March 12, 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 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, MB5436, AND MB5437)

Dear Mr. Christian:

By letter dated June 13, 2002, as supplemented by letter dated November 25, 2002, Virginia Electric and Power Company (VEPCO) requested approval of Topical Report DOM-NAF-1, entitled "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations."

The NRC staff has found that Topical Report DOM-NAF-1 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.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

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. 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

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT DOM-NAF-1

QUALIFICATION OF THE STUDSVIK CORE MANAGEMENT SYSTEM REACTOR

PHYSICS METHODS FOR APPLICATION TO

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 June 13, 2002, as supplemented by letter dated November 25, 2002, (References 1 and 2), Virginia Electric and Power Company (the licensee) requested approval 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." This approval would permit the use of the Studsvik Core Management System (Studsvik/CMS) code package to support the reload design analyses for North Anna and Surry Power Stations, Units 1 and 2. The Studsvik/CMS primarily consists of the CASMO-4 and SIMULATE-3 computer codes. This report demonstrates the validity and accuracy of the Studsvik/CMS package at North Anna and Surry for core reload design, core follow, and calculation of key core parameters for reload safety analysis.

2.0 REGULATORY EVALUATION

10 CFR 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 verify that the analyses remain bounding, the licensees confirm that the inputs to the safety analyses are conservative with respect to the current design cycle. These inputs are checked by using core design codes and methodologies.

3.0 TECHNICAL EVALUATION

Currently, the North Anna and Surry Power Stations use the NOMAD computer code and model, PDQ two-zone model, TIP/CECOR computer code package, and various methodologies. In its submittal, the licensee requested to replace its codes with the Studsvik/CMS. The CASMO-4, CMS-LINK, and SIMULATE-3 computer codes comprise the Studsvik/CMS package.

Enclosure

The CASMO-4 computer code is the Studsvik Scandpower, Inc. lattice code. The CASMO-4 computer code, a multi-group, two-dimensional transport theory code used for depletion and branch calculations for a single assembly, is used to generate the lattice physics parameters. These parameters include the cross sections, nuclide concentrations, pin power distributions and other nuclear data used as input to the SIMULATE-3 program for pressurized-water reactor core performance analyses. CASMO-4, which is an improved version of CASMO-3, incorporates the microscopic depletion of burnable absorbers into the main calculations, introduces a heterogeneous model for the two-dimensional calculation, and incorporates the use of the characteristics form for solving the transport equation.

Studsvik/CMS also supplies the SIMULATE-3 code. It is a two-group, 3-dimensional nodal program based on the NRC staff-approved QPANDA neutronics model that employs fourth-order polynomial representations of the intranodal flux distributions in both the fast and thermal neutron groups. This code is based on the modified coarse mesh (nodal) diffusion theory calculational technique, coupled with thermal hydraulic and Doppler feedback. This program explicitly models the baffle/reflector region, eliminating the need to normalize to higher-order fine mesh calculations. It also includes the following modeling capabilities: solution of the two-group neutron diffusion equation, fuel assembly homogenization, explicit reflector cross-section model, cross-section depletion and pin power reconstruction.

In order to ensure flux continuity at nodal interfaces and perform an accurate determination of pin-wise power distributions, SIMULATE-3 uses assembly discontinuity factors that are pre-calculated by CASMO-4. These factors are related to the ratio of the nodal surface flux in the actual heterogenous geometry to the cell averaged flux in an equivalent homogeneous model, and are determined for each energy group as a function of exposure, moderator density and control-rod-state.

The two-group model solves the neutron diffusion equation in three dimensions, and the assembly homogenization employs the flux discontinuity correction factors from CASMO-4 to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution. The flux discontinuity concept is also applied to the baffle/reflector region in both radial and axial directions to eliminate the need for normalization or other adjustment at the core/reflector interface.

The SIMULATE-3 fuel depletion model uses tabular and functionalized macroscopic or microscopic, or both, cross sections to account for fuel exposure without tracking the individual nuclide concentrations. Depletion history effects are calculated by CASMO-4 and then processed by the CMS-LINK code for generation of the cross-section library used by SIMULATE-3. SIMULATE-3 can be used to calculate the three-dimensional pin-by-pin power distribution in a manner that accounts for individual pin burnup and spectral effects. Furthermore, SIMULATE-3 calculates control rod worth and moderator, Doppler, and xenon feedback effects.

3.1 Model Benchmarking

The licensee’s June 13, 2002, submittal compared the CASMO-4 and SIMULATE-3 predictions of key physics parameters against plant data, critical experiments, and Monte Carlo calculations. As part of the development of the North Anna and Surry models, the licensee compared CASMO and Monte Carlo code calculations of reactivity worth for soluble boron, burnable poison rods, silver-indium-cadmium control rods, hafnium flux suppression rods, temperature defect, and Doppler defect.

For the SIMULATE-3 benchmarking, the licensee performed most of the calculations using full-core, 32 axial node, 2x2 X-Y mesh per assembly geometry. The comparison of SIMULATE predictions to measured data incorporates bias and uncertainty for both the predictions and the measured data. The licensee used statistical methods to account for these uncertainties. The licensee also used the CASMO, SIMULATE, and Monte Carlo code calculations in combination with the normalized flux map reaction rate comparisons to determine appropriate peaking factor uncertainty factors.

The licensee intends to use the CASMO-4 and SIMULATE-3 programs in licensing applications, including calculations for core reload design, core follow, and calculation of key core parameters for reload safety analyses of North Anna and Surry Power Stations, Units 1 and 2. The licensee used data from the North Anna, Units 1 and 2, operating cycles 1 through 15, and Surry, Units 1 and 2, operating cycles 1 through 17 to benchmark the CASMO-4 and SIMULATE-3 models. These cycles covered core design changes over more than 60 cycles of operation, including transitions in fuel enrichment, fuel density, fuel loading pattern strategy, spacer grid design and material, fuel vendor, core operating conditions (full-power average moderator temperature and rated thermal power), and burnable poison material and design. The fuel loading pattern variations include out-in and low-low-leakage designs, axially zoned fixed poison rods for reactor pressure vessel fluence reduction, transition to axially and radially zoned burnable poisons, and a range of operating cycle lengths from 202 to 582 effective full-power days with and without temperature and power coastdown. The licensee used critical boron concentration measurements, startup physics testing data, estimated critical position information, flux maps, and operational transient data to conduct model benchmarking. The good agreement between the measured and the calculated values presented in the June 13, 2002, submittal, is used to validate the licensee’s application of these computer programs for analysis of the North Anna and Surry Power Stations, Units 1 and 2. For the parameters compared, the licensee calculated a sample mean and standard deviation of the observed differences. In addition, the licensee determined bias to describe the statistical difference between predicted and reference values.

The licensee demonstrated that the CASMO-4 and SIMULATE-3 models, in conjunction with the indicated reliability factors, adequately represent the operating characteristics of the North Anna and Surry Power Stations. Additionally, the licensee did not change key aspects of its core design and analysis methodology, while maintaining code and quality assurance practices that provided assurances that future changes to the core, fuel, and burnable poison design will be modeled with accuracy and conservatism. Since the Studsvik/CMS package adequately represents the operating characteristics, and the licensee will maintain this code and methodology with its existing quality assurance practices, the NRC staff finds the use of the Studsvik/CMS package acceptable for the North Anna Power Station Units 1 and 2, and Surry Power Station Units 1 and 2.

However, the licensee did not include benchmarking or physics information for mixed oxide fuels at the North Anna and Surry Power Stations. Additionally, the NRC staff has yet to approve of the Studsvik/CMS code package for use with mixed oxide fuels. Therefore, the NRC staff does not approve this topical report for use with mixed oxide fuels. Similarly, introduction of new fuel designs or fuel designs that are significantly different from those analyzed in this topical report will require further validation of the above-stated physics methods for application to the North Anna and Surry Power Stations and will require NRC staff approval.

3.2 Statistics

The NRC staff reviewed VEPCO's application for statistical content. The statistical issues revolved around the 95/95 tolerance limit calculations for each parameter of interest. The calculations give 95-percent assurance that at least 95 percent of the population will not exceed the tolerance limit.

The procedure used in the tolerance limits depended on whether the data could be assumed to be distributed normally. The licensee used several established techniques for testing normality and assumed normality only if the majority of the techniques validated that assumption. This approach is acceptable to the NRC staff.

When the normal distribution was applicable, the licensee used the traditional one-sided tolerance calculations (Ref. 3). Otherwise, they used the nonparametric method of Summerville (Ref. 4). Both the parametric and the nonparametric approaches, in their proper context, are acceptable to the NRC staff.

4.0 CONCLUSION

The licensee has performed extensive benchmarking using the CASMO-4 and SIMULATE-3 methodology. Its effort consisted of conducting detailed comparisons of calculated key physics parameters with measurements obtained from over 60 operating cycles of the North Anna and Surry Power Stations, Units 1 and 2. These results were then used to determine the set of 95/95 tolerance limits for application to the calculation of the stated physics parameters.

Based on the review of the analyses and results presented in the June 13, 2002, submittal, the NRC staff has concluded that the CASMO-4 and SIMULATE-3 methodology, as validated by the licensee, can be applied to the North Anna and Surry steady-state physics calculations for reload applications as described in the above technical evaluation.

Based on the considerations discussed above, the NRC staff has concluded 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) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 CONDITIONS AND LIMITATIONS

The NRC staff's approval of this topical report does not apply to the use of mixed oxide fuel. Furthermore, introduction of significantly different or new fuel designs will require further validation of the above-stated physics methods for application to North Anna and Surry by the licensee and will require NRC staff approval.

6.0 REFERENCES

1. Letter from Leslie N. Hartz, Vice President Nuclear Engineering, Virginia Electric and Power Company to the USNRC, “Virginia Electric and Power Company Request for Approval 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,” June 13, 2002.

2. Letter from Leslie N. Hartz, Vice President Nuclear Engineering, Virginia Electric and Power Company to the USNRC, “Virginia Electric and Power Company Request for Additional Information, Topical Report DOM-NAF-1, Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations,” November 25, 2002.

c. R. E. Odeh and D. B. Owen, “Tables for Normal Tolerance Limits,” Marcel Dekker, Inc., New York, 1980.

d. Summerville, “Annals of Mathematical Statistics,” Vol. 29, p. 599, 1958.

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