the radial power peaking factor, and (3) the DNB analysis at the conditions determined by items 1 and 2, respectively. These analyses are performed using a parametric approach so that cycle specific conditions may be evaluated using the data generated in the three analyses mentioned above.

Westinghouse, in WCAP-12282 (Reference 11.1), provided generic guidelines that established a common approach for implementation of the revised dropped rod methodology. WCAP-12282 indicated that the core physics correlations and transient statepoints generated for the methodology described in WCAP-11394 apply to all Westinghouse plants with 12 or 14 foot cores. However, due to the plant specific nature of the core physics characteristics and the thermal-hydraulic dropped rod limit lines, a generic safety analysis which bounds all plants is not feasible. Therefore, for every fuel cycle, plant specific data are combined with the appropriate set of correlations and statepoints to verify that the DNB design basis is met for the dropped rod event. The transient statepoints have been generated to be independent of reload considerations. The thermal-hydraulic limit lines are determined on a plant specific basis using currently licensed thermal-hydraulic models. The core physics data required for the analysis are generated during the normal course of the reload design.

The NRC, in Question No. 7 of the request for additional information for WCAP-11394, queried whether the plant/cycle specific calculations are really performed for the items mentioned, or have bounding values been used. The response in WCAP-11394-P-A states that "...the statepoints and R factors are not required to be calculated on a plant or cycle specific basis. Figures IV-1 through IV-8 show the generic applicability of the models used for various fuel types and cycle designs. However, the statepoints and/or R factors would be reassessed for new plants or fuel designs."

As described in WCAP-11394, the transient analysis consists of generating statepoint information (reactor power, temperature, and pressure) for a large number of dropped rod transient events. These statepoints cover a range of reactivity insertion mechanisms for use in the nuclear analysis: the worth of the dropped rod, the moderator temperature coefficient, and the total rod worth available in the control bank which is withdrawn by the Rod Control System when it attempts to restore power to the nominal value. Statepoint data for a large number of transient events, generated by Westinghouse, were used in application of this methodology to North Anna and Surry Power Stations. The statepoint data are influenced by NSSS and protection system features, and were generated to accommodate a wide range of potential core physics conditions. The validity of the statepoint data is, thus, not affected by the transition to Framatome ANP fuel.

The dropped rod methodology employs a bounding empirical correlation between dropped rod worth, $F\Delta H$, and MTC to relate the power change associated with a dropped rod (or rods) to the increase in peaking factor caused by the dropped rod. In order for this correlation to become non-conservative, either the peaking factor change associated with a dropped rod of a particular worth must increase or the power change associated with the dropped rod reactivity insertion must decrease. As indicated in the response to Question 2, the core physics characteristics of the Framatome ANP fuel are nearly identical to the Westinghouse fuel it will replace. There is no change in loading pattern strategy associated with Framatome ANP fuel that would cause a change in the range of dropped rod worth or in the relationship between dropped rod worth and peaking factor increase. Reload cores, therefore, will not respond in a fundamentally different way to the dropped rod event due to the use of Framatome ANP fuel.

The final portion of the dropped rod methodology is the DNB analysis at the conditions determined from the statepoints (reactor power, temperature, and pressure) and the radial power peaking factor. For the DNB analysis, the methodology employs dropped rod limit lines that are representations of the core conditions (inlet temperature, pressure, core power level, and FΔH) for which the DNBR is equal to the DNBR design limit. The dropped rod limit lines for the resident Westinghouse fuel were shown to be applicable for both fuel types.

Therefore, the methodology described in Westinghouse WCAP-11394-P-A is applicable for Framatome ANP fuel.

Reference:

11.1 R. L. Haessler, "Implementation Guidelines for WCAP-11394 (Methodology for the Analysis of the Dropped Rod Event)," WCAP-12282, June 1989

Question 12:

Section 3.5 - Nuclear Design Report, Operator Curves, and Core Follow Data included the following changes to the list of design report reload parameters:

a. Iodine has replaced Samarium worth, and
b. K-effective at refueling conditions as a function of temperature and rod configuration has been removed from the list.

Please provide the technical basis for these changes.

Response:

Part a:

Iodine has not replaced samarium. Iodine has been added to the xenon information. Samarium has been replaced by "Reactivity due to isotopic decay," which includes the contribution of samarium as well as less significant nuclides which build up or decay after shutdown on a time scale similar to samarium.

Part b:

The K-effective for refueling data is now transmitted to the power station prior to issuance of the design report. This was an administrative change to support outage planning and not a change in methodology.

APPENDIX B

RAI Questions and Responses Set # 2

(Sent by Dominion letter Serial No. 02-662 to NRC dated December 2, 2002)

Total Pages : 8

Attachment

REQUEST FOR ADDITIONAL INFORMATION
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT
VEP-FRD-42, Revision 2

North Anna Power Station Units 1 and 2
Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Background

In a letter dated October 8, 2001 (Serial No. 01-628) Virginia Electric and Power Company (Dominion) submitted Revision 2 of VEP-FRD-42, "Reload Nuclear Design Methodology Topical Report," for NRC review and approval. During review of the topical report, the NRC staff identified additional information that is needed to complete their review. The additional information was requested in a letter from the NRC dated October 25, 2002. The requested information is delineated below.

NRC Request for Additional Information:

"VEPCO is requested to confirm that the submittals listed below are the latest revisions for these codes that have not received NRC staff approval.

1. PDQ - The staff will review Topical Report VEP-NAF-1, July, 1990, submitted in a letter from VEPCO to NRC dated October 1, 1990.

2. NOMAD - The staff will review Topical Report VEP-NFE-1A, Supplement 1, September 1996, submitted in a letter from VEPCO to NRC dated November 11, 1996.

3. TIP/CECOR - The staff will review Topical Report VEP-NAF-2, November 1991, submitted in a letter from VEPCO to NRC dated December 20, 1991.

4. RETRAN - The staff will review the information submitted in a letter from VEPCO to NRC dated August 10, 1993. The information provided in this submittal was only applicable for North Anna, Units 1 and 2."

Dominion Response:

PDQ and NOMAD Codes & Models

For PDQ, the report submitted by letter Serial No. 90-562, dated October 1, 1990 is the latest revision that has not received NRC staff approval. Likewise, the NOMAD report submitted by letter Serial No. 96-319, dated November 13, 1996 (versus November 11, 1996 stated above) is the latest revision that has not received NRC staff approval. For both PDQ and NOMAD, the referenced reports are accurate representations of current codes and models with regard to methodology. That is, the theory, sources of input data, solution schemes, geometric mesh structure, energy group structure, and use of the models in the core modeling process have not changed. There have been subsequent code changes to correct minor errors and to accommodate new code edits and additional computing platforms. There have been changes in input to accommodate the evolution of core design features including increased fuel enrichments, changes in BP design, and use of vessel fluence suppression neutron absorber rods. Throughout this period, accuracy of the PDQ model (and by extension the NOMAD model, since PDQ is the source of data and normalization for NOMAD) has been verified each cycle during startup physics testing and during routine core follow. For each cycle, a Startup Physics Test Report and a Core Performance Report is issued to document the behavior of the core relative to the model predictions.

TIP/CECOR Code & Model

The topical VEP-NAF-2, submitted by letter Serial No. 91-746, dated December 20, 1991, is the latest revision of TIP/CECOR that has not received NRC staff approval. However, Dominion does not consider review of TIP/CECOR necessary for review of VEP-FRD-42 Rev. 2 (the Reload Topical) for several reasons. First, the focus of the Reload Topical is on core design and safety analysis methodology, not core surveillance. TIP/CECOR is not directly discussed in VEP-FRD-42 Rev. 2 because it is not part of the reload methodology. TIP/CECOR uses data provided by the PDQ model (Reload Topical Section 2.1.1, paragraph 2) to perform core power distribution surveillance. Second, TIP/CECOR is not new methodology for measurement of core power distributions. USNRC review and approval for use of CECOR in the synthesis of core power distributions using fixed in-core detector data is documented in a 1980 Combustion Engineering Topical Report (Reference 5 of VEP-NAF-2). TIP/CECOR, the Dominion version of the model, uses the same solution schemes and techniques but employs data at 61 axial points rather than just a few. Finally, although the current interpretation of "essentially the same" had not yet been applied to 10CFR50.59 evaluations in 1992, the TIP/CECOR Topical Report and the 10CFR50.59 evaluation performed prior to use of the code clearly demonstrate that TIP/CECOR results are essentially the same as those of the previous measurement code (INCORE). The reason for replacing INCORE with CECOR was not to gain analytical margin, but to be able to accept input representing physically different regions of newer, axially non-homogenous cores.

RETRAN Code & Model

Consistent with approaches employed by NSSS vendors, Dominion's RETRAN model is qualified on the basis of the plant class for which it will be used. There is not a separate Surry-specific RETRAN model document that parallels the content of the report submitted in Reference 1. However, as discussed further below, the material in Reference 1 is equally applicable to the Surry and North Anna models. The Surry 3-loop model, which was completed after the submittal of Reference 1, uses the same noding, modeling philosophy and code options as the North Anna model. The following description provides some background discussion relating to the RETRAN models in use for North Anna and Surry.

Dominion's reload methodology incorporates the RETRAN-02 code, which was generically approved by the NRC via Reference 2. Dominion is currently using RETRAN-02, Mod 5.2. The NRC issued a generic approval, transmitted in Reference 3, for RETRAN-02 Mod 5.0. Discussions between the utilities and the NRC led to the conclusion that Mods 5.1 and 5.2, which were essentially maintenance upgrades, did not require additional NRC review for utility implementation (References 4 and 5).

Dominion's RETRAN models and capability were approved in Reference 6. As noted in the SER, the Virginia Electric and Power Company (Dominion) Topical Report was supplemented in three subsequent submittals (References 7, 8, 9) prepared in response to NRC Requests for Additional Information.

The RETRAN Topical SER (Reference 6) recognized that model maintenance activities would be performed under the utility 10 CFR 50 Appendix B QA program:

> "The staff requires that all future modifications of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures."

Dominion has followed the requirements specified in the SER for VEP-FRD-41 in updating our RETRAN models. Updated models and the qualification results were documented per our 10 CFR 50 Appendix B QA program and provided to the USNRC for information in Reference 1. The qualification, documentation and implementation of these new models was done in a manner that meets the programmatic elements of Generic Letter 83-11, Supplement 1.

Reference 1 presented the 3-loop RETRAN model and qualification results using the North Anna version of the model. The Surry 3-loop model is the same with regard to noding, options and system and component modeling techniques. The Surry and North Anna models differ in order to appropriately reflect plant specific design features such as RCS geometry, system and pump characteristics and setpoint values. Dominion concludes that the model description in Reference 1 accurately describes the key features of the models in use for both Surry and North Anna power stations.

Dominion continues to perform model maintenance activities in accordance with the provisions of the SER and 10 CFR 50 Appendix B. Dominion has made model changes in the past to refine treatment of certain features, to address industry issues or to reflect changes to the plants. These changes were evaluated under the provisions of 10CFR50.59, which will continue to be employed to assess future changes. The following list summarizes several enhancements which are illustrative of the changes that have been made to the models:

- The current models use the 1979 ANS Decay Heat model option.
- More detailed main steam safety valve (MSSV) modeling was added to ensure that the concerns raised in NRC Information Notice 97-09, "Inadequate Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping" are adequately addressed.
- Hydraulic characteristics in the core regions have been adjusted to reflect current fuel assembly designs.
- More detailed, mechanistic models for the pressurizer and steam generator level instrumentation were added.
- A detailed rod control system model was added.

Dominion's Process for the Maintenance and Modification of "NRC Approved" Methodologies

Section 2.3 of VEP-FRD-42, Rev. 2, entitled "Analytical Model and Method Approval Processes," indicates several acceptable means by which either analytical models or methods can achieve approved status for use in Dominion's reload methodology. The following discussion describes Dominion's approach in performing maintenance and modifications of NRC Approved methodologies. This approach is applied to all models and methodologies that are employed in Dominion's reload design methodology, and which may be cited either by reference within VEP-FRD-42 or in the COLR.

The determination of the requirement to submit methodology changes to NRC for approval prior to application is based on published NRC guidance, i.e.:

- Generic Letter 88-16, "Removal Of Cycle-Specific Parameter Limits From Technical Specifications"
- 10 CFR 50.59, and in particular 10 CFR 50.59c(2)(viii): "*(2) A licensee shall obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would (viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.*"
- NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Evaluations"
- Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments" (endorses NEI 96-07 Rev. 1)
- Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses"

Relevant sections of these documents upon which we base our determination process are as follows:

1. Generic Letter 88-16 establishes the concept of reload cycle dependent operating limits in the Technical Specifications.

   "Generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. These changes also allow the NRC staff to trend the values of these limits relative to past experience. This alternative allows continued trending of these limits without the necessity of prior NRC review and approval."

2. NEI 96-07, Rev. 1, as endorsed by Reg. Guide 1.187, provides guidance for evaluating changes to methods under the provisions of 10CFR50.59. For example, Paragraph 4.3.8.1, states:

4.3.8.1, Guidance for Changing One or More Elements of a Method of Evaluation

"The definition of "departure ..." provides licensees with the flexibility to make changes under 10 CFR 50.59 to methods of evaluation whose results are "conservative" or that are not important with respect to the demonstrations of performance that the analyses provide. Changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods."

3. USNRC Generic Letter 83-11 Supplement 1 provides a method for utility qualification of analysis methodologies, including those used to establish core operating limits, without formal NRC review and approval:

   "The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplement to Generic Letter (GL) 83-11 to notify licensees and applicants of modifications to the Office of Nuclear Reactor Regulation (NRR) practice regarding licensee qualification for performing their own safety analyses. This includes the analytical areas of reload physics design, core thermal-hydraulic analysis, fuel mechanical analysis, transient analysis (non-LOCA), dose analysis, setpoint analysis, containment response analysis, criticality analysis, statistical analysis, and Core Operating Limit Report (COLR) parameter generation. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this supplement to the generic letter are not NRC requirements; therefore, no specific action or written response is required."

   "To help shorten the lengthy review and approval process, the NRC has adopted a generic set of guidelines which, if met, would eliminate the need to submit detailed topical reports for NRC review before a licensee could use approved codes and methods. These guidelines are presented in the Attachment to this Generic Letter. Using this approach, which is consistent with the regulatory basis provided by Criteria II and III of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), the licensee would institute a program (such as training, procedures, and benchmarking) that follows the guidelines, and would notify NRC by letter that it has done this and that the documentation is available for NRC audit."

Reflecting this NRC and industry guidance, Dominion's process for maintaining and modifying approved methodologies encompasses these elements:

- Dominion can change, under the provisions of 10 CFR 50.59(c)(2)(viii), NRC approved codes and methodologies used to establish core operating limits, via the processes outlined in NEI 96-07, Rev. 1, without additional NRC review and approval of these changes.

- Dominion can implement or substitute, under 10 CFR 50.59(c)(2)(viii), NRC approved codes and methodologies for use in establishing core operating limits via the processes outlined in Generic Letter 83-11 Supplement 1, without additional NRC review and approval of these methods.

- Dominion concludes that, in updating the list of approved methodologies for establishing core operating limits in the Technical Specifications, utility affirmation that the changes to the methodologies have been done as described by either of the above is adequate to retain the "approved" status for these methods.

References:

1. Letter from M. L. Bowling (Virginia Electric and Power Company) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1&2, Supplemental Information on the RETRAN NSSS Model," Serial No. 93-505, August 10, 1993.

2. 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.

3. Letter from A. C. Thadani (NRC) to W. J. Boatwright (RETRAN02 Maintenance Group), Acceptance for Use of RETRAN02 MOD005.0, November 1, 1991.

4. Letter from M. J. Virgilio (NRC) to C. R. Lehmann (RETRAN Maintenance Group), Acceptance for Referencing of the RETRAN-02 MOD005.1 Code, April 12, 1994.

5. Letter from G. L. Swindlehurst (RETRAN Maintenance Group) to T. E. Collins (NRC/RSB), RETRAN-02 MOD005.2 Code Version, Notification of Code Release, November 24, 1997.

6. Letter from C. O. Thomas (NRC) to W. L. Stewart (Virginia Power), Acceptance for Referencing of Licensing Topical Report VEP-FRD-41, "Virginia Power Reactor System Transient Analyses Using the RETRAN Computer Code," April 11, 1985.

7. Letter from W. L. Stewart (Vepco) to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses, Supplemental Information," Serial No. 060, February 27, 1984.

8. Letter from W. L. Stewart (Vepco) to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses," Serial No. 376, July 12, 1984.

9. Letter from W. L. Stewart (Vepco) to H. R. Denton (USNRC), "Vepco Reactor System Transient Analyses," Serial No. 376A, August 24, 1984.

APPENDIX C

RAI Questions and Responses Set # 3

(Sent by Dominion letter Serial No. 03-183 to NRC dated March 21, 2003)

Total Pages : 56

Attachment 1

Responses to NRC Questions on RETRAN

Virginia Electric and Power Company (Dominion) North Anna and Surry Power Stations

RETRAN Code and Model Review –VEPCO Letter dated August 10, 1993

NRC RETRAN QUESTION 1

1. In the generic RETRAN Safety Evaluation Report (SER), dated September 4, 1984 (Reference 1), the NRC staff approved the use of RETRAN-01/MOD003 and RETRAN-02/MOD002 subject to the limitations and restrictions outlined in the SER. By letter dated April 11, 1985, the NRC staff approved the use of RETRAN-01/MOD003 for VEPCO, although the staff stated in this SER that VEPCO had not provided an input deck to the staff nor had it provided the information needed to address the restrictions listed in the staff SER dated September 4, 1984. The NRC staff's SER dated September 4, 1984, had requested this input deck submittal as a condition of approval to use the RETRAN Code.

   a. VEPCO is currently using RETRAN02/MOD005.2. Please provide information describing how each of the limitations, restrictions, and items identified as requiring additional user justification in the generic staff SERs for RETRAN02/MOD002 through RETRAN02/MOD005.0 (References 1-3) are satisfied for the North Anna and Surry RETRAN models.

   b. As required by the staff SERs (References 1-3), please submit RETRAN input decks that represent the current models and code options used for both North Anna and Surry. For each station, please provide input decks initialized to hot full power and hot zero power conditions in electronic format.

DOMINION RESPONSE TO QUESTION 1a

Dominion responses to the limitations in the RETRAN-02 Safety Evaluation Reports (SERs) in References 1-3 are divided into three sections to distinguish between the different SERs: I) RETRAN02/MOD002; II) RETRAN02/MOD003 and MOD004; and III) RETRAN02/MOD005. The responses are applicable to the North Anna and Surry pressurized water reactor RETRAN models. References for responses to Question 1a are included at the end of the attachment.

I. RETRAN 02/MOD002 Restrictions

The Dominion treatment of each RETRAN limitation from Section II.C in Reference 1 is described. The responses address Limitations a through z, two items on page E2-54 that "require further justification", and eight "implications of the limitations" on page E2-55.

a) Multidimensional neutronic space-time effects cannot be simulated, as the maximum number of dimensions is one. Conservative usage has to be demonstrated.

Dominion Evaluation

The point kinetics approximation is used in the Dominion RETRAN model, consistent with standard industry safety analysis practice. Reactivity effects are modeled using standard fuel and moderator temperature coefficients and control bank worths which are shown to be bounding for Dominion cores using static core physics models which account for full 3-D effects.

Most non-LOCA transients do not involve significant temporal variations in the core power distributions, and industry experience over many years has shown the point kinetics approximation to be valid for this type of accident. Two notable exceptions are the control rod ejection and main steam line break events.

For the control rod ejection event, Dominion uses a point kinetics model to calculate the core average power response. The Doppler feedback is calculated using a spatial power weighting factor that is a function of the radial power peaking factor in the vicinity of the ejected rod, which is calculated using static neutronics calculations. Local power peaking is also calculated via static methods. The power peaking and core average time dependent power responses are then used in conjunction with a conservative hot spot fuel pin model to calculate the limiting local fuel thermal response. Dominion's rod ejection methods have been benchmarked against full 3-D space-time kinetics calculations and shown to be conservative in VEP-NFE-2-A [Reference 4].

Dominion's methodology for steam line break is described in Sections 5.2.3.4 and 5.2.3.5 of VEP-FRD-41-A [Reference 5]. Asymmetric reactivity effects associated with the cold leg temperature imbalance and the assumption of a stuck control rod are modeled by breaking the core into two azimuthal sectors and providing an empirical weighting factor to the moderator temperature coefficients in the two sectors. Fluid mixing between the two regions is modeled based on scale model mixing tests performed by Westinghouse.

Power reactivity feedback is also modeled with an empirical curve of reactivity feedback versus heat flux. The validity of these curves is checked for every reload by static neutronics methods that show that the magnitude of the post-trip return to power predicted by RETRAN is conservatively high. Local power peaking is also calculated using static neutronics methods. Core DNB performance is calculated in a separate code (e.g. COBRA or VIPRE).

This approach for using a combination of point kinetics and static 3-D neutronics calculations for analyzing the steam line break event is similar to that used by fuel vendors (see for example References 6-8).

b) There is no source term in the neutronics models and the maximum number of energy groups is two. The space-time option assumes an initially critical system. Initial conditions with zero fission power cannot be simulated by the kinetics. The neutronic models should not be started from subcritical or with zero fission power without further justification.

Dominion Evaluation

Dominion meets this restriction. Dominion initiates low power events, such as rod withdrawal from subcritical, and the hot zero power rod ejection event from a critical condition with a low initial power level representative of operation within the range of operability for the source range nuclear instrumentation channels. For the "zero power" steam line break, the models are initialized in the same way, and then the design shutdown margin is simulated by a rapid negative reactivity insertion coincident with the break opening.

c) A boron transport model is unavailable. User input models will have to be reviewed on an individual basis.

Dominion Evaluation

A generalized boron transport model was added to RETRAN02/MOD005 [Reference 3]. However, Dominion uses the RETRAN control system to model boron transport in the reactor coolant system for steam line break analyses.

During initial steamline break model development, RETRAN's general transport model was considered but not selected. The primary reason this option was not chosen was that the general transport model uses the default assumption of perfect mixing. Non-mixing regions like pipes cannot be conveniently modeled with a delay-type of behavior. The user may adjust mixing by changing the junction efficiency with a control system. However, this results in just as many control system cards devoted to mixing efficiency calculation as a control block based, full-transport model. Therefore, boron transport is modeled with a control system as in previous analyses. The general modeling philosophy is consistent with that described in Figure III-12 of Reference 19, which was submitted to support the original VEP-FRD-41 review. However, the model in Reference 19 assumed a constant reactor coolant system flow rate. The model was made more robust by incorporating variable transport delays and a dynamic plenum mixing model as described below, so that variable RCS flows are now handled accurately.

The boron transport model is broken into four major parts: 1) Refueling Water Storage Tank (RWST) to Boron Injection Tank (BIT); 2) the BIT; 3) BIT to the Reactor Coolant System (RCS); and 4) the RCS.

<u>BIT Mixing Model</u>
The BIT mixing model begins with the same basic equations as the RCS mixing model. The model makes the approximation that the density of the BIT is constant and is also equal to the density of the incoming fluid.

Following are the mixing region equations:

$$\frac{dC}{dt} = w_i c_i - w_o c_o$$

$$\frac{dC}{dt} = \frac{Mdc}{dt} + \frac{cdM}{dt}$$

$$\frac{dc}{dt} = \frac{w}{M}(c_i - c_o)$$

$$c(t) = \int \frac{dc}{dt} + c_o$$

The first equation states that the rate of change of the mass times the concentration is equal to the mass flow rates in and out times their respective concentrations. The second equation expands the large C derivative into its constituents. The dM/dt term in the second equation is assumed to be zero and w_i is assumed to be equal to w_o. The third equation is formed by combining the first two with dM/dt = 0. The integral of dc/dt provides the dynamic concentration out of the BIT.

By assuming that the density of the BIT and the incoming fluid are equal, the w/M term is equal to the volumetric flow divided by the volume. The equations above are represented with the appropriate control blocks.

BIT to RCS Transport

The transport time through the BIT to RCS piping is calculated in several pieces: the common BIT to SI header delay, and the individual delays from the header to each cold leg. A DIV control block divides the BIT to HDR volume by the total flow rate. The transport time is then used as input to a DLY control block. The same function is performed for each of the header-to-loop segments. The fluid is assumed to be at an initial boron concentration of zero ppm.

RCS Boron Transport

The RCS is broken into several regions for boron transport:

1) the cold leg between the SI point and the vessel (DELAY)
2) the downcomer and lower plenum (MIXING)
3) each core section (DELAY)
4) core bypass (DELAY)
5) the outlet plenum (MIXING)
6) the hot leg, SG tubes, loop seal, RCP, and cold leg between the RCP and SI point. (DELAY)

The model used to represent the transport through each region is noted in parentheses above. The upper head concentration is assumed to be zero for the duration of the transient.

The technique used in each "DELAY" region is as follows:

1) Total "boron flowrate" entering the region is computed by summing the inlet fluid flows times their respective boron concentrations.
2) Total fluid flow entering the region is computed by summing the inlet fluid flows.
3) The total "boron flowrate" is divided by the total fluid flowrate to get a mixed boron concentration.
4) The masses of the volumes in the transport region are summed.
5) The total mass is divided by the total fluid flow to get the transport delay for the region.
6) The mixed boron concentration is propagated to the next region using the transport delay.

The technique used in each "MIXING" region is as follows:

1) The net "boron flowrate" in a region is computed by summing the inlet and outlet fluid flows times their respective boron concentrations.
2) This represents the rate of change of region mass times concentration (dC/dt) which is then integrated to determine C(t).
3) The concentration (c(t)) is then calculated by dividing (C(t)) by the region mass (M).

For the steamline break event, the peak core heat flux is sensitive to the timing of the initial boron increase in the core (i.e., the transport delay from the safety injection system to the core inlet) and is not sensitive to the exact shape of the boron buildup curve. Core inlet boron is only a few ppm at the time of peak heat flux. Dominion's model and vendor models predict comparable times for the introduction of boron to the core as shown in benchmark calculations.

d) Moving control rod banks are assumed to travel together. The BWR plant qualification work shows that this is an acceptable approximation.

Dominion Evaluation

Control rod motion in the Dominion RETRAN point kinetics models is simulated by a reactivity input calculated from a time-dependent control bank position and a function generator containing integral bank worth versus position. For cases with automatic rod control simulated, the bank worth model is typically associated with the D-control bank only, which is the only bank in the core at or near full power.

For cases with reactor trip, the integral worth assumed is that associated with all control and shutdown banks at the power dependent insertion limit, less the most reactive control assembly in the core, which is assumed not to insert. The shape of the integral worth curve is based on a conservative bottom-skewed power distribution which delays the reactivity effects. This integral worth curve is checked for every reload core.

e) The metal-water heat generation model is for slab geometry. The reaction rate is therefore underpredicted for cylindrical cladding. Justification will have to be provided for specific analyses.

Dominion Evaluation

The rod ejection accident is the only non-LOCA transient analyzed with RETRAN where the metal-water reaction is applied. Dominion's RETRAN hot pin model was benchmarked against a similar vendor model and produced consistent temperature transients for consistent transient pin powers. These results are discussed in Reference 4, which documents Dominion's rod ejection methodology in its entirety.

f) Equilibrium thermodynamics is assumed for the thermal hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Dominion Evaluation

The current version of RETRAN-02 in use at Dominion (MOD005.2) allows for multiple nonequilibrium volumes. In Dominion RETRAN models, the nonequilibrium region option is generally only used for the pressurizer, except when applied to the reactor vessel upper head in main steamline break analyses. Toward the end of the transient, the upper head, which has experienced drainage, flashing and phase separation during the cooldown, will begin to refill due to continued operation of safety injection. An equilibrium model in the head can produce nonphysical pressure oscillations. While this phenomenon generally occurs beyond the time of interest for evaluating core performance, the nonphysical behavior is avoided by using a nonequilibrium model in the upper head. This is physically reasonable for the head geometry and the limited hydraulic communication between the head and the upper plenum.

Section 5.3.3 of VEP-FRD-41-A presented comparisons of RETRAN pressure predictions to plant data for a cooldown and safety injection transient at North Anna. The nonequilibrium pressurizer model response was in good agreement with the observed plant response.

g) While the vector momentum model allows the simulation of some vector momentum flux effects in complex geometry the thermal hydraulics are basically one-dimensional.

Dominion Evaluation

Dominion RETRAN models do not currently use the vector momentum option. As discussed in the response to Limitation A, incomplete fluid mixing between loops is modeled for steam line break based on the Indian Point 1/7 scale model mixing tests performed by Westinghouse. This is done by dividing the downcomer into two azimuthal sectors and specifying cross-flow junctions between the cold legs and downcomer sectors with form-loss coefficients to give the proper steady state mixing flows.

h1) Further justification is required for the use of the homogeneous slip option with BWRs.

Dominion Evaluation

This limitation is not applicable to Dominion PWR RETRAN models.

h2) The drift flux correlation used was originally calibrated to BWR situations and the qualification work for both this option and for the dynamic slip option only cover BWRs. The drift flux option can be approved for BWR bundle geometry if the conditions of (n2) are met.

Dominion Evaluation

Dominion RETRAN models specify the use of the dynamic slip option on the primary side and zero slip on the secondary side of the steam generator (SG) tubes. However, two-phase flow is not normally encountered on the primary side during non-LOCA PWR transients. The exception is for steam line break, where the pressurizer may drain during the cooldown, and the upper head may flash, resulting in some carryunder to the upper plenum region as the head drains. The RCS pressure response obtained in Dominion steam line break analyses, including the effects of pressurizer and upper head flashing and drainage, is consistent with that obtained by vendor models as discussed in VEP-FRD-41-A.

Dominion does have a multi-node steam generator secondary model overlay that uses dynamic slip modeling. This model is not used in licensing calculations, but it is occasionally used in studies to confirm that the standard steam generator models are providing conservative results. The standard model features involve a single-node secondary side model and the associated heat transfer response and level-versus inventory correlations that are used to model low and low-low

SG level reactor protection. The multi-node model treats the horizontal flow between the lower downcomer and tube bundle as bubbly flow.

Reference 9 presented comparisons between the multi-node and single-node SG versions of the model for a complete loss of load and for a 200%/minute turbine runback transient at full power. The response comparisons for pressurizer pressure and liquid volumes, RCS temperature, and steam pressure showed essentially identical responses for the two models. The most pronounced differences were in predicted changes in steam generator level and inventory, as expected.

i) **The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these options refer to (n3).**

Dominion Evaluation

Refer to the response to Limitation h2.

j) **Only one dimensional heat conduction is modeled. The use of the optional gap linear thermal expansion model requires further justification.**

Dominion Evaluation

The core conductor model in Dominion RETRAN system models does not use the gap expansion model. Dominion's hot spot model for calculating the hot pin thermal transient in rod ejection analyses models rapid gap closure following the ejection with an essentially infinite gap thermal conductivity, as described in Reference 4. Qualification comparisons of the hot spot model to vendor calculations are presented in Section 4.3.2 of Reference 4.

k) **Air is assumed to be an ideal gas with a constant specific heat representative of that at containment conditions. It is restricted to separated and single phase vapor volumes. There are no other non-condensables.**

Dominion Evaluation

Dominion PWR RETRAN models do not use air.

l) **The use of the water properties polynomials should be restricted to the subcritical region. Further justification is required for other regions.**

Dominion Evaluation

Dominion models have not been applied in the supercritical region. Dominion notes that this restriction has been substantially reduced for RETRAN-3D [Reference 10], and the NRC staff has approved RETRAN-3D for ATWS analysis, with a caution for evaluating calculations in the region of enthalpy > 820 Btu/lbm and pressures between 3200 and 4200 psia. Dominion has not yet formally implemented RETRAN-3D nor applied it to ATWS analyses.

Also note that the design basis for the ATWS Mitigation System Actuation Circuitry (AMSAC) for Westinghouse PWRs is to limit the maximum RCS pressure to less than 3200 psig [Reference 11]. Therefore, analytical results which yield supercritical conditions in the RCS are not anticipated for Dominion's nuclear units.

m) A number of regime dependent minimum and maximum heat fluxes are hardwired. The use of the heat transfer correlations should be restricted to situations where the pre-CHF heat transfer or single-phase heat transfer dominates.

Dominion Evaluation

Dominion PWR RETRAN system models use heat transfer correlations in three areas:

- Reactor core conductors
- Primary (RCS) side of the steam generator tubes
- Secondary (steam) side of the steam generator tubes

For all non-LOCA accident analyses, the core heat transfer remains in the single-phase convection and subcooled nucleate boiling regions. The event that presents the most severe challenge to subcooled nucleate boiling on a corewide basis is the locked reactor coolant pump rotor event presented in Sections 15.4.4 and 14.2.9.2 of the North Anna and Surry UFSARs, respectively. For the locked rotor event, the heat transfer mode remains subcooled forced convection at the core inlet node and nucleate boiling at the mid core and top core node throughout the event.

Similarly, subcooled forced convection is the dominant heat transfer mode on the inside of the steam generator tubes for all non-LOCA events.

On the secondary (steam) side of the steam generator tubes, the heat transfer mode is typically saturated nucleate boiling (Mode 2) for non-LOCA transients. Exceptions occur when:

- a steam generator approaches dryout, such as for the North Anna feedline break accident
- a steam generator blows down, as in the main steam line break event.
- there is no flow through the single-node secondary side of the steam generator, such as during a loss of load (turbine trip) with feedline isolation.

These cases will be addressed in turn.

For cases where significant steam generator dryout is anticipated, Dominion uses the RETRAN local conditions heat transfer option in conjunction with the single-node steam generator secondary side model. Dominion has performed analyses to evaluate the physical realism of the modeling results, including a steam generator tube noding sensitivity study. The behavior of the model is such that nucleate boiling heat transfer (RETRAN Mode 2) is predicted for nodes below the collapsed liquid level. For nodes above the collapsed level, the model predicts a rapid transition from single-phase convection to steam (RETRAN Mode 8).

For the steam line break calculation, Dominion uses a set of overlay cards to predict a conservatively large heat transfer coefficient on the secondary side, in order to maximize the RCS cooldown. This is done using control blocks.

For nodes below the collapsed liquid level, the overlay model applies a separate heat transfer coefficient to the secondary side of each steam generator conductor based on the maximum of the following, independent of which regime the RETRAN logic would pick:

- Rohsenow pool boiling
- Schrock-Grossman forced convection vaporization
- Thom nucleate boiling
- Chen combined nucleate boiling and forced convection vaporization
- Single phase conduction to steam (Dittus-Boelter)

This maximum coefficient represents the heat transfer for the "wet" heat transfer surface in the steam generator.

To better represent the variation of the film coefficient for the conductors at different elevations, a model was developed to calculate a collapsed liquid level and apply the maximum "wet" coefficient below this level and the forced convection to steam above this level. This provides a realistic and smooth transition in heat transfer capability as the steam generator inventory is depleted.

For cases with no flow calculated through the single-node secondary side (e.g., turbine trip with no condenser dumps and assumed feedwater line isolation at the time of turbine trip), the heat transfer on the entire secondary surface of the tubes will rapidly transition to forced convection vaporization with a very small heat transfer coefficient. This behavior is non-physical, because a significant portion of the tube bundle remains covered with two-phase mixture and would remain in the nucleate boiling regime. However, the results are conservative and Dominion's experience has been that this calculational anomaly only occurs for brief periods of time such that the key results (e.g., peak RCS pressure) are not significantly impacted.

In summary, the limitations of RETRAN's regime-dependent heat transfer models are considered in Dominion licensing analyses. Appropriate assumptions and approximations are made to ensure that the accident analyses are conservative.

n1) The Bennett flow map should be used for vertical flow within the conditions of the database and the Beattie two-phase multiplier option requires qualification work.

Dominion Evaluation

Dominion RETRAN models are not used for conditions involving two-phase horizontal flow. The models use the RETRAN application of Baroczy's correlation for two-phase friction effects, as opposed to Beattie. For steam generator tube rupture calculations, break flow is calculated using a junction loss coefficient computed from Blasius' smooth tube frictional pressure drop assuming single-phase flow. This model overpredicts the actual observed break flow in the 1987 North Anna Unit 1 double-ended rupture.

n2) No separate effects comparisons have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests (5) before the approval of the algebraic slip option.

Dominion Evaluation

Dominion RETRAN models specify the use of the dynamic slip option on the primary side and zero slip on the secondary side. Refer to the response to Limitation h2.

n3) While FRIGG tests comparisons have been presented for the dynamic slip option the issues concerning the Shrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required.

Dominion Evaluation

Refer to the response to Limitation h2.

o) The nonequilibrium pressurizer model has no fluid boundary heat losses, cannot treat thermal stratification in the liquid region and assumes instantaneous spray effectiveness and a constant rainout velocity. A constant L/A is used and flow detail within the component cannot be simulated. There will be a numerical drift in energy due to the inconsistency between the two regions and the mixture energy equations but it should be small. No comparisons were presented involving a full or empty pressurizer. Specific application of this model should justify the lack of fluid boundary heat transfer on a conservative basis.

Dominion Evaluation

VEP-FRD-41-A [Reference 5] describes that the Dominion RETRAN pressurizer model uses the non-equilibrium model to ensure accurate modeling of transient conditions that may involve a surge of subcooled liquid into the pressurizer or to ensure appropriate treatment of pressurizer spray and heaters. While a wall heat transfer model, including vapor condensation, was added in version MOD003 [Reference 2], Dominion continues to model the non-equilibrium volume walls as an adiabatic surface.

The North Anna Unit 2 Natural Circulation Tests conducted in July 1980 measured the effect of convective heat losses from the pressurizer with all heaters secured. The observed effect was about 5 F/hr liquid temperature cooldown and about 38 psi/hr pressure loss [Reference 12]. The significant plant response for UFSAR non-LOCA transients occurs within the first 30 minutes of the event initiator. Therefore, pressurizer wall heat transfer is a phenomenon that is not significant over the time frame of interest for UFSAR non-LOCA analyses.

Section 5.3.3 of VEP-FRD-41-A includes a RETRAN simulation of a North Anna cooldown event, demonstrating the adequacy of the RETRAN pressurizer modeling assumptions compared to actual plant response. Both the observed data and the model indicated that level indication was lost for a brief portion of the transient. Overall, the RETRAN prediction of pressurizer pressure and level indicate that the non-equilibrium pressurizer model adequately describes the behavior for large swings in pressure and level. In addition, the model predicted the time when level indication was lost close to the observed data. Therefore, the RETRAN non-equilibrium pressurizer model is able to perform accurate predictions of a draining pressurizer.

Reference 9 included a RETRAN simulation comparison to the 1987 North Anna steam generator tube rupture event. Figures 71 and 72 demonstrate that the RETRAN non-equilibrium pressurizer model provides good predictions of pressure and level behavior over a wide range of actual accident conditions. The model closely predicted the pressurizer level recovery near 1700 seconds.

RETRAN has been used to analyze the North Anna main feedwater line break (MFLB) UFSAR event, which reaches a pressurizer fill condition. The RETRAN analysis was benchmarked to the licensed LOFTRAN analysis and showed good agreement for pressurizer pressure and water volume. The codes predicted similar times for the pressurizer to reach a fill condition and similar RCS conditions long-term after the pressurizer is filled. Dominion RETRAN simulations for the MFLB event do not exhibit any unusual pressurizer behavior or numerical discontinuities when the pressurizer fills and remains filled.

The results of RETRAN comparisons to plant operational data in References 5 and 9 and to other licensed transient analysis codes demonstrate that the non-equilibrium pressurizer model is adequate over the expected range of pressurizer conditions that occur in North Anna and Surry UFSAR non-LOCA events analyzed with RETRAN.

p) **The nonmechanistic separator model assumes quasi-statics (time constant - few tenths seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of the default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrants is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific application to BWR pressurization transients under those conditions should be justified.**

Dominion Evaluation

The non-mechanistic separator model is not applied in Dominion PWR RETRAN models.

q) **The centrifugal pump head is divided equally between the two junctions of the pump volume. Bingham pump and Westinghouse pump data are used for the default single phase homologous curves. The SEMISCALE MOD-1 pump and Westinghouse Canada data are used for the degradation multiplier approach in the two phase regime. Use of the default curves has to be justified for specific applications. Pump simulation should be restricted to single phase conditions.**

Dominion Evaluation

VEP-FRD-41-A describes that the plant-specific pump head vs. flow response for first quadrant operation is used in the Dominion RETRAN models. The homologous curves in the model represent single-phase conditions. The RETRAN default curves are not used. The pump coastdown verifications in Section 5.3 of VEP-FRD-41-A demonstrate the adequacy of the centrifugal reactor coolant pump model versus plant-specific operational test data. Changes to the RCP coastdown model were made in Reference 9 to provide conservative coastdown flow predictions for loss of flow events relative to the actual coastdown measured at the plant. The latest Westinghouse locked rotor/sheared shaft coefficients have also been implemented.

r) **The jet pump model should be restricted to the forward flow quadrant, as the treatment in the other quadrants is conceptually not well founded. Specific modeling of the pumps in terms of volumes and junction is at the user's discretion and should therefore be reviewed with the specific application.**

Dominion Evaluation

The jet pump model is not applied in Dominion PWR RETRAN models.

s) **The nonmechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant L/A, and a pressure behavior dictated by a constant loss coefficient. It should only be used for quasistatic conditions and in the normal operating quadrant.**

Dominion Evaluation

The non-mechanistic turbine model is not applied in Dominion PWR RETRAN models.

t) **The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendation (4) for the bubble departure point. It is used only for the void reactivity computation and has no direct effect on the thermal hydraulics. Comparisons have only been presented for BWR situations. The model should be restricted to the conditions of the qualification database. Sensitivity studies should be requested for specific applications. The profile blending algorithm used will be reviewed when submitted as part of the new manual (MOD03) modifications.**

Dominion Evaluation

The Dominion PWR RETRAN models do not use the subcooled void model to calculate the neutronic feedback from subcooled boiling region voids. Dominion models use a moderator temperature coefficient except for the steamline break event, which applies an empirical curve of reactivity feedback versus core average power. This curve is validated as conservative on a reload basis using static, 3-D, full-core neutronics calculations with Dominion's physics models [Reference 15]. Dominion experience has indicated that the calculated DNBR's for the limiting steamline break statepoints show a weak sensitivity to the effects of void reactivity. The profile blending algorithm approved for RETRAN-02 MOD003 resolved this limitation [Reference 10, page 29].

u) **The bubble rise model assumes a linear void profile; a constant rise velocity (but adjustable through the control system); a constant L/A; thermodynamic equilibrium and makes no attempt to mitigate layering effects. The bubble mass equation assumes**

zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified.

Dominion Evaluation

Dominion PWR RETRAN models use bubble rise in the pressurizer, reactor vessel upper head, and steam generator dome regions [Reference 9, Table 1].

The upper head applies the bubble rise model to provide complete phase separation to account conservatively for upper head flashing during a main steam line break (MSLB). Complete separation ensures that only liquid will be delivered to the upper plenum during transients that exhibit upper head flashing. The effect of upper head flashing is seen in the abrupt change in slope in the reactor coolant system pressure following a MSLB. Dominion's RETRAN model predicts results that are similar to the licensed FSAR MSLB analysis in VEP-FRD-41-A (Figure 5.47).

The single-node steam generator secondary model is initialized with a low mixture quality so that the steady-state initialization scheme selects a large bubble rise velocity. The initialization models complete phase separation as a surrogate for the operation of the mechanical steam separators and dryers in the steam generators.

The pressurizer model applies the maximum bubble density at the interface between the mixture and vapor region. The use of the bubble rise model in the pressurizer has been qualified against licensed transient analysis codes and plant operational data as follows:

- VEP-FRD-41-A RETRAN analyses show pressurizer conditions similar to the vendor FSAR analyses for several accidents: uncontrolled rod withdrawal at power, loss of load event, main steamline break, and excessive heat removal due to feedwater system malfunction.
- VEP-FRD-41-A, Section 5.3.3, RETRAN simulations show good agreement with pressurizer response operational data from the 1978 North Anna cooldown transient.
- Reference 9 RETRAN simulations show good agreement of transient pressurizer conditions compared to the 1987 North Anna Unit 1 steam generator tube rupture event.

Implicit in the agreement between plant operational data and RETRAN is that the bubble rise model accurately predicts conditions in the pressurizer over a wide range of temperature, pressure, and level transient conditions. Therefore, Dominion has justified appropriate use of the bubble rise model through adequate benchmarking against physical data and other licensed transient analysis codes.

v) The transport delay model should be restricted to situations with a dominant flow direction.

Dominion Evaluation

Dominion RETRAN models use the enthalpy transport delay model in the reactor coolant system piping and core bypass volume, where a dominant flow direction is expected. Flow reversal is not normally encountered in these volumes during non-LOCA accident analyses. For accidents that produce a flow reversal or flow stoppage, the analyst may use the transport delay model if it adds conservatism to the results (e.g., if RCS pressure is higher during a locked rotor event with the model activated).

w) **The stand alone auxiliary DNBR model is very approximate and is limited to solving a one-dimensional steady state simplified HEM energy equation. It should be restricted to indicating trends.**

Dominion Evaluation

Dominion PWR RETRAN models do not employ the auxiliary DNBR model.

x) **Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model. For heat addition with multidirectional, multijunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MOD03) modifications.**

Dominion Evaluation

Dominion PWR RETRAN models do not use the enthalpy transport model in separated volumes. The enthalpy transport model is used only for the reactor core and the steam generator tubes primary side. The restriction is met.

y) **The local conditions heat transfer model assumes saturated fluid conditions, one-dimensional heat conduction and a linear void profile. If the heat transfer is from a local conditions volume to another fluid volume, that fluid volume should be restricted to a nonseparated volume. There is no qualification work for this model and its use will therefore require further justification.**

Dominion Evaluation

As discussed in the response to Limitation m, Dominion restricts use of the local conditions heat transfer model to loss of secondary heat sink events. The model predicts a rapid transition from nucleate boiling to single-phase convection to steam on the secondary side as the tube bundle dries out.

Nodal sensitivity studies were performed to show that the default tube bundle noding provides an adequate representation of the primary to secondary heat transfer. The single-node secondary side is initialized with a low mixture quality. As a result, a high bubble rise velocity is calculated by the steady state initialization routine. This drives the RETRAN calculated mixture level to the collapsed liquid level and conservatively maximizes the rate of tube bundle uncovery as the inventory is depleted. The fluid condition on the inside of the tubes remains single phase, and thus the restriction is met.

z) **The initializer does not absolutely eliminate all ill-posed data and could have differences with the algorithm used for transient calculations. A null transient computation is recommended. A heat transfer surface area adjustment is made and biases are added to feedwater inlet enthalpies in order to satisfy the steady state heat balances. These adjustments should be reviewed on a specific application basis.**

Dominion Evaluation

Dominion's RETRAN user guidelines contain appropriate guidance and cautions about the potential impact of the feedwater enthalpy bias term on transient results. The guidance for initializing the models for other than the default conditions instructs the user to run a null transient and check the results for a stable solution, and to check the calculated heat transfer area on the steam generators to ensure that primary and secondary side conditions are properly matched.

<u>Technical Evaluation Report (TER) "Items Requiring Further Justification"</u>

The RETRAN-02/MOD002 TER, page E2-54, includes two items that require further justification for PWR systems analysis. Dominion responses to these items are provided below.

i) **Justification of the extrapolation of the FRIGG data or other data to secondary side conditions for PWRs should be provided. Transient analyses of the secondary side must be substantiated. For any transient in which two-phase flow is encountered in the primary, all the two-phase flow models must be justified.**

Dominion Evaluation

These restrictions were addressed in the evaluations for Limitations h2, m, n1, u, x, and y.

ii) **The pressurizer model requires qualification work for the situations where the pressurizer either goes solid or completely empties.**

Dominion Evaluation

Refer to the response to Limitation o. Dominion has shown that the non-equilibrium pressurizer model is adequate over the expected range of pressurizer conditions that occur in North Anna and Surry UFSAR non-LOCA events analyzed with RETRAN. Specifically,

- The UFSAR main steam line break events analyzed with RETRAN show a response for a drained pressurizer that is consistent with vendor methods [Reference 5, Figure 5.47].
- The North Anna UFSAR main feedline break event (case with offsite power available), which results in a filled pressurizer, shows a response that is consistent with vendor results.
- Comparisons to the North Anna Cooldown Transient [Reference 5, Section 5.3.3] and Steam Generator Tube Rupture [Reference 9, Section 3.2] shows reasonable agreement with plant data for the case of pressurizer drain and subsequent refill.

<u>Technical Evaluation Report "Implications of these Limitations"</u>

The RETRAN-02/MOD002 TER includes "implications of these limitations" on page E2-55. Dominion responses to the eight implications are provided.

i) Transients which involve 3-D space time effects such as rod ejection transients would have to be justified on a conservative basis.

Dominion Evaluation

See the response to Limitation a and Topical Report VEP-NFE-2-A.

ii) Transients from subcritical, such as those associated with reactivity anomalies, should not be run.

Dominion Evaluation

See the response to Limitation b.

iii) Transients where boron injection is important will require separate justification for the user specified boron transport model.

Dominion Evaluation

See the response to Limitation c.

iv) For transients where mixing and cross flow are important the use of various cross flow loss coefficients have to be justified on a conservative basis.

Dominion Evaluation

See the responses to Limitations a and g.

v) ATWS events will require additional submittals.

Dominion Evaluation

See the response to Limitation l.

vi) For PWR transients where the pressurizer goes solid or completely drains the pressurizer behavior will require comparison against real plant or appropriate experimental behavior.

Dominion Evaluation

See the response to Limitation o and "Item For Additional Justification Item ii". Dominion notes that the RETRAN 3-D pressurizer model has been explicitly approved for filling and draining events [Reference 10].

vii) PWR transients, such as steam generator tube rupture, should not be analyzed for two-phase conditions beyond the point where significant voiding occurs on the primary side.

Dominion Evaluation

Dominion meets this restriction with the exception of the main steam line break event analysis, which produces a limited amount of flashing in the stagnant upper head volume. Refer to Dominion's Evaluation of Limitations F and U for justification of the use of the bubble rise model with complete phase separation for the upper head volume in the reactor coolant system.

viii) BWR transients where asymmetry leads to reverse jet pump flow, such as the one recirculation pump trip, should be avoided.

Dominion Evaluation

This caution does not apply to Dominion PWR RETRAN models.

II. RETRAN 02/MOD003-004 Restrictions

Section 3.0 of Reference 2 presents six restrictions for RETRAN02/MOD003 and MOD004 code versions. The Dominion evaluation for each is provided.

1. The RETRAN code is a generically flexible computer code requiring the users to develop their own nodalization and select from optional models in order to represent the plant and transients being examined. Thus, as specified in the original SER (Ref. 1), RETRAN users should include a discussion in their submittals as to why the specific nodalization scheme and optional models chosen are adequate. These should be performed on a transient by transient basis.

Dominion Evaluation

VEP-FRD-41-A documents the NRC-approved RETRAN analysis methodology employed by Dominion. The topical report included 1-loop and 2-loop RETRAN models, their nodalization schemes, and specific comparisons to licensed FSAR analyses and to plant operational events. Reference 9 notified the NRC of modifications to the RETRAN models, including development of a 3-loop model and the primary and secondary systems nodalization schemes. The Dominion 3-loop models include discrete noding for every major geometry feature in the reactor coolant system. The steam generator secondary model is a lumped volume; Dominion experience has confirmed the adequacy and conservative nature of this model.

Analyses from the qualification set were provided in References 5 and 9 to demonstrate the adequacy and conservatism of the model nodalization and selection of model options. Dominion meets the NRC SER restrictions and has justified the model options over the range of conditions expected for non-LOCA transients for North Anna and Surry. The RETRAN user manual and training describe the limitations for the selected optional models to ensure appropriate use within the qualified range of application.

Dominion has qualified its RETRAN models against plant operational data and other licensed transient analysis codes sufficiently to justify the nodalization schemes and the model options that are used for non-LOCA transients analyzed with RETRAN.

2. **Restrictions imposed on the use of RETRAN02 models (including the separator model, boron transport, jet pump and range of applicability, etc.) in the original SER (Ref. 1) have not been addressed in the GPU submittal and therefore remain in force for both MOD003 and MOD004.**

Dominion Evaluation

Dominion treatment of the RETRAN02/MOD002 SER restrictions is provided earlier in this attachment.

3. **The countercurrent flow logic was modified, but continues to use the constitutive equations for bubbly flow; i.e., the code does not contain constitutive models for stratified flow. Therefore, use of the hydrodynamic models for any transient which involves a flow regime which would not be reasonably expected to be in bubbly flow will require additional justification.**

Dominion Evaluation

Refer to the response to RETRAN02/MOD002 SER Limitation h2.

4. **Certain changes were made in the momentum mixing for use in the jet pump model. These changes are acceptable. However, those limitations on the use of the jet pump momentum mixing model which are stated in the original SER (Ref. 1) remain in force.**

Dominion Evaluation

Dominion PWR RETRAN models do not use jet pump models.

5. **If licensees choose to use MOD004 for transient analysis, the conservatism of the heat transfer model for metal walls in non-equilibrium volumes should be demonstrated in their plant specific submittals.**

Dominion Evaluation

Dominion RETRAN models do not use the wall heat transfer model for non-equilibrium volumes. Dominion RETRAN comparisons to plant transients show that adiabatic modeling of the pressurizer walls is adequate (see response to RETRAN02/MOD002 SER Limitation o).

6. **The default Courant time step control for the implicit numerical solution scheme was modified to 0.3. No guidance is given to the user in use of default value or any other values. In the plant specific submittals, the licensees should justify the adequacy of the selected value for the Courant parameter.**

Dominion Evaluation

Dominion RETRAN models use the iterative solution technique. This technique allows the results of the time advancement to be evaluated before the solution is accepted. If a converged solution is not achieved in a given number of iterations, the time advancement can be reevaluated with a smaller time step. The Courant limit default value of 0.3 is applied in Dominion models.

The default value limits the time step size to less than 1/3 of the time interval required for the fluid to traverse the most limiting (i.e. fastest sweep time) control volume in the system. This is considered a very robust method for ensuring that the Courant limit is not exceeded.

Dominion user guidelines require that time step studies be performed for each new RETRAN analysis to ensure that a converged numerical solution is reached. This practice eliminates the impact of variations in the selected Courant limit input constant.

III. RETRAN 02/MOD005.0 Restrictions

The Dominion treatment of each limitation from Reference 3, Section 4.0, is described.

1. **The user must justify, for each transient in which the general transport model is used, the selected degree of mixing with considerations as discussed in Section 2.1 of this SER.**

Dominion Evaluation

Dominion does not use the general transport model. A description of the Dominion boron transport modeling for steamline break analyses is provided in the response to Limitation c in Section I.

2. **The user must justify, for each use of the ANS 1979 standard decay heat model, the associated parameter inputs, as discussed in Section 2.2 of this SER.**

Dominion Evaluation

Section 2.2 of the RETRAN-02 MOD005.0 SER specifies the following parameter inputs:

a. power history
b. fission fraction
c. energy per fission of each isotope
d. neutron capture in fission products by use of a multiplier
e. production rate of 239 isotopes
f. activation decay heat other than 239
g. delayed fission kinetic modeling
h. uncertainty parameters

The Dominion RETRAN models use the following assumptions in the calculation of decay heat:

- An operating period of 1,500 days with a load factor of 100% is input to the Dominion RETRAN models.
- The model assumes 190 MeV/fission. The reduction of the Q value to 190 MeV/fission from the default RETRAN value of 200 MeV/fission is conservative since, in the 1979 ANS Standard, decay heat power is inversely proportional to Q.
- There is no neutron capture component.
- Decay heat fissioning is solely from U-235. The assumption that all decay heat is produced from U-235 fissioning nuclides is conservative.
- The RETRAN actinide correlation is that of Branch Technical Position APCSB9-2 [References 17 and 18]. The RETRAN input of the breeding ratio UDUF (i.e., the number of Pu-239 atoms produced per U-235 atoms fissioned) is 0.77 and only impacts the calculation of the actinide contribution. The greater the value of UDUF, the higher the predicted decay heat fraction.
- A value of 1.0 is input for the RETRAN model for the decay heat multiplier.

The results of a RETRAN calculation with the 1979 decay heat model and the assumptions listed above were compared to a vendor calculated decay heat curve based on the 1979 ANS standard with 2-sigma uncertainty added. The results indicated that the decay heat fraction calculated with RETRAN is higher than the vendor calculated decay heat. Therefore, the Dominion application of the ANS 1979 standard decay heat model is conservative.

3. **Because of the inexactness of the new reactivity edit feature, use of values in the edit either directly or as constituent factors in calculations of parameters for comparisons to formal performance criteria must be justified.**

Dominion Evaluation

The editing feature provided in RETRAN 02/MOD005.0 is not used as a quantitative indicator of reactivity feedback and is not used to report analysis results.

<u>DOMINION RESPONSE TO QUESTION 1b</u>

As required by the VEP-FRD-41-A SER, Dominion provided RETRAN model decks to NRC in 1985 as described in Reference 13. Therefore, Dominion satisfied the VEP-FRD-41-A SER requirement. The SER Conclusions section for VEP-FRD-41-A states "The staff requires that all future modification of VEPCO RETRAN model and the error reporting and change control models should be placed under full quality assurance procedures." Dominion has complied with this requirement. Dominion does not interpret the original SER restriction to require submission of model decks after changes are made, especially for changes to plant inputs. Reference 13 was provided to NRC staff on February 26, 2003.

NRC RETRAN QUESTION 2

2. Doppler Reactivity Feedback (page 8 of the submittal dated August 10, 1993)

a. The Doppler reactivity feedback is calculated by VEPCO's correlation of Doppler reactivity as a function of core average fuel temperature and core burnup. Please provide a technical description of how this correlation is derived, including the codes and methods used. Discuss any limitations or restrictions regarding the use of this correlation.

b. Discuss the method of calculation and application of suitable weighting factors used to acquire a target Doppler temperature coefficient or Doppler power defect. Indicate the Updated Final Safety Analysis (UFSAR) transients that use this method.

DOMINION RESPONSE TO QUESTION 2.a

The North Anna and Surry Version 1 RETRAN models use a Doppler feedback correlation that is derived from data that models the dependence of Doppler Temperature Coefficient (DTC) on changes in fuel temperature, boron concentration, moderator density and fuel burnup. Through sensitivity studies using the XSDRNPM computer code [Reference 14], the DTC at various conditions was determined. XSDRNPM is a member of the SCALE code package.

The data gathered for North Anna and Surry was used to develop models to predict DTCs. A procedure to calculate a least squares fit to non-linear data with the Gauss-Newton iterative method was used to determine fit coefficients for the collected data. The model values and the percentage difference between the model and XSDRNPM values were determined. The model was also compared to 2D PDQ and 3D PDQ quarter core predictions. The PDQ code is described in Reference 15. The largest percentage difference between the model and the XSDRNPM and PDQ cases is within the nuclear reliability factor for DTC in Reference 16 over the range of conditions of interest to non-LOCA accident analysis.

It was shown that the effect of burnup, boron, and moderator specific volume could be represented as multipliers to the base DTC versus fuel temperature curve. The Doppler correlation has a core average fuel temperature component, DTC_{Tf}, and a burnup component, BURNMP. Since during a transient the burnup may be assumed to be constant, the burnup multiplier of the Doppler correlation is also assumed to be constant. To separate the reactivity feedbacks into a prompt and slower component, the impact of boron concentration and moderator density changes on the Doppler are assumed to be accounted for in the moderator feedback modeling, as these are slower feedback phenomena. Hence, the Doppler reactivity feedback is dependent only on changes in fuel temperature, which provides the prompt feedback component. The boron concentration and moderator density (specific volume) multipliers in the DTC correlation are thereby set to 1.

The DTC correlation is qualified over the range of core design DTC limits for North Anna and Surry and is described by the following equation:

$$DTC(pcm/°F) = DTC_{Tf} * BURNMP * WF$$

where

DTC_{Tf}, the fuel temperature dependence, equals $A*T_f^{0.5} + B*T_f + C$

T_f is the effective core average fuel temperature in °F and A, B, and C are correlation coefficients

BURNMP, which models burnup changes, equals DTC_{ref}/DTC_{Tf547}

DTC_{ref} is the reference DTC at the burnup of interest at hot-zero-power with 2000 ppm boron (pcm/°F)

DTC_{Tf547} is the solution to the above DTC_{Tf} equation at 547 °F.

WF is the user supplied weighting factor term that allows the user to adjust the design information to bound specific Doppler defects.

DOMINION RESPONSE TO QUESTION 2.b

The Doppler feedback can be adjusted to a target DTC at a given fuel temperature by changing the weighting factor. For FSAR analyses in which the Doppler reactivity feedback is a key parameter, the target DTC used in RETRAN is either a least negative or most negative DTC. The RETRAN Doppler weighting factor is set so that RETRAN will initialize to the Reload Safety Analysis Checklist (RSAC) DTC limit at a core average fuel temperature that corresponds to the conditions at which the RSAC DTC limit was set.

To set the weighting factor to provide a least negative DTC, the DTC correlation is solved for the Doppler weighting factor, WF, for the appropriate core average fuel temperature and least negative DTC values. This value of the weighting factor is then entered in RETRAN control input. Likewise, to set the weighting factor to provide a most negative DTC, the weighting factor is solved using the DTC correlation with the appropriate core average fuel temperature and most negative DTC value.

All non-LOCA UFSAR transient RETRAN analyses, with the exception of the rod ejection event, apply an appropriate weighting factor to acquire a target Doppler temperature coefficient.

The rod ejection event requires additional Doppler reactivity feedback. This additional feedback is calculated as a PWF (power weighting factor), and the Doppler weighting factor calculated as described herein needs to be multiplied by the PWF before being input to the RETRAN model. The application of the power weighting factor to rod ejection analyses is described in Section 2.2.3 of Reference 4.

NRC RETRAN QUESTION 3

3. By letter dated August 10, 1993, VEPCO discussed the expansion of the North Anna RETRAN model from two geometric configurations to four geometric configurations. The model options increased from a one-loop and two-loop reactor coolant system (RCS) geometry with a single-node steam generator secondary side, to one-loop and three-loop RCS geometry with either single- or multi-node steam generator secondary side. Please discuss the process used for choosing which of the four configurations to use for a particular transient, and identify which model is used for each of the North Anna and Surry UFSAR, Chapter 15, transients that were evaluated using RETRAN.

DOMINION RESPONSE TO QUESTION 3

Historically, choosing between the 1-loop and 2-loop RCS RETRAN models was based on the expected plant response from the transient and on the importance of modeling differences between RCS loops. For example, a steamline break affects the conditions in the faulted steam generator RCS loop different from the other loops. When advances in computer processor speed and memory eliminated the need to collapse symmetric loops, Dominion developed 3-loop RCS models and retired the 1-loop and 2-loop models. Some UFSAR analyses of record reflect 1-loop and 2-loop RETRAN analyses because the events have not been reanalyzed since the implementation of the 3-loop models. RETRAN analyses in the UFSAR use the single-node SG secondary model. Dominion uses the multi-node steam generator secondary model for sensitivity studies to confirm the conservatism in the single-node SG secondary. Subsequent to retirement of the 1-loop and 2-loop models, licensing analyses have used the 3-loop RCS geometry with a single-node steam generator. Dominion anticipates that this will continue to be our RETRAN analysis model going forward.

Tables 3a and 3b below show the selected RCS model type for each UFSAR event analyzed with RETRAN for North Anna and Surry, respectively. All analyses use a single-node steam generator secondary model. Note that some UFSAR non-LOCA events have not been analyzed with RETRAN. Future applications of RETRAN may involve analyzing these events to remove the dependence on the vendor. Those analyses would be performed in accordance with regulatory requirements and limitations in the RETRAN SERs and VEP-FRD-41-A.

Table 3a: North Anna UFSAR Chapter 15 Event and RETRAN Model

Event	UFSAR Section	RETRAN Model
Condition II: Events of Moderate Frequency		
Uncontrolled Rod Cluster Control Assembly from a Subcritical Condition	15.2.1	1-Loop
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power	15.2.2	3-Loop
Uncontrolled Boron Dilution	15.2.4	1-Loop
Loss of External Electric Load and/or Turbine Trip	15.2.7	3-Loop
Loss of Normal Feedwater	15.2.8	3-Loop
Loss of Offsite Power to the Station Auxiliaries	15.2.9	3-Loop
Excessive Heat Removal Due to Feedwater System Malfunctions	15.2.10	2-Loop
Excessive Load Increase Incident	15.2.11	1-Loop, 3-Loop
Accidental Depressurization of the Reactor Coolant System	15.2.12	1-Loop
Accidental Depressurization of the Main Steam System	15.2.13	3-Loop
Condition III: LOCA and Related Accidents		
Minor Secondary System Pipe Breaks	15.3.2	3-Loop
Complete Loss of Forced Reactor Coolant Flow	15.3.4	1-Loop
Condition IV: Limiting Faults		
Major Secondary System Pipe Rupture	15.4.2	3-Loop
Steam Generator Tube Rupture	15.4.3	2-Loop and 3-Loop
Locked Reactor Coolant Pump Rotor	15.4.4	2-Loop and 3-Loop
Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	15.4.6	1-Loop

Note that the Rupture of a Control Rod Drive Mechanism Housing, Complete Loss of Forced Reactor Coolant Flow, and Locked Reactor Coolant Pump Rotor analyses have been performed with the RETRAN 3 Loop model as part of the transition to Framatome fuel. These evaluations are currently being reviewed by the NRC and are therefore not incorporated in the current North Anna UFSAR.

Table 3b: Surry UFSAR Chapter 14 Event and RETRAN Model

Event	UFSAR Section	RETRAN Model
Condition II: Events of Moderate Frequency		
Uncontrolled Control-Rod Assembly Withdrawal From a Subcritical Condition	14.2.1	1-Loop
Uncontrolled Control-Rod Assembly Withdrawal at Power	14.2.2	1-Loop
Chemical and Volume Control System Malfunction	14.2.5.2.3	1-Loop
Excessive Heat Removal Due to Feedwater System Malfunctions	14.2.7	FW Temp. Reduction - 3-Loop Excess Feedwater Flow - 2-Loop
Excessive Load Increase Incident	14.2.8	3-Loop
Loss of Reactor Coolant Flow Flow Coastdown Incidents	14.2.9.1	1-Loop
Locked Rotor Incident	14.2.9.2	3-Loop
Loss of External Electrical Load	14.2.10	3-Loop
Loss of Normal Feedwater	14.2.11	3-Loop
Loss of all Alternating Current to the Station Auxiliaries	14.2.12	3-Loop
Standby Safeguards Analyses		
Steam Generator Tube Rupture	14.3.1	2-Loop
Rupture of a Main Steam Pipe (DNB)	14.3.2	3-Loop
Rupture of a Control Rod Drive Mechanism Housing (Control Rod Assembly Ejection)	14.3.3	1-Loop
Feedline Break outside Containment	Appendix 14B	3-Loop

References used in Dominion Responses to RETRAN Questions

1) Letter from C.O. Thomas (USNRC) 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 2, 1984.

2) Letter from A. C. Thadani (USNRC) to R. Furia (GPU), "Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004," October 19, 1988.

3) Letter from A. C. Thadani (USNRC) to W. J. Boatwright (RETRAN02 Maintenance Group), "Acceptance for Use of RETRAN02 MOD005.0," November 1, 1991.

4) Virginia Power Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient", NRC SER dated September 26, 1984.

5) Virginia Power Topical Report VEP-FRD-41-A, "VEPCO Reactor System Transient Analysis using the RETRAN Computer Code," May 1985.

6) Westinghouse report WCAP-9227, "Reactor Core Response to Excessive Secondary Steam Releases," January 1978.

7) Westinghouse report WCAP-8844, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," November 1977.

8) Westinghouse report WCAP-7907-A, "LOFTRAN Code Description," April 1984.

9) Letter, M.L. Bowling (VEPCO) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Supplemental Information on the RETRAN NSSS Model," Serial 93-505, August 10, 1993.

10) Letter, Stuart A. Richards (USNRC) to Gary Vine (EPRI), "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," January 25, 2001.

11) Westinghouse report WCAP-10858-P-A, "AMSAC Generic Design Package," October 1986.

12) Letter from W. L. Stewart (VEPCO) to H. R. Denton (USNRC), "Virginia Electric Power Company, North Anna Power Station Unit No. 2, Response to the Additional Request for Information Concerning Low Power Natural Circulation Testing," Serial No. 427A, August 25, 1983.

References used in Dominion Responses to RETRAN Questions (continued)

13) Letter, W. L. Stewart (VEPCO) to Harold R. Denton (USNRC), "Virginia Power, Surry and North Anna Power Stations, Reactor System Transient Analyses," Serial No. 85-570, August 21, 1985.

14) ORNL-NUREG-CSD-2-Vol 2, Rev. 1, "XSDRNPM-S: A One-Dimensional Discrete-Ordinates Code for Transport Analysis," June 1983.

15) Virginia Power Topical Report VEP-NAF-1, "The PDQ Two Zone Model," July 1990.

16) Virginia Power Topical Report VEP-FRD-45A, "VEPCO Nuclear Design Reliability Factors," October 1982.

17) Branch Technical Position APCSB9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," 1975.

18) EPRI Report, EPRI-NP-1850-CCM-A, Volume 1, Rev. 4, "RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems."

19) Letter from W. L. Stewart (VEPCO) to Harold R. Denton (USNRC), "VEPCO Reactor System Transient Analyses", Serial No. 376, July 12, 1984.

Attachment 2

Response to NRC
PDQ Two Zone Model Questions

Virginia Electric and Power Company
(Dominion)
North Anna and Surry Power Stations

PDQ Code and Model Review, Topical Report VEP-NAF-1, "PDQ Two Zone Model," VEPCO submittal dated October 1, 1990

NRC PDQ QUESTION 1

1. By letter dated December 2, 2002, VEPCO stated that the accuracy of the PDQ model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the PDQ predictions) that demonstrate the accuracy of this model.

DOMINION RESPONSE TO QUESTION 1

The following results are from the N1C16 startup physics tests in October, 2001.

N1C16 STARTUP PHYSICS TESTING RESULTS (October, 2001)

Parameter	Measured	Predicted	Difference (P-M) or (P-M)/M*100	Nuclear Reliability Factor
Critical Boron Concentration (HZP, ARO) ppm	2109	2133	24	± 50
Critical Boron Concentration (HZP, reference bank in) ppm	1897	1917	20	± 50
Critical Boron Concentration (HFP, ARO, EQ XE) ppm	1405	1429	24	± 50
Isothermal Temperature Coefficient (HZP, ARO) pcm/°F	-2.87	-3.29	-0.42	± 3.0
Differential Boron Worth (HZP, ARO) pcm/ppm	-6.59	-6.46	-2.0%	1.10
Reference Bank Worth (B-bank, dilution) pcm	1393.2	1396	0.2%	1.10
D-bank Worth (Rod Swap), pcm	944.6	979	3.6%	1.10
C-bank Worth (Rod Swap), pcm	760.4	779.3	2.5%	1.10
A-bank Worth (Rod Swap), pcm	356.6	348.4	-2.3%	1.10
SB-bank Worth (Rod Swap), pcm	930.5	969.8	4.2%	1.10
SA-bank Worth (Rod Swap), pcm	1012.5	1003.4	-0.9%	1.10
Total Bank Worth, pcm	5397.6	5476	1.5%	1.10
HFP ARO EQ XE FΔH (BOC)	1.405	1.378	-1.9%	1.05
HFP ARO EQ XE F_Q (BOC)	1.654	1.601	-3.2%	1.075
HFP ARO EQ XE Axial Offset (BOC)	-2.5	-3.0	-0.5%	N/A

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
29% POWER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1			PREDICTED MEASURED PCT DIFFERENCE				0.246 0.249 1.0	0.275 0.280 1.8	0.242 0.246 1.6				PREDICTED MEASURED PCT DIFFERENCE			1
2					0.331 0.336 1.5	0.649 0.656 1.0	1.053 1.061 0.8	0.836 0.844 1.0	1.045 1.062 1.6	0.647 0.674 4.3	0.331 0.338 2.2					2
3				0.365 0.381 4.3	1.179 1.200 1.8	1.299 1.313 1.1	1.170 1.177 0.6	1.301 1.301 0.0	1.168 1.175 0.6	1.299 1.312 1.0	1.183 1.196 1.1	0.368 0.379 3.0				3
4			0.368 0.381 3.6	0.898 0.919 2.3	1.333 1.358 1.9	1.323 1.335 0.9	1.305 1.309 0.3	1.243 1.244 0.0	1.304 1.303 -0.1	1.324 1.318 -0.4	1.338 1.343 0.4	0.897 0.908 1.2	0.365 0.369 1.1			4
5		0.356 0.367 3.1	1.204 1.259 4.6	1.346 1.370 1.8	1.221 1.228 0.6	1.262 1.265 0.2	1.168 1.168 0.0	1.236 1.233 -0.2	1.168 1.164 -0.3	1.261 1.258 -0.3	1.220 1.222 0.1	1.339 1.358 1.4	1.197 1.212 1.2	0.355 0.361 1.9		5
6		0.665 0.668 0.5	1.321 1.334 0.9	1.334 1.338 0.3	1.266 1.252 -1.1	1.017 1.012 -0.5	1.145 1.139 -0.6	1.071 1.065 -0.5	1.143 1.133 -0.8	1.017 1.012 -0.4	1.265 1.261 -0.3	1.332 1.333 0.1	1.319 1.330 0.8	0.665 0.677 1.9		6
7	0.258 0.256 -1.0	1.070 1.060 -0.9	1.186 1.166 -1.7	1.317 1.306 -0.8	1.174 1.164 -0.9	1.145 1.135 -0.9	1.038 1.027 -1.1	1.005 0.993 -1.2	1.038 1.018 -1.9	1.148 1.135 -1.1	1.174 1.163 -0.9	1.316 1.294 -1.7	1.187 1.195 0.7	1.074 1.108 3.2	0.259 0.267 3.1	7
8	0.284 0.282 -1.0	0.853 0.845 -1.0	1.320 1.298 -1.7	1.255 1.244 -0.9	1.243 1.232 -0.9	1.074 1.065 -0.9	1.005 0.994 -1.1	0.996 0.984 -1.2	1.005 0.991 -1.4	1.074 1.063 -1.0	1.243 1.233 -0.7	1.255 1.251 -0.3	1.320 1.333 1.0	0.853 0.879 3.0	0.284 0.293 2.9	8
9	0.259 0.258 -0.6	1.074 1.069 -0.5	1.187 1.181 -0.5	1.316 1.310 -0.5	1.174 1.172 -0.2	1.148 1.139 -0.8	1.038 1.025 -1.2	1.005 0.991 -1.4	1.038 1.024 -1.3	1.145 1.133 -1.1	1.174 1.167 -0.6	1.317 1.319 0.2	1.186 1.201 1.3	1.070 1.097 2.5	0.258 0.264 2.4	9
10		0.665 0.665 0.1	1.319 1.326 0.6	1.332 1.330 -0.1	1.265 1.259 -0.5	1.017 1.007 -0.9	1.143 1.123 -1.7	1.071 1.053 -1.7	1.145 1.127 -1.6	1.017 1.003 -1.4	1.266 1.255 -0.9	1.334 1.339 0.4	1.321 1.344 1.8	0.665 0.691 3.8		10
11		0.355 0.356 0.3	1.197 1.203 0.5	1.339 1.337 -0.1	1.220 1.211 -0.7	1.261 1.246 -1.2	1.168 1.147 -1.8	1.236 1.211 -2.0	1.168 1.146 -1.9	1.262 1.229 -2.6	1.221 1.198 -1.9	1.346 1.361 1.1	1.204 1.235 2.6	0.356 0.366 3.0		11
12			0.365 0.373 2.4	0.897 0.897 -0.1	1.338 1.329 -0.7	1.324 1.307 -1.3	1.304 1.276 -2.2	1.243 1.218 -2.0	1.305 1.286 -1.5	1.323 1.309 -1.1	1.333 1.334 0.1	0.898 0.939 4.5	0.368 0.407 10.6			12
13				0.368 0.367 -0.3	1.183 1.179 -0.3	1.299 1.286 -1.0	1.168 1.149 -1.6	1.301 1.270 -2.4	1.170 1.163 -0.7	1.299 1.292 -0.5	1.179 1.180 0.1	0.365 0.372 1.9				13
14					0.331 0.342 3.4	0.647 0.644 -0.4	1.045 1.037 -0.7	0.836 0.834 -0.1	1.053 1.077 2.3	0.649 0.653 0.6	0.331 0.332 0.2					14
15			STANDARD DEVIATION =1.235				0.242 0.243 0.3	0.275 0.276 0.2	0.246 0.251 1.8				AVERAGE PCT DIFFERENCE = 1.3			15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

SUMMARY

MAP NO: N1-16-01 DATE: 10/10/01 POWER: 29%

CONTROL ROD POSITIONS: F-Q(Z) = 2.108 CORE TILT:

D BANK AT 150 STEPS F-DH(N) = 1.546

NW 1.0037	NE 1.0031
SW 0.9922	SE 1.0010

F(Z) = 1.283

BURNUP = 5.0 MWD/MTU A.O. = -6.233

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
74% POWER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1			PREDICTED MEASURED PCT DIFFERENCE.				0.271 0.272 0.4	0.311 0.311 0.1	0.267 0.269 0.9				PREDICTED MEASURED PCT DIFFERENCE.			1
2					0.339 0.342 0.8	0.660 0.664 0.6	1.082 1.088 0.6	0.921 0.927 0.7	1.074 1.087 1.2	0.657 0.678 3.1	0.339 0.344 1.6					2
3				0.369 0.376 2.0	1.140 1.152 1.1	1.258 1.267 0.7	1.164 1.169 0.4	1.297 1.300 0.2	1.162 1.168 0.6	1.258 1.268 0.9	1.143 1.152 0.8	0.372 0.375 0.8				3
4			0.371 0.381 2.5	0.880 0.892 1.4	1.278 1.293 1.2	1.285 1.292 0.6	1.275 1.279 0.3	1.227 1.230 0.2	1.274 1.276 0.2	1.285 1.286 0.1	1.282 1.287 0.4	0.879 0.886 0.8	0.368 0.371 0.7			4
5		0.361 0.369 2.1	1.159 1.199 3.5	1.287 1.302 1.2	1.196 1.198 0.2	1.253 1.255 0.1	1.172 1.174 0.2	1.227 1.227 0.0	1.171 1.172 0.1	1.253 1.254 0.1	1.195 1.199 0.3	1.281 1.296 1.1	1.154 1.164 0.9	0.360 0.362 0.5		5
6		0.673 0.672 0.0	1.274 1.279 0.4	1.291 1.291 0.0	1.256 1.244 -0.9	1.103 1.101 -0.2	1.176 1.173 -0.3	1.100 1.098 -0.2	1.173 1.171 -0.2	1.103 1.104 0.1	1.255 1.255 0.0	1.290 1.292 0.2	1.272 1.281 0.7	0.673 0.682 1.3		6
7	0.283 0.279 -1.3	1.094 1.081 -1.3	1.174 1.152 -1.9	1.282 1.270 -0.9	1.175 1.167 -0.7	1.175 1.169 -0.6	1.084 1.077 -0.7	1.053 1.048 -0.5	1.084 1.079 -0.5	1.177 1.173 -0.4	1.175 1.171 -0.4	1.282 1.269 -1.0	1.176 1.184 0.7	1.098 1.125 2.4	0.284 0.290 2.2	7
8	0.319 0.315 -1.3	0.935 0.923 -1.3	1.310 1.282 -2.1	1.235 1.223 -1.0	1.231 1.226 -0.4	1.101 1.097 -0.4	1.053 1.048 -0.6	1.049 1.043 -0.5	1.053 1.048 -0.5	1.101 1.096 -0.5	1.231 1.228 -0.3	1.235 1.235 0.0	1.310 1.321 0.8	0.935 0.951 1.7	0.319 0.324 1.6	8
9	0.284 0.281 -1.0	1.098 1.089 -0.9	1.176 1.166 -0.8	1.282 1.276 -0.5	1.175 1.178 0.2	1.177 1.173 -0.3	1.084 1.078 -0.6	1.053 1.046 -0.7	1.084 1.078 -0.6	1.175 1.169 -0.5	1.175 1.174 -0.1	1.282 1.287 0.4	1.174 1.186 1.0	1.094 1.112 1.6	0.283 0.284 0.5	9
10		0.673 0.671 -0.3	1.272 1.275 0.2	1.290 1.287 -0.2	1.255 1.252 -0.3	1.103 1.098 -0.4	1.173 1.163 -0.9	1.100 1.089 -0.9	1.176 1.166 -0.9	1.103 1.097 -0.6	1.256 1.254 -0.2	1.291 1.303 0.9	1.274 1.291 1.4	0.673 0.692 2.8		10
11		0.360 0.360 -0.1	1.154 1.155 0.1	1.281 1.278 -0.2	1.195 1.190 -0.5	1.253 1.243 -0.8	1.171 1.158 -1.1	1.227 1.211 -1.3	1.172 1.158 -1.2	1.253 1.234 -1.5	1.196 1.186 -0.8	1.287 1.302 1.1	1.159 1.176 1.5	0.361 0.368 1.9		11
12			0.368 0.371 0.7	0.879 0.877 -0.3	1.282 1.274 -0.6	1.285 1.273 -0.9	1.274 1.256 -1.5	1.227 1.209 -1.5	1.275 1.261 -1.1	1.285 1.275 -0.8	1.278 1.282 0.3	0.880 0.914 3.9	0.371 0.380 2.3			12
13				0.372 0.370 -0.5	1.143 1.138 -0.5	1.258 1.247 -0.9	1.162 1.146 -1.4	1.297 1.270 -2.1	1.164 1.155 -0.7	1.258 1.248 -0.8	1.140 1.140 0.0	0.369 0.375 1.6				13
14					0.339 0.343 1.2	0.657 0.653 -0.7	1.074 1.064 -0.9	0.921 0.916 -0.5	1.082 1.096 1.3	0.660 0.661 0.1	0.339 0.339 0.0					14
15			STANDARD DEVIATION =0.694				0.267 0.263 -1.4	0.311 0.310 -0.4	0.271 0.273 0.9				AVERAGE PCT DIFFERENCE. = 0.8			15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

MAP NO: N1-16-02

DATE: 10/11/01

SUMMARY

POWER: 74%

CONTROL ROD POSITIONS:

D BANK AT 192 STEPS

F-Q(Z) = 1.848

F-DH(N) = 1.451

F(Z) = 1.184

BURNUP = 24.0 MWD/MTU

CORE TILT:

NW 1.0014	NE 1.0039
SW 0.9933	SE 1.0014

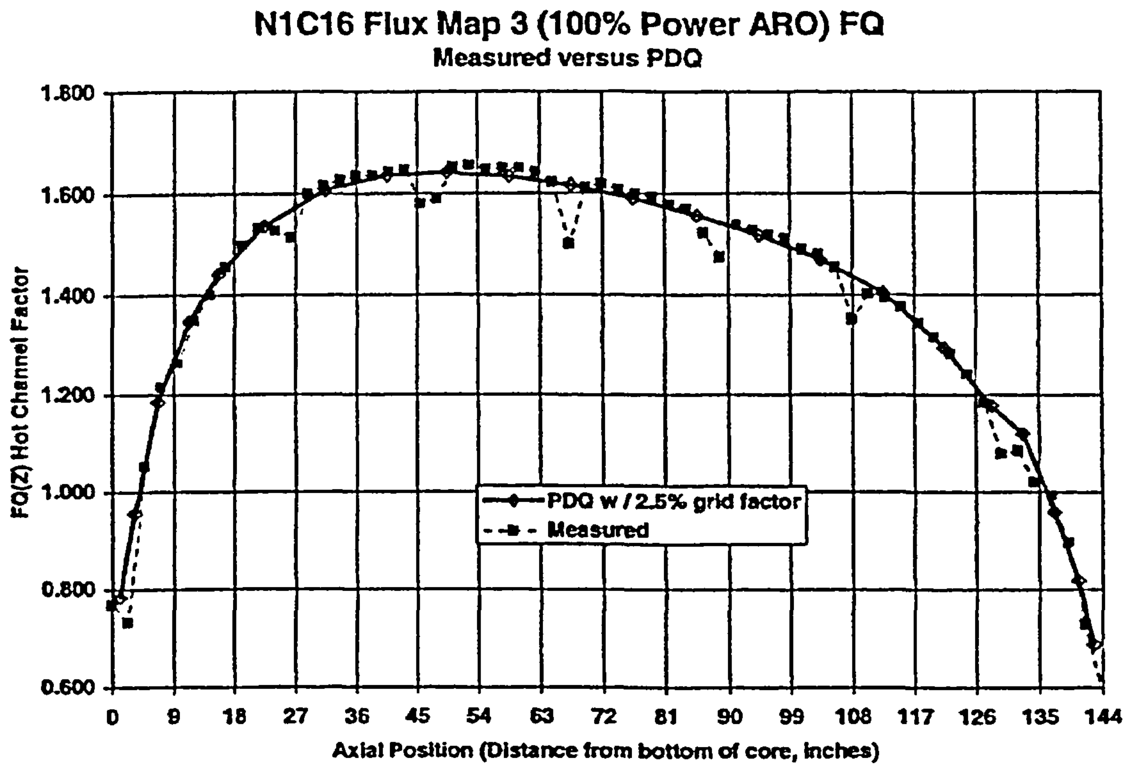
A.O. = 0.088

NORTH ANNA UNIT 1 - CYCLE 16 STARTUP PHYSICS TESTS
ASSEMBLYWISE POWER DISTRIBUTION
100% POWER

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1		PREDICTED					0.283	0.330	0.279			PREDICTED				1
		MEASURED					0.283	0.334	0.281			MEASURED				
		PCT DIFFERENCE					0.1	1.3	0.9			PCT DIFFERENCE				
2					0.342	0.662	1.087	0.963	1.080	0.660	0.342					2
					0.343	0.662	1.085	0.961	1.089	0.682	0.349					
					0.2	-0.1	-0.2	-0.2	0.8	3.3	2.0					
3				0.371	1.118	1.239	1.156	1.297	1.154	1.239	1.121	0.373				3
				0.384	1.122	1.238	1.150	1.279	1.155	1.251	1.135	0.384				
				3.4	0.4	-0.1	-0.5	-1.4	0.1	1.0	1.3	2.9				
4			0.372	0.871	1.255	1.263	1.265	1.217	1.265	1.264	1.259	0.870	0.370			4
			0.379	0.878	1.254	1.262	1.262	1.213	1.266	1.269	1.268	0.876	0.371			
			1.6	0.8	-0.1	-0.1	-0.3	-0.3	0.1	0.4	0.7	0.6	0.4			
5		0.362	1.133	1.263	1.185	1.258	1.173	1.228	1.173	1.258	1.185	1.258	1.128	0.361		5
		0.367	1.160	1.272	1.190	1.257	1.173	1.230	1.176	1.262	1.192	1.257	1.133	0.371		
		1.2	2.3	0.7	0.4	-0.1	0.0	0.2	0.3	0.3	0.6	-0.1	0.4	2.8		
6		0.673	1.251	1.268	1.260	1.158	1.196	1.110	1.194	1.158	1.259	1.267	1.250	0.673		6
		0.669	1.248	1.261	1.242	1.154	1.199	1.119	1.198	1.158	1.259	1.262	1.250	0.679		
		-0.6	-0.2	-0.5	-1.5	-0.3	0.3	0.9	0.4	0.0	0.0	-0.4	0.0	0.9		
7	0.294	1.097	1.164	1.270	1.276	1.195	1.100	1.067	1.100	1.197	1.176	1.270	1.165	1.101	0.295	7
	0.289	1.079	1.139	1.255	1.165	1.192	1.108	1.074	1.107	1.202	1.173	1.253	1.159	1.117	0.299	
	-1.6	-1.6	-2.1	-1.2	-0.9	-0.2	0.7	0.7	0.6	0.4	-0.2	-1.4	-0.5	1.4	1.5	
8	0.337	0.975	1.307	1.223	1.231	1.111	1.067	1.062	1.067	1.111	1.231	1.223	1.307	0.975	0.337	8
	0.332	0.958	1.276	1.208	1.222	1.109	1.069	1.065	1.073	1.126	1.231	1.213	1.282	0.994	0.343	
	-1.7	-1.8	-2.4	-1.2	-0.7	-0.2	0.2	0.3	0.5	1.4	0.0	-0.8	-1.9	2.0	1.9	
9	0.295	1.101	1.165	1.270	1.176	1.197	1.100	1.067	1.100	1.195	1.176	1.270	1.164	1.097	0.294	9
	0.291	1.087	1.152	1.264	1.184	1.198	1.100	1.066	1.098	1.187	1.174	1.272	1.167	1.116	0.300	
	-1.3	-1.2	-1.1	-0.5	0.7	0.1	-0.1	-0.1	-0.2	-0.7	-0.1	0.1	0.3	1.7	2.0	
10		0.673	1.250	1.267	1.259	1.158	1.194	1.110	1.196	1.158	1.260	1.268	1.251	0.673		10
		0.668	1.245	1.265	1.261	1.156	1.189	1.103	1.188	1.150	1.260	1.281	1.270	0.696		
		-0.7	-0.4	-0.1	0.1	-0.1	-0.4	-0.6	-0.7	-0.7	0.0	1.0	1.5	3.5		
11		0.361	1.128	1.258	1.185	1.258	1.173	1.228	1.173	1.258	1.185	1.263	1.133	0.362		11
		0.361	1.131	1.259	1.186	1.252	1.162	1.214	1.161	1.239	1.183	1.285	1.157	0.372		
		-0.1	0.3	0.1	0.1	-0.4	-0.9	-1.1	-1.1	-1.5	-0.2	1.7	2.1	2.6		
12			0.370	0.870	1.259	1.264	1.265	1.217	1.265	1.263	1.255	0.871	0.372			12
			0.381	0.874	1.258	1.255	1.242	1.198	1.252	1.254	1.264	0.913	0.389			
			3.0	0.4	-0.1	-0.7	-1.8	-1.6	-1.1	-0.7	0.7	4.8	4.5			
13				0.373	1.121	1.239	1.154	1.297	1.156	1.239	1.118	0.371				13
				0.374	1.122	1.231	1.138	1.267	1.147	1.225	1.120	0.379				
				0.2	0.1	-0.6	-1.4	-2.3	-0.8	-1.2	0.2	2.1				
14					0.342	0.660	1.080	0.963	1.087	0.662	0.342					14
					0.351	0.659	1.073	0.959	1.108	0.664	0.343					
					2.6	-0.2	-0.6	-0.4	1.9	0.2	0.1					
15		STANDARD					0.279	0.330	0.283			AVERAGE				15
		DEVIATION					0.279	0.330	0.287			PCT DIFFERENCE				
		=0.906					0.2	0.0	1.4			= 0.9				
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

SUMMARY

MAP NO: N1-16-03	DATE: 10/23/01	POWER: 100%
CONTROL ROD POSITIONS:	F-Q(Z) = 1.786	CORE TILT:
D BANK AT 226 STEPS	F-DH(N) = 1.405	NW 0.9982 | NE 1.0035
	F(Z) = 1.139	SW 0.9954 | SE 1.0029
	BURNUP = 436.4 MWD/MTU	A.O. = -2.537



NRC PDQ QUESTION 2

There do not appear to be any limitations or restrictions associated with the use of PDQ Two Zone as described in VEP-NAF-1. Please justify that PDQ Two Zone is applicable over all ranges of operation expected for North Anna and Surry.

DOMINION RESPONSE TO QUESTION 2

Use of the PDQ Two Zone Model is limited to North Anna and Surry cores containing fuel that is similar to existing 17x17 and 15x15 designs. The range of applicability is stated in general terms in Section 2.1 of VEP-FRD-42 Rev 2:

"These models have been used to model the entire range of cores at the Surry and North Anna power stations, including evolutionary changes in fuel enrichment, fuel density, loading pattern strategy, spacer grid design and material, fuel clad alloy, and burnable poison material and design. Some of these changes were implemented as part of various Lead Test Assembly programs, and have included fuel assemblies from both Westinghouse and Framatome-ANP. The predictive accuracy of the models throughout these changes demonstrates that incremental design variations in fuel similar to the Westinghouse design are well within the applicable range of the core design models. Each model has sufficient flexibility such that minor fuel assembly design differences similar to those noted can be adequately accounted for using model design input variables."

Limitations associated with the PDQ Two Zone models stem primarily from consideration of the source of collapsed cross section data (primarily CELL2, a pin cell model) and from practical considerations involving the level of complexity that can be accommodated in PDQ. Based on these considerations, the scope of benchmarking that has been performed to date, and the range of core designs successfully modeled in the past, the PDQ Two Zone model should be restricted according to the following characteristics:

1) Geometry
   a) Square pitch fuel (cylindrical fuel pellets and rods)
   b) 15x15 or 17x17 design
   c) 5x5 mesh blocks per assembly (x-y)
   d) 26 axial nodes (22 in the fuel region)
   e) ¼ core or full core representation
2) Fuel Material
   a) Low enriched UO_2 (4.6 w/o U_{235} or less)
      i) Cores with fuel up to 4.45 w/o have been successfully modeled to date
      ii) Cross section behavior (enrichment trends and fidelity to CELL2) has been checked up to 4.6 w/o U_{235} for burnups up to 76 GWD/T.
   b) Fuel pin burnup of approximately 70 GWD/T has been achieved in PDQ Two Zone designed cores as part of a high burnup demonstration program.
3) Burnable poisons
   a) Discrete rods inserted into fuel assembly guide thimbles
      i) Both annular borosilicate glass and solid B4C in alumina designs have been well predicted throughout many cycles of operation
      ii) Both SS304 and zirconium based cladding has been used
   b) Modeling flexibility has been demonstrated for BP configuration (number of fingers, boron enrichment, poison length, and poison stack axial alignment)
4) Control rods
   a) Ag-In-Cd rods with stainless steel clad (extensive validation and experience)
   b) Hf metal rods in zirconium based clad have been used for vessel fluence reduction in Surry Unit 1
5) Fuel assembly
   a) Modeling flexibility has been demonstrated for Inconel and zirconium based grids of various designs and sizes

There are no current plans for fuel design, core design, or operating strategy changes that would exceed the design characteristics outlined above. There are fuel products in use in the industry, which would be technically possible, but impractical to model in the PDQ Two Zone model (such as fuel with integral poisons). No further development is planned for PDQ and NOMAD. Rather, Dominion plans to transition from using PDQ and NOMAD as primary design tools to use of the CMS models (principally CASMO-4 and SIMULATE-3) as soon as practicable. Topical Report DOM-NAF-1 was submitted in June of 2002. The NRC SER for DOM-NAF-1 was received on March 12, 2003.

NRC PDQ QUESTION 3

PDQ Two Zone cross section representation has been improved through the addition of multiple G-factor capability. Please discuss the methodologies used to determine these factors and discuss when and how they are applied. Include a discussion of the "fictitious crod isotope" mentioned on page 2-23 of your dated October 1, 1990.

DOMINION RESPONSE TO QUESTION 3

The addition of multiple G-factor capability was required to meet these goals for the PDQ Two Zone model:

1) A unified set of cross section data to accurately span the entire operating range of the cores (i.e., temperatures, boron concentration, BP combinations, burnup, etc.)
2) A system with the flexibility to model variations such as spacer grid changes, BP enrichment variations, fuel enrichment changes, and clad isotopic changes without requiring the generation of new cross section data.

The process used for G-factor selection can be broken down as follows:

1) Identify known required physical variables (such as moderator temperature, moderator density, fuel temperature, and soluble boron concentration).
2) Identify significant isotopic inter-dependencies (such as the U-235 / Pu-239 interaction in thermal absorption and thermal fission cross sections) using CELL-2.
3) Sort in order of importance and modeling complexity.
4) Develop the primary dependence tables.
5) Develop the G-factor (multiplier) tables.

The importance of a particular factor was judged by estimating the first-order reactivity impact (essentially a partial derivative). The complexity of modeling varies according to the degree of separability from other variables. PDQ uses a table system to represent cross sections. The first table for a particular cross section represents the variation of the cross section using the three most important variables. Additional tables are treated as multipliers (G-factors) on the interpolant from the first table.

Each table has a primary variable (called the diagonal) and up to two secondary variables. The diagonal represents the nominal combination of the three variables. Branch cases are used to perturb each secondary variable. The tables can be considered a dual 2-D representation and not a true 3-D representation since the secondary variables cannot be changed simultaneously.

For example, the U^{235} microscopic thermal absorption cross section is a function of the U^{235} number density, the Pu^{239} number density, and the Pu^{241} number density. The diagonal represents the U^{235} cross section at combinations of the three nuclides found in a CELL-2 depletion of a particular enrichment at nominal conditions. The branch cases vary the quantity of Pu^{239} or Pu^{241} at several of the nominal burnup points. In this way, the second order reactivity impact of depleting a fuel assembly in PDQ at off-nominal conditions (such as more BP, hotter moderator temperatures, or more soluble boron) resulting in more Pu is directly captured without use of a "history" variable. In addition, this type of representation makes the model flexible for modeling different fuel enrichments (typically within ± 0.2 w/o of the CELL-2 enrichment).

Important cross section effects that are not captured in the main cross section table are applied by use of multiplicative G-factors. Each G-factor table is constructed in the same manner as the main cross section table. Using the previous example for U^{235}, one G-factor for the thermal absorption cross section is a function of moderator temperature, moderator density, and fuel burnup. The value of the G-factor at the "reference" moderator temperature (583.4 °F for North Anna) is 1.0. The ratio of the U^{235} thermal absorption cross section at other temperatures to the reference value at 583.4 °F is provided at several diagonal points ranging from HFP to CZP temperatures. The variation in these ratio values caused by changes in moderator density (same moderator temperature but a different pressure) or burnup is provided at the branch points.

An important factor in this method of cross section representation is that PDQ Two Zone features a predominantly microscopic model. That is, most cross sections are represented by means of direct tracking of nuclide number densities via depletion chains coupled with microscopic cross section data. A total of 34 physical nuclides are tracked in addition to several pseudo-nuclides which represent state variables (such as moderator temperature) or lumped macroscopic effects (such as the remaining fission products or control rod insertion). Tracking individual nuclides means that the first order effect on reactivity of a change in nuclide concentration is directly modeled even with a constant microscopic cross section. Complex representation of microscopic cross section dependence serves to provide accuracy at the second and third order level even over an extended range of state variables, and provides modeling flexibility for physical changes in fuel design (such as grid material or grid volume changes).

The cross section modeling process described is complex and was designed to be a one-time event. Sufficient modeling flexibility was designed in to preclude the need for core designers to perform cross section modeling in addition to core design work. Over the 14 years since the G-factor strategy was developed, few changes have been made. These changes have been predominantly to extend capabilities rather than revise strategy. One such change was the addition of cross section data to model use of Hafnium rods for reactor vessel fluence suppression.

An important component of cross section modeling is the verification that the cross section representation is accurate and robust. Part of the G-factor development process involved comparison of PDQ single assembly model eigenvalues to CELL-2 using a wide range of state variables and burnup. A goal of matching reactivity within 100 pcm was usually met for cases using unrodded fuel (the only comparison to a pin cell model that can be made accurately). In addition, comparisons to KENO calculations were made for fresh fuel over a wide range of state variables, with and without control rods and BP rods. The KENO benchmarking / normalization loop is shown in Figure 2-1 of VEP-NAF-1.

The "crod" isotope is one of the pseudo-nuclides mentioned above. Because CELL-2 is a pin cell model and cannot properly represent control rod insertion, control rod macroscopic cross sections were obtained from a KENO model. These cross sections include not only the primary effect of a change in macroscopic absorption, but also the net change in fuel macroscopic cross sections (including removal and fission). In order to overlay these macroscopic changes on the fuel cross sections, the control rod insertion is treated as the addition of a nuclide named "crod" with a number density of 1.0. The macroscopic cross section changes are represented in tables as microscopic cross sections. When multiplied by the crod number density of 1.0, the full macroscopic effect of the rod insertion is obtained. This model also makes possible an approximate modeling of fractional control rod insertion (insertion into only part of a node axially) by specifying a volume weighted value for the crod nuclide. For insertion into the top half of a node, the crod nuclide number density is set to 0.5 in that node. Because the crod number density and cross sections are non-physical for a microscopic model, the crod nuclide is specified as non-depleting.

NRC PDQ QUESTION 4

Table 3.2 of this submittal lists the existing nuclear reliability factors and the PDQ Two Zone nuclear uncertainty factors (NUF). Please discuss the methodology used to calculate each of the PDQ NUF values, and indicate when NRC approval was obtained.

DOMINION RESPONSE TO QUESTION 4

VEP-FRD-19A (The PDQ 07 Discrete Model, SER dated May 18, 1981) and VEP-FRD-45A (VEPCO Nuclear Design Reliability Factors, SER dated August 5, 1982) are two NRC approved references relevant to a discussion of nuclear reliability factor methodology.

In VEP-FRD-19A, a total of four cycles of data (startup physics measurements, flux map data, and boron letdown curves) were provided for comparison between predictions and measurements. Overall averages of vendor code differences (measured versus predicted) were also presented. No statistical methodology was used. In the conclusion section, results were stated to be *"predicted typically within"* the following percentages:

- Assembly average power, 2% standard deviation
- Peak F∆H, 2.5%
- Assembly average burnup, 2.5%
- Critical soluble boron concentration, 30 ppm
- Boron worth, 3%
- Integral control rod worth, 6%

The SER for VEP-FRD-19A restates these values and provides the following assessment, which indicates the acceptability of using *"sufficient examples"* which support reasonable uncertainties:

"We have reviewed the data presented to support the conclusions regarding the uncertainties in the calculated results. We conclude the sufficient examples of comparisons between calculation and measurement to permit the evaluation of calculational uncertainties. We concur with the particular values of uncertainties given in the topical report and repeated in Section 1 above."

In VEP-FRD-45A, a more statistically rigorous method was used to derive the NUF/NRF for the total peaking factor F_Q. Flux map data processed by the INCORE code was used to compare measured and predicted peak pin power in monitored fuel assemblies. Comparisons were made conservatively at points axially mid-way between spacer grids (PDQ does not model the grid depressions or the between grid power peaking) for assemblies of greater than average power. Flux maps from three cycles were included in the data.

The Kolmogorov-Smirnov test (the D test) was used to assess the assumption of normality for the percent difference data. The assumption of normality was found to be acceptable for the pooled data for each of the three cycles based on the results of the D test. A one-sided upper tolerance limit was defined as:

$$TL = X + (K \times S)$$

where K is the one sided tolerance factor for 95% probability and a 95% confidence level (95/95). X is the mean and S is the standard deviation of the % difference data. VEP-FRD-45A references USNRC Regulatory Guide 1.126, Rev. 1 (March 1978) as a source for values of K based on sample size. The NUF was defined as:

$$NUF = 1 + (TL/100)$$

For example, if the value of TL is 10%, the NUF is 1.10. The NRF is then set to conservatively bound the NUF. A discussion of this methodology may be found in Sections 3.1, 3.2, and 3.3 of VEP-FRD-45A. The statistical approach was only used for the F_Q NRF. As stated in the SER:

"Only the total peaking factor NRF is derived from comparisons of predicted and measured power distributions. The NRFs for the first four parameters are derived from analytical engineering arguments"

"We find this reliability factor to be acceptable, based on comparisons with the uncertainties which have been obtained with other currently approved design methods."

"Sufficient information is presented in the report to permit a knowledgeable person to conclude that the NRFs established by Vepco for the Doppler coefficient, the delayed neutron parameters, and the total peaking factor are conservative and acceptable."

The SER therefore considers engineering arguments, statistical data from comparisons of measurements and predictions, and consistency with uncertainty factors approved for other codes to be valid methods of assessing the adequacy of reliability factors. The PDQ Two Zone model NUFs were determined based on a similar combination of comparison to measured data, statistical treatment of the comparisons where appropriate, analytical engineering arguments, and comparisons to reliability factors obtained with other approved models. Because VEP-NAF-1 contains comparisons with 31 operating cycles of measured data, there is greater reliance on statistical treatment of the differences than was possible in the previous reports. Dominion concurs with the use of these methods for determining appropriate reliability factors, and believes that the data presented in VEP-NAF-1 is sufficient to support use of the reliability factors indicated.

One issue that arises in VEP-NAF-1 is the treatment of data for which the hypothesis of normality is rejected (based on the D test). The non-parametric method of Sommerville described and referenced in USNRC Regulatory Guide 1.126, Rev. 1 was used for such samples to construct a 95/95 one-sided upper tolerance limit. This method effectively requires sorting of the data by sign and magnitude and choosing the n^{th} value from the sorted list starting from the most non-conservative value (n=1). The value of n is based on the sample size and is applicable for sample sizes of 60 or greater. The Tables below indicate for each NUF the method used to derive the NUF, associated statistics, and any special considerations used.

NUF Derivation Methods

Parameter	Primary NUF technique(s)	Comments
Control Rod Worth – Integral worth, individual banks	Statistical	Statistics use comparisons to measured rod worth data from 31 cycles of startup physics tests. Assessment of impact of reactivity computer bias included. NRF of 1.10 supported with or without accounting for reactivity computer contribution to uncertainty.
Control Rod Worth – Integral worth, all banks combined	Engineering arguments	The cumulative bank uncertainty is bounded by the individual bank uncertainty.
Differential Bank Worth	Engineering arguments	A qualitative assessment of 14 plots of measured and predicted differential rod worth from 11 cycles (startup physics testing) was performed. All plots are included in the report. This is similar to the treatment used in VEP-FRD-24A for the FLAME model.
Critical Boron Concentration	Statistical	Statistics use comparisons to critical boron measurements from startup physics testing as well as post-outage restarts during each cycle. Conclusions are supported qualitatively by HFP boron letdown curves (measured and predicted) from 30 operating cycles included in the report.
Differential Boron Worth	Statistical and Engineering arguments	Statistics use comparisons to boron worth measurements from startup physics testing. Due to a proportionally large contribution from measurement uncertainty, comparison statistics alone do not lead to a physically reasonable NRF. Engineering arguments were used to assess the level of measurement uncertainty and to support a reasonable NRF via indirect evidence (primarily critical boron concentration).
Moderator Temperature Coefficient	Statistical	Statistics use comparisons to isothermal temperature coefficient measurements from startup physics testing. There is a relatively small Doppler component included, but the range of measured ITCs (-14 to +3 pcm/°F) ensures that the comparison is valid for determining MTC uncertainty. Any uncertainty contribution from the Doppler component is included in the statistics.
$F\Delta H$	Statistical	Statistics use comparisons to measured $F\Delta H$ from incore flux maps for assemblies of greater than average relative power.
F_Q	Statistical	Statistics use comparisons to measured F_Q from incore flux maps for assemblies of greater than average relative power.
Doppler Temperature or Power Coefficient	Engineering Arguments	ECP critical boron predictions (effectively an observation of consistency between HFP and HZP critical boron agreement) are mentioned as indirect evidence supporting the NRF determined for previous models (1.10). Arguments in VEP-FRD-45A remain the primary basis for this NRF. Because it was not explicitly treated for the Two Zone model, this NRF is not listed in the report.

NUF Derivation Methods (Continued)

Parameter	Primary NUF technique(s)	Comments
Effective Delayed Neutron Fraction and Prompt Neutron	None	Arguments in VEP-FRD-45A remain the basis for these NRFs. Because they were not explicitly treated for the Two Zone model, these NRFs are not listed in the report.

Additional Information for Statistically Derived NUF Data

Parameter	Number of observations	Mean	Standard Deviation	Normality assumed?	Standard Deviation Multiplier (K)	N^{th} value (n)
Control Rod Worth – Integral worth, individual banks (raw data)	157	1.0%	4.5%	Yes	1.88	N/A
Critical Boron Concentration	54	6.3 ppm	20.0 ppm	Yes	2.05	N/A
Differential Boron Worth (raw data)	30	-0.3%	4.4%	No	N/A	N/A
Isothermal Temperature Coefficient	57	-0.8 pcm/°F	0.96 pcm/°F	No	N/A	1
FΔH (North Anna)	1479	0.1%	1.9%	No	N/A	60
FΔH (Surry data)	1878	0.0%	1.7%	No	N/A	78
F_Q(North Anna)	9046	-2.2%	2.8%	No	N/A	401
F_Q(Surry data)	9372	-2.6%	3.0%	No	N/A	416

Notes:

1) Difference is defined as Measured – Predicted or as (Measured – Predicted)/Measured.
2) The W test (Shapiro and Wilk) for normality was used for the differential boron worth because the sample size was too small for the D test. A physically realistic uncertainty factor could not be developed based on this non-normal small sample, therefore indirect evidence was presented in the Topical Report in support of the DBW NRF.

NRC PDQ QUESTION 5

Please discuss how the measured data used for statistical comparison to the PDQ Two Zone predicted values were obtained. How were uncertainties in the measured data addressed in the statistical analyses?

DOMINION RESPONSE TO QUESTION 5

Measured data is routinely collected as part of plant operations. Sources of measured data for VEP-NAF-1 include startup physics testing, daily critical boron concentration measurements, criticality condition data, and flux maps (from both startup physics testing and monthly peaking factor surveillance). Much of the data is summarized in a Startup Physics Test Report published following each initial core load or refueling and in a Core Performance Report published following the end of each cycle. The Table below indicates the source of each measured value and an indication of the measurement technique involved.

Measured Parameter	Source	Techniques Involved
Control Rod Worth – Integral bank worth	Startup physics testing (HZP)	Dilution (periodic reactivity computer measurements during a controlled boron dilution) and rod swap (swap of the test bank with a reference bank previously measured by dilution).
Control Rod Worth – Differential bank worth	Startup physics testing (HZP)	Dilution.
Critical boron concentration	Startup physics testing (HZP), daily boron measurements (HFP), ECP procedure (used for mid-cycle return to critical; HZP)	RCS samples are measured by chemical titration. Multiple measurements are used during startup physics testing.
Differential Boron Worth	Startup physics testing (HZP)	Derived from measured reference bank worth and the ARO and reference bank inserted critical boron concentrations. Boron concentrations are measured by chemical titration.
Isothermal Temperature Coefficient	Startup physics testing (HZP)	Reactivity computer measurements during controlled temperature change at HZP.
$F\Delta H$, F_Q	In–core flux maps	Flux maps in this report are taken with movable incore detectors and transformed into measured power distributions using the INCORE code. Maps were taken during startup physics testing (typically <5% power, ~30% power, ~70% power, and ~100% power) and monthly throughout the cycle (typically near HFP).

Measurement uncertainty is inherently and conservatively included in the differences between measured and predicted quantities. NUFs and NRFs derived from such comparisons effectively attribute any measurement uncertainty present to model predictive uncertainty. This type of "raw" comparison data supports all NRFs derived in this report, with the exception of the differential boron worth NRF. Only in the case of the differential boron worth NRF is it necessary to address the effects of measurement uncertainty to support the NRF.

Attachment 3

Responses to NRC Questions on NOMAD

Virginia Electric and Power Company
(Dominion)
North Anna and Surry Power Stations

NOMAD Code Model Review, Topical Report VEP-NFE-1-A. Supplement 1, "VEPCO NOMAD Code and Model," VEPCO Submittal dated Novcemver 13, 1996

NRC NOMAD QUESTION 1

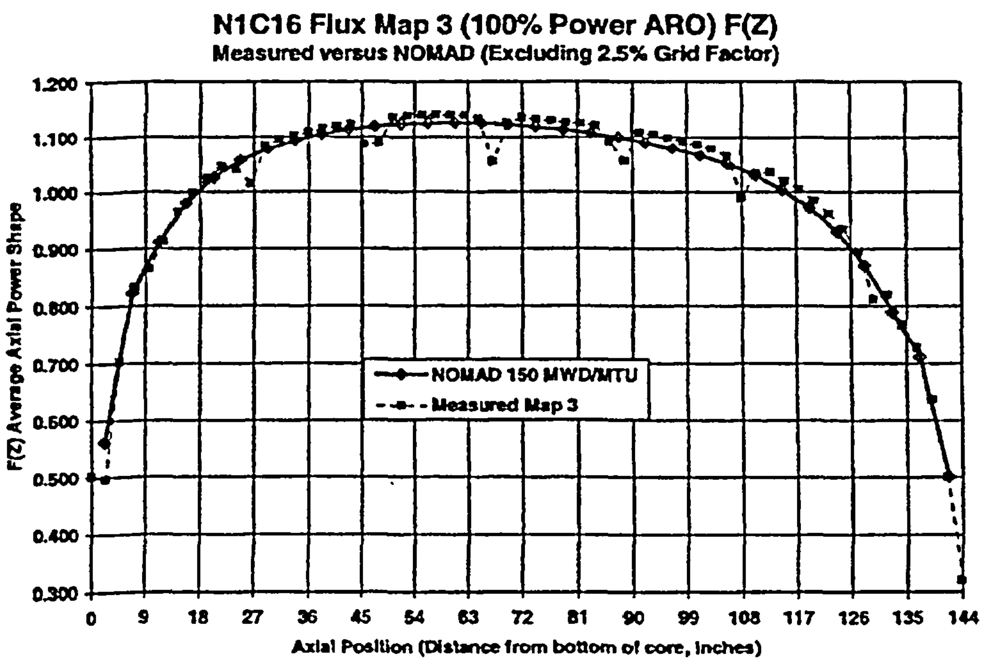
By letter dated December 2, 2002, VEPCO stated that the accuracy of the NOMAD model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the NOMAD predictions) that demonstrate the accuracy of this model.

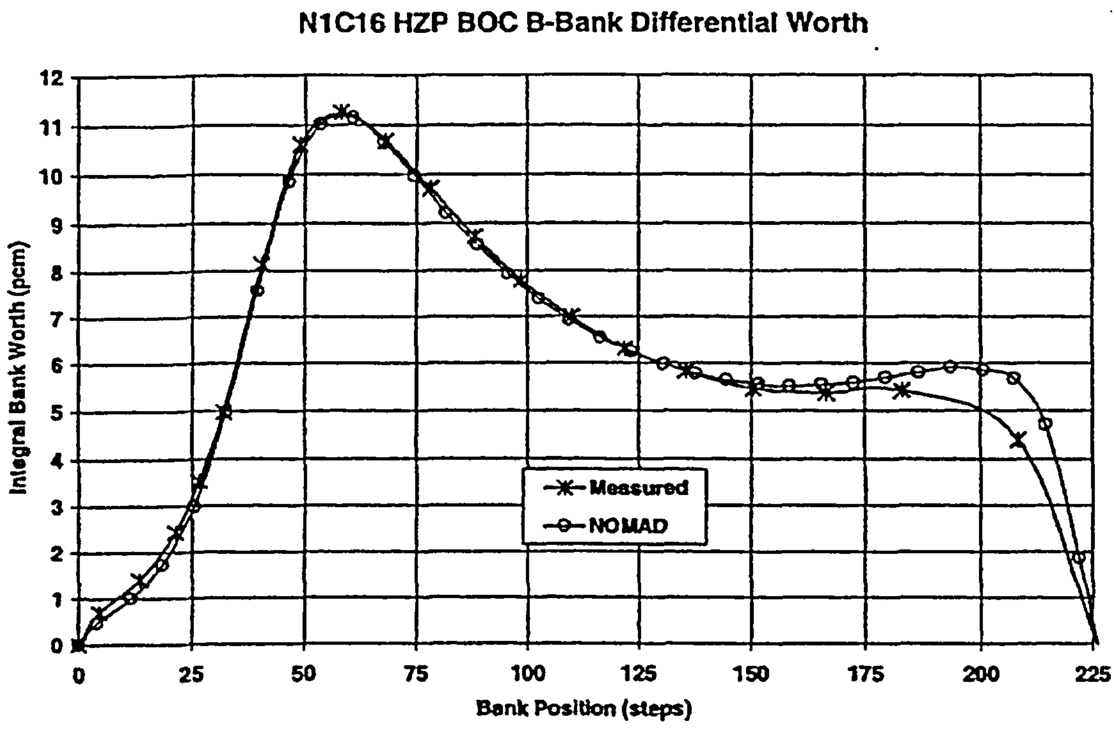
DOMINION RESPONSE TO QUESTION 1

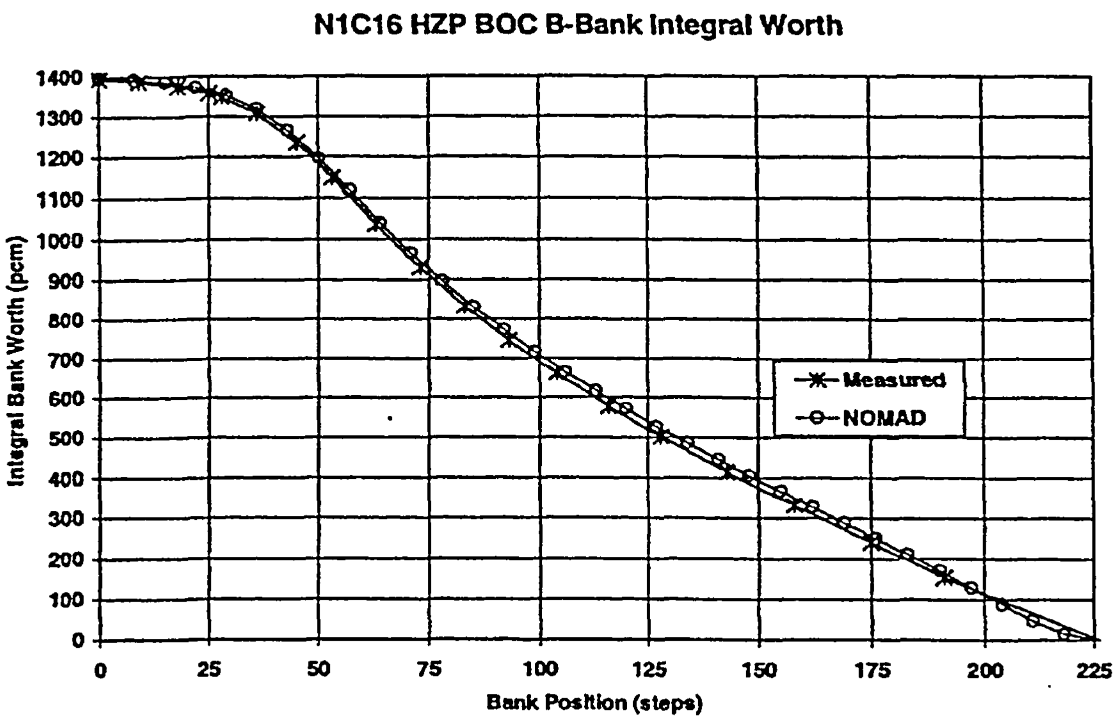
Verification of NOMAD accuracy comes primarily by extension through comparison to PDQ Two Zone model (Topical Report VEP-NAF-1) predictions during the NOMAD model setup process (see also the response to questions 3 and 7). The NOMAD model setup procedure provides specific power distribution and reactivity acceptance criteria for these comparisons that must be met. There are, however, a few direct comparisons to startup physics test data that can be made. The following results are from the N1C16 startup physics tests in October 2001.

N1C16 STARTUP PHYSICS TESTING RESULTS (October, 2001)

Parameter	Measured	Predicted	Difference	Nuclear Reliability Factor
Critical Boron Concentration (HFP, ARO, EQ XE) ppm	1405	1429	24	±50 ppm
HFP ARO EQ XE Axial Offset	-2.5	-3.0	-0.5%	N/A




N1C16 HZP BOC B-Bank Differential Worth
Integral Bank Worth (pcm)
Bank Position (steps)
Measured
NOMAD
0
1
2
3
4
5
6
7
8
9
10
11
12
0
25
50
75
100
125
150
175
200
225


N1C16 HZP BOC B-Bank Integral Worth
Integral Bank Worth (pcm)
Bank Position (steps)
Measured
NOMAD
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
0
25
50
75
100
125
150
175
200
225

NRC NOMAD QUESTION 2

There do not appear to be any limitations or restrictions associated with the use of NOMAD as described in this submittal. Please justify that NOMAD is applicable over all ranges of operation expected for North Anna and Surry.

DOMINION RESPONSE TO QUESTION 2

NOMAD is by design constrained by the limitations of the PDQ Two Zone Model. All cycle-dependent NOMAD input data comes from the PDQ Two Zone model, and the quality control process used to verify the NOMAD model for each core involves comparison to PDQ Two Zone model predictions. Therefore NOMAD should have the same restrictions and limitations as listed for the PDQ Two Zone model. The PDQ Two Zone model is restricted according to the following characteristics:

1) Geometry
   a) Square pitch fuel (cylindrical fuel pellets and rods)
   b) 15x15 or 17x17 design
   c) 5x5 mesh blocks per assembly (x-y)
   d) 26 axial nodes (22 in the fuel region)
   e) ¼ core or full core representation
2) Fuel Material
   a) Low enriched UO_2 (4.6 w/o U_{235} or less)
      i) Cores with fuel up to 4.45 w/o have been successfully modeled to date
      ii) Cross section behavior (enrichment trends and fidelity to CELL2) has been checked up to 4.6 w/o U_{235} for burnups up to 76 GWD/T.
   b) Fuel pin burnup of approximately 70 GWD/T has been achieved in PDQ Two Zone designed cores as part of a high burnup demonstration program.
3) Burnable poisons
   a) Discrete rods inserted into fuel assembly guide thimbles
      i) Both annular borosilicate glass and solid B4C in alumina designs have been well predicted throughout many cycles of operation
      ii) Both SS304 and zirconium based cladding has been used
   b) Modeling flexibility has been demonstrated for BP configuration (number of fingers, boron enrichment, poison length, and poison stack axial alignment)
4) Control rods
   a) Ag-In-Cd rods with stainless steel clad (extensive validation and experience)
   b) Hf metal rods in zirconium based clad have been used for vessel fluence reduction in Surry Unit 1
5) Fuel assembly
   a) Modeling flexibility has been demonstrated for Inconel and zirconium based grids of various designs and sizes

There are no current plans for fuel design, core design, or operating strategy changes that would exceed the design characteristics outlined above. There are fuel products in use in the industry which would be technically possible but impractical to model in the PDQ Two Zone and NOMAD models (such as fuel with integral poisons). No further development is planned for PDQ and NOMAD. In addition, the simplicity of the NOMAD control rod cross section model requires normalization for low temperature use (significantly below 547 °F). This precaution is listed in the NOMAD Code Manual. There are no current uses for NOMAD at low temperatures.

NRC NOMAD QUESTION 3

Please discuss the user-defined tolerances used in the Radial Buckling Coefficient model, including how they are calculated and used in the model. Also discuss the process in place that ensures that correct values are calculated and entered into the model by the user.

DOMINION RESPONSE TO QUESTION 3

The great majority of radial buckling effects are automatically captured (without any user intervention) via the data handling routines that collapse the 3-D PDQ Two Zone model data into 1-D NOMAD data. Design procedures indicate that reactivity agreement within 250 pcm of PDQ (HZP and HFP from BOC-EOC) is normally achieved using the "raw" (pre-buckling search) NOMAD model. Axial offset agreement within 2% is also typical. The buckling search can therefore be thought of as the means of capturing second and third order effects.

User defined tolerances control the rate and degree of convergence of the radial buckling search. Convergence is determined automatically in NOMAD by comparison of the NOMAD eigenvalue, peak nodal power, and individual node powers to the corresponding PDQ Two Zone values. Design procedures specify a standard set of convergence tolerances for use in the NOMAD model setup and review. Design procedures also require independent review of each NOMAD model setup prior to use in the core design process.

The values of the standard tolerance set are based on experience with previous NOMAD model setups (in particular the models which produced the benchmark data in Supplement 1 to VEP-NFE-1A) and represent the level of convergence normally achievable for a correctly constructed NOMAD model. These values were set at a level that would assure convergence consistent with Supplement 1 models, that would assure convergence as tight as reasonably achievable, but that could result in occasional minor non-convergence events.

If convergence is not achieved for a particular case, a warning message is printed that prompts a review of the model setup. One option available to the user is to change the rate of convergence (by changing the relaxation parameters) to reduce the chance of overshoot or undershoot. Cases of non-convergence are evaluated according to which parameter failed to converge and the degree of non-convergence involved. A large violation of a convergence tolerance is a good indication of a model error. Based on prior experience, non-convergence incidents are rare and of very small magnitude. Documentation for the most recent NOMAD model setups for North Anna and Surry indicates that convergence was achieved within the standard tolerances using the standard relaxation parameters.

There are other user-adjustable buckling parameters that are provided to accommodate the fact that the automated buckling search is only performed at HFP. Parameters are provided to improve axial offset and reactivity agreement between NOMAD and PDQ for lower power levels. In essence, these factors control the portion of the buckling search adjustments that are retained as power is reduced. Once again, a standard set of values is provided for use in the design procedures based on prior model setup experience. The adequacy of the standard values is verified directly by comparison of NOMAD and PDQ results at low power during the model setup process. A review of the history of NOMAD model setups revealed only one change to the standard values that has been implemented in order to meet the model acceptance criteria. Guidance for achieving an acceptable NOMAD model, including the user actions described above are incorporated in design procedures.

NRC NOMAD QUESTION 4

The xenon model in NOMAD allows a user-supplied multiplier to be applied to the xenon or iodine production terms. Please discuss the purpose of this multiplier and how the value is determined. Also discuss the process in place that ensures that correct values are calculated and entered into the model by the user.

DOMINION RESPONSE TO QUESTION 4

Iodine and xenon production multipliers were included in the NOMAD model for investigative purposes and possible future applications, but were never incorporated into the normal model design process. There are no current uses for these multipliers. Design procedures specify a value of 1.0 for these values. The xenon model requires very little user intervention and is verified by direct comparison to PDQ xenon concentration and xenon offset. Design procedures require independent review of each NOMAD model setup prior to use in the core design process.

NRC NOMAD QUESTION 5

The Control Rod Model requires several user input constants or multipliers. Please discuss the purpose of these user inputs, and the methods used to determine their values. Also discuss the process in place that ensures that correct values are calculated and entered into the model by the user.

DOMINION RESPONSE TO QUESTION 5

The Control Rod Model is very similar to the Radial Buckling Coefficient model in that a large majority of the NOMAD control rod information is obtained automatically from PDQ via data processing codes without any user-adjustable input. For the remaining effects, user input constants are provided in each of the following four categories:

A) Cusping corrections
B) Second order temperature or density effects
C) Geometry data (physical control rod overlap)
D) Worth normalization

The control rod cusping model accounts for the approximation made for control rod insertions in which the rodded/unrodded axial boundary occurs between nodal boundaries (partial insertions). For partial insertions NOMAD volume weights the control rod effects and applies the weighted values over the entire node. Without cusping corrections, the differential control rod worth shape exhibits a sawtooth behavior as the control rods are inserted in small steps. The cusping model corrects for this effect using two alternate approximations. The first alternative recognizes that the degree of cusping is a function of node size and insertion fraction. The second recognizes that the degree of cusping is a function of the local power gradient and insertion fraction. User input allows for the use and scaling of either alternative. Although cusping is not a significant practical problem due to the relatively small node size in NOMAD, standard input factors determined during the development of NOMAD were shown to significantly reduce the magnitude of cusping. These factors have not been changed since their development because neither the control rod type nor the NOMAD mesh structure have changed. Design procedures specify use of the recommended values for NOMAD model setup.

In the HZP-HFP operating range, control rod cross sections do not vary significantly. The small variation that exists is approximated by linear coefficients of moderator temperature or density. Based on PDQ Two Zone model control rod cross section data, a standard set of coefficients were developed during NOMAD development. These coefficients have not been changed because the control rod design has not changed. Design procedures specify use of the recommended values for NOMAD model setup. In the event of a control rod design change, detailed calculations are referenced in the design procedure that provide the techniques used to calculate these parameters.

User input is provided for the control rod ARO position and the normal operation control rod overlap. This input is based on actual core operating limits and specifications set each cycle.

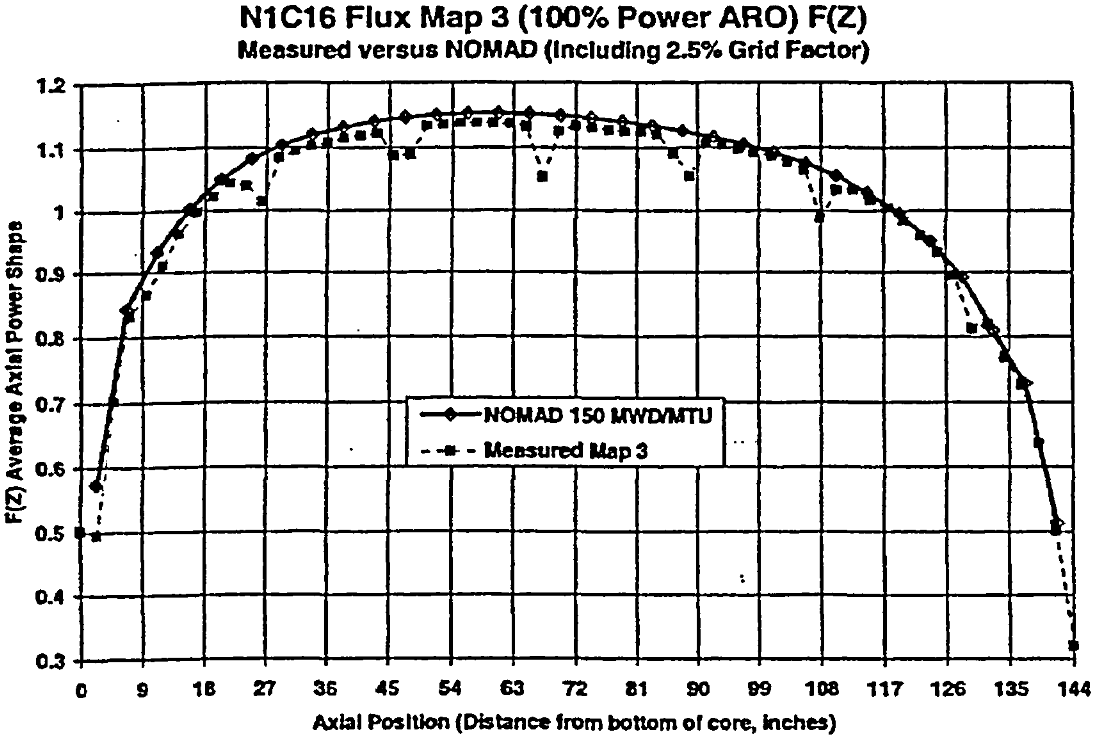
The final element of the control rod model is the ability to normalize bank worth to the PDQ Two Zone value. Although NOMAD was designed to produce acceptable control rod worth results without normalizing to PDQ, normalization is performed routinely for many design calculations to eliminate any difference between PDQ and NOMAD. In this way, calculations involving data from both models is completely consistent. In addition, normalization permits the modeling of non-physical part-length rods that are used to conservatively skew the axial power shape for certain types of calculation. Design procedures provide specific normalization instructions for each type of calculation. Design procedures also require independent review of each NOMAD model setup prior to use in the core design process.

NRC NOMAD QUESTION 6

In the $F_Q(z)$ x relative power calculations, a correction factor for grids is applied. Please discuss the method used to calculate these correction factors. Discuss how the correction factors change as the location of interest moves away from a grid location and provide typical values for these correction factors as a function of axial location.

DOMINION RESPONSE TO QUESTION 6

The grid factor is a constant multiplier of 1.025 that is conservatively applied to all axial locations rather than just between grids. The magnitude was retained from previous models but can be justified both qualitatively and quantitatively. A qualitative example is the power shape plot below. This is the same plot presented in the answer to NOMAD question 1, except that the grid factor has been applied. The predicted power shape effectively bounds the measured shape in this example, demonstrating that for this core and at this time in life, the grid factor is conservative.



Quantitatively, the grid factor can be determined from the mean of the Fz data presented in Table 3.0.3 of VEP-NFE-1A Supplement 1. Both the measured and predicted Fz shapes are normalized to an average value of 1.0 by definition. The Fz mean in Table 3.0.3 is the average difference between NOMAD and measured Fz at positions mid-way between grids for flux map data acquired during five different cycles. These are the axial positions where the NOMAD model exhibits the greatest degree of under-prediction due to the effect of the grids on the measured power shape. The mean difference of –2.4% is consistent with the magnitude of the NOMAD grid factor (1.025 or 2.5%).

NRC NOMAD QUESTION 7

Regarding the method of qualifying the NOMAD model, please address why data from only a few select operating cycles for North Anna, Unit 1, and Surry, Unit 2, were chosen for benchmarking purposes. Are the number of data points used for the various verifications adequate for a statistically significant decision?

DOMINION RESPONSE TO QUESTION 7

Unlike the PDQ Two Zone model, NOMAD is not developed sequentially by building on the depletion from the previous cycle. NOMAD is set up directly from the PDQ Two Zone model. Consequently, there was not a NOMAD model available for each historical cycle as a result of the development process. The primary use of NOMAD is for FAC (Final Acceptance Criteria) or RPDC (Relaxed Power Distribution Control) modeling, which involves the use of load follow transient axial power shapes. With this in mind, the cycles presented were chosen based on three criteria:

1) Availability of measured operational transient data.
2) Representation of the full range of cycle designs for Surry and North Anna.
3) Quantity of data similar to or greater than presented for the approved NOMAD model documented in VEP-NFE-1A.

The following Table summarizes the cycles used to support conclusions in VEP-NFE-1A and in Supplement 1.

Parameter	VEP-NFE-1A Cycles	Supplement 1 Cycles
Startup Physics Measurements	N1C2, N1C3, N1C4, N2C2, S1C6, S1C7	N1C3, N1C6, N1C9, S2C2, S2C11, S2C13
Operational Transients	N1C2, N1C3	N1C3, N1C6, N1C9, S2C2, S2C11, N1C11
Flux Maps (Fz and F_Q comparisons)	N/A*	N1C3, N1C6, N1C11, S2C2, S2C13
Estimated Critical Position (ECP; Mid-cycle HZP criticality measurements)	N/A	N1C9, S2C11, S2C13
FAC Analysis	N2C2, N1C4 (Verbal description of comparison to vendor model results)	S2C13 (Graphical comparison to approved NOMAD model F_Q envelope)
RPDC N(Z)	N/A (Pre-RPDC)	N1C11 (Graphical comparison to approved NOMAD model N(Z) function)

* BOC Fz plots were provided for 5 cycles (N1C2, N1C3, N1C4, N2C2, and S1C6)

As shown in the Table, Supplement 1 provides more NOMAD verification information than did the approved NOMAD Topical Report VEP-NFE-1A. There is no direct development of reliability factors in VEP-NFE-1A and no discussion of specific NOMAD reliability factors in the SER. The NOMAD SER cites comparisons to measurements, comparisons to higher order calculations (FLAME and PDQ), and the NOMAD normalization process as reasons for the approval. In particular, the normalization of NOMAD to FLAME is mentioned as a means of ensuring agreement with higher order calculations. NOMAD therefore was implicitly considered to share reliability factors with the models to which it is normalized.

The enhanced NOMAD model described in Supplement 1 can be supported based on this normalization argument and based on statistical comparisons to measured data. Design procedures specify these acceptance guidelines (comparison to PDQ Two Zone model predictions) to be met to support the conclusion that a NOMAD model has been set up properly:

1) Peak nodal power within 0.5% (HFP depletion)
2) All nodal powers within 2.5% (HFP depletion)
3) Equilibrium Xenon concentration within 0.5% (BOC and EOC)
4) Xenon offset within 0.2%
5) Axial offset within 2% (BOC-EOC, HZP and HFP)
6) Reactivity within 10 pcm (BOC-EOC, HFP)
7) Total power defect within 100 pcm (BOC, MOC, EOC)
8) HFP fuel temperature within 10 °R (BOC and EOC)
9) Calculation specific rod worth normalization

Because of these normalization requirements and the designed-in close connection between NOMAD and the 3D PDQ Two Zone model, the PDQ reliability factors (based on far more data) can be extended to the NOMAD model. This is analogous to the extension of FLAME reliability factors to the approved NOMAD version.

Although the number of observations in the measurement comparison data presented in Supplement 1 is not in all cases sufficient for a statistics-based determination of NOMAD uncertainty factors, the data presented is sufficient to demonstrate consistency with PDQ Two Zone Model comparisons. The conclusion in Supplement 1 that *"comparison of NOMAD uncertainty factors to Nuclear Reliability Factors.....verify.... the applicability of the NRF's for NOMAD calculations"* is not clearly qualified to indicate that the only parameters for which NOMAD uncertainty factors were directly statistically developed in Supplement 1 are Fz and F_Q. For other parameters, a better characterization is that comparison of NOMAD *results* to Nuclear Reliability Factors verify the accuracy of the NOMAD model and the applicability of the NRF's for NOMAD calculations.

For Fz and F_Q, a total of 134 observations were available for both, and the derived F_Q uncertainty factor is nearly identical to that calculated for the PDQ model (6.9% versus PDQ values of 6.7% for North Anna and 7.2% for Surry). The F_Q NRF of 1.075 conservatively bounds all these values.

The Table below compares PDQ Two Zone model and NOMAD statistics (differences between model predictions and measurements) for other parameters. PDQ statistics are contained in Topical Report VEP-NAF-1. Note that for critical boron and ITC, the sign of the NOMAD mean has been changed to reflect different definitions used in the respective reports and allow appropriate comparison to PDQ results. The range of NOMAD differences is bounded by the range of PDQ model differences, and the

NOMAD standard deviations are similar to or smaller than the corresponding PDQ standard deviations. The means show more variation, but are reasonable considering the sample sizes and the relative magnitude of the standard deviations. The comparison supports a conclusion that the PDQ Two Zone model reliability factors are appropriate for use with the closely related NOMAD model. Note that only the un-normalized (raw) rod worth results were presented in Supplement 1. The Table below also includes the normalized rod worth results (see the response to NOMAD question 5).

Comparison of NOMAD and PDQ Statistical Data

Parameter	Model	Number of observations	Mean	Standard Deviation	Maximum	Minimum
Control Rod Worth – Rod Swap	PDQ	95	1.8%	4.2%	11.5%	-11.3%
	NOMAD (raw)	25	2.99%	5.1%	11.4%	-7.8%
	NOMAD (normalized)	25	-0.1%	4.5%	7.6%	-8.1%
Control Rod Worth – Dilution	PDQ	62	-0.2%	4.8%	10.7%	-9.9%
	NOMAD (raw)	7	-0.6%	4.4%	7.1%	-6.7%
	NOMAD (normalized)	7	0.8%	4.1%	7.2%	-3.5%
Boron Worth	PDQ	30	-0.3	4.4%	7.4%	-6.1%
	NOMAD	6	-2.2%	2.3%	1.4%	-4.1%
HZP Critical Boron Concentration	PDQ	54	6 ppm	20 ppm	58 ppm	-30 ppm
	NOMAD	13	21 ppm	17 ppm	36 ppm	-17 ppm
HZP ITC (pcm/°F)	PDQ	57	-0.8	1.0	2.6	-2.9
	NOMAD	9	0.2	0.6	1.5	-0.5

NRC NOMAD QUESTION 8

Please discuss the methodology used to calculate each of the NOMAD NUF and indicate when NRC approval was obtained.

DOMINION RESPONSE TO QUESTION 8

As indicated in the response to NOMAD question 7, the only parameters for which NOMAD uncertainty factors were directly statistically developed in Supplement 1 are Fz and F_Q. The methodology is described briefly in Supplement 1, Section 3.1.4.1. This methodology is ultimately rooted in VEP-FRD-45A (SER date August 5, 1982) and is the same as described for the PDQ Two Zone model F_Q NRF. The only difference is that only the peak F_Q at each axial level can be used for the 1-D NOMAD comparisons rather than individual assembly F_Q's used for the 3-D PDQ model comparisons. A full discussion of the comparison and statistical methodology is provided in the response to PDQ question 4.

For all other parameters, uncertainty factors derived for other models were shown to be reasonable for use with NOMAD. VEP-FRD-45A summarizes the reliability factors derived for the PDQ Discrete model (VEP-FRD-19A, SER date May 18, 1981), the PDQ One Zone model (VEP-FRD-20A, SER date May 20, 1981), and the FLAME model (VEP-FRD-24A, SER date May 13, 1981). These same reliability factors were re-validated for the PDQ Two Zone model in VEP-NAF-1. Most of the approved reliability factors summarized in VEP-FRD-45A were approved not based on statistics, but on a combination of engineering arguments and consistency with uncertainty factors approved for other models (see the response to PDQ question 4). This is the approach taken in Supplement 1, except that more statistical data based on comparisons to measured data have been provided than in the approved NOMAD Topical. Dominion concurs with the use of these methods for determining appropriate reliability factors, and believes that the data presented in Supplement 1 is sufficient to support use of the reliability factors indicated.

NRC NOMAD QUESTION 9

Please discuss how the measured data used for statistical comparison to the NOMAD predicted values were obtained. How were uncertainties in the measured data addressed in the statistical analyses?

DOMINION RESPONSE TO QUESTION 9

Please refer to the response to PDQ question 5. Plant transient data (not used for statistical comparisons) was obtained either from plant computer records (delta-I based on ex-core detectors, calorimetric power based on the plant computer heat balance calculations, and control rod position indications) or from routine periodic measurements (critical boron concentration). No corrections for measurement bias or uncertainty were applied to the plant transient data.