June 11, 2003

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

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,

/RA/

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

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,

/RA/

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

DISTRIBUTION:

PUBLIC	ACRS	M. Kowal	PDII-1R/F	C. Gratton
E. Dunnington	K. Landis, RII	OGC	J. Nakoski	S. Monarque
F. Akstulewicz	U. Shoop	F. Orr	J. Uhle	

Accession Number: ML031621014

OFFICE	PDII/LA	PDII-1/PM	PDII-1/PM	SRXB/SC	PDII-1/SC	PDII
NAME	EDunnington	SMonarque	CGratton	FAkstulewicz	JNakoski	SMoore
DATE	06/02/03	05/30/03	05/30/03	06/03/03	06/03/03	06/11/03

OFFICIAL RECORD COPY

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.

RETRAN

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.

Principal Contributor: M. Kowal

Date: June 11, 2003

Mr. David A. Christian
Virginia Electric and Power Company

cc:
Ms. Lillian M. Cuoco, Esq.
Senior Counsel
Dominion Resources Services, Inc.
Millstone Power Station
Building 475, 5th Floor
Rope Ferry Road
Rt. 156
Waterford, Connecticut 06385

Mr. Richard H. Blount, II
Site Vice President
Surry Power Station
Virginia Electric and Power Company
5570 Hog Island Road
Surry, Virginia 23883-0315

Senior Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
5850 Hog Island Road
Surry, Virginia 23883

Chairman
Board of Supervisors of Surry County
Surry County Courthouse
Surry, Virginia 23683

Dr. W. T. Lough
Virginia State Corporation
Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, Virginia 23209

Robert B. Strobe, M.D., M.P.H.
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
P.O. Box 2448
Richmond, Virginia 23218

Mr. William R. Matthews
Vice President - Nuclear Operations
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060-6711

Office of the Attorney General
Commonwealth of Virginia
900 East Main Street
Richmond, Virginia 23219

Mr. Chris L. Funderburk, Director
Nuclear Licensing & Operations
Support
Innsbrook Technical Center
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Mr. David A. Heacock
Site Vice President
North Anna Power Station
Virginia Electric and Power Company
P. O. Box 402
Mineral, Virginia 23117-0402

Mr. C. Lee Lintecum
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
1024 Haley Drive
Mineral, Virginia 23117