August 30, 2007

Mr. David A. Christian
President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - SAFETY EVALUATION FOR TOPICAL REPORT DOM-NAF-5 (TAC NO. MD2829)

Dear Mr. Christian:

On August 16, 2006, as supplemented on December 6, 2006, April 16, May 4, and June 12, 2007, Dominion Energy Kewaunee submitted Topical Report (TR) DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis methods to Kewaunee Power Station (KPS)."

The Nuclear Regulatory Commission (NRC) staff has found that DOM-NAF-5 is acceptable for referencing in licensing applications for KPS to the extent specified and under the limitations delineated in the TR and in the enclosed safety evaluation (SE). The SE defines the basis for acceptance of the TR.

The NRC staff's acceptance applies only to material provided in the subject TR. The staff does not intend to repeat its review of the acceptable material described in the TR. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards. If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Dominion Energy Kewauee will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosure:
Safety Evaluation

cc w/encls: See next page

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cc:

Resident Inspectors Office
U.S. Nuclear Regulatory Commission
N490 Hwy 42
Kewaunee, WI 54216-9510

Ms. Leslie N. Hartz
Dominion Energy Kewaunee, Inc.
Kewaunee Power Station
N 490 Highway 42
Kewaunee, WI 54216

Mr. Chris L. Funderburk
Director, Nuclear Licensing and
  Operations Support
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

Mr. Thomas L. Breene
Dominon Energy Kewaunee, Inc.
Kewaunee Power Station
N490 Highway 42
Kewaunee, WI 54216

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO FACILITY OPERATING LICENSE NO. DPR-43

DOMINION ENERGY KEWAUNEE, INC.

KEWAUNEE POWER STATION

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated August 16, 2006 (Reference 1), as supplemented on December 6, 2006, April 16, May 4, and June 12, 2007 (Agencywide Documents Access Management System (ADAMS) Accession Nos. ML070120088, ML063410177, ML071060392, ML071270780, and ML071630521, respectively), Dominion Energy Kewaunee, Inc. (the licensee) requested approval of Topical Report DOM-NAF-5, "Application Of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)." The report describes the various in-scope design and analysis methodologies and documents the assessments of the applicability of these methodologies to KPS. The approval by the Nuclear Regulatory Commission (NRC) would permit the licensee to subsequently request an amendment to the Technical Specifications (TSs) to apply the Dominion Energy (Dominion) nuclear core design and safety analysis methods to the KPS design and licensing analyses.

The nuclear core design methods addressed by the report include the Reload Nuclear Design Methodology, Relaxed Power Distribution Control (RPDC) Methodology, and the Studsvik Core Management System (CMS) Reactor Physics Methods. The safety analysis methods covered by the report include the Vepco Reactor System Transient Analyses using the RETRAN Computer Code, Statistical Departure from Nucleate Boiling ration (DNBR) Evaluation Methodology, and the Reactor Core Thermal-Hydraulics using the VIPRE-D Computer Code. Attachments A (Reference 2) and B (Reference 3) to DOM-NAF-5 provide supplemental material documenting the applicability of Studsvik CMS Reactor Physics methods and Dominion's RETRAN methods to KPS.

2.0 REGULATORY EVALUATION

The NRC staff used the following requirements and guidance documents in evaluating the licensee's amendment request:

Section 50.34, "Contents of Applications; Technical Information," of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that Safety Analysis Reports 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, they confirm that the inputs to the safety analyses are conservative with respect to the current design cycle. These inputs are checked using analytical models, and if key safety analysis parameters are not bounded, further analysis of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

3.0 TECHNICAL EVALUATION OF ANALYTICAL METHODS AND APPLICABILITY

The core reload design and safety analysis process is currently performed by the KPS fuel supplier (Westinghouse), whereas other Dominion nuclear plants rely on the Dominion nuclear core design and safety analysis methods. In TR DOM-NAF-5, the licensee has proposed to apply the Dominion nuclear core design and safety analysis methods to KPS, although the KPS fuel supplier will continue to license the fuel design, perform fuel rod design analysis for reload fuel performance assessment and perform certain specific safety analyses, such as small-break and large-break loss-of-coolant accident (LOCA) analyses. The Dominion methods detailed herein are to be applied to KPS in a manner consistent with the conditions and limitations of this safety evaluation report (SER), other relevant NRC SERs and the relevant Dominion TRs.

3.1 Reload Nuclear Design Methodology

The reload nuclear design methodology in Dominion TR VEP-FRD-42, Revision 2.0-A, “Reload Nuclear Design Methodology” (Reference 6), consists of the analytical models, methods, reload design and reload safety analysis, and an overview of analyzed accidents. It is an iterative process that involves the determination of a core loading pattern that fulfills cycle energy requirements and the demonstration that the plant with the reload core satisfies the constraints of the plant design basis and safety analysis limits.

The reload safety evaluation uses a bounding analysis concept in which key analysis parameters with limiting directions are identified such that, if all key analysis parameters are conservatively bounded, a reference safety analysis is applicable and no further analysis is necessary. If any values are not bounded, further analysis of the transient or accident in question is performed, the applicable safety analyses are revised, or changes are made in the operating requirements to satisfy applicable safety analysis criteria. The safety analysis process typically consists of steady state nuclear calculations used to derive the core physics related key analysis parameters as well as a dynamic accident analysis that utilizes these parameters to determine the accident result.

The Dominion nuclear design methodology and the current KPS reload design methodology are similar and share a common basis in Westinghouse TR WCAP-9272, “Westinghouse Reload Safety Evaluation” (Reference 7). Specific differences in nuclear steam supply system (NSSS), reactor protection system (RPS), and fuel features between KPS and other Dominion Westinghouse units are capable of being reflected via modeling inputs in VEP-FRD-42 analytical methods, without changing the methodology. Implementation of this TR at KPS will be done in a manner consistent with the conditions and limitations identified in DOM-NAF-5. The staff finds the reload nuclear design methodology applicable to KPS as detailed in DOM-NAF-5.

3.2 Relaxed Power Distribution Control (RPDC) Methodology

The RPDC methodology, VEP-NE-1 (Reference 8), is a Dominion method for axial power distribution control with a variable axial flux difference (delta-I) band that provides an increasing delta-I band with decreasing power in order to maintain approximately constant analysis margin at all power levels. RPDC provides several operational benefits, such as increased ability to return to power after a trip, reduced control rod motion to compensate for delta-I band restrictions, and reduced reactor coolant system (RCS) boration and dilution requirements.

The RPDC analysis determines acceptable delta-I bands to maintain design bases margin. The process consists of: the generation of power shapes that bound the delta-I range; the selection of delta-I bands such that all bands satisfy the core operating limits report (COLR) height dependent hot channel factor, FQ(Z), limit with verification that the proposed delta-I bands satisfy LOCA FQ and loss of flow accident (LOFA) thermal-hydraulic evaluations; the examination of limiting Condition II events; the verification that over-power delta-temperature (OPΔT) and over-temperature delta-T (OTΔT) limits are conservative; and N(Z) functions are formulated to support the implementation of FQ TSs surveillance.

Similarity between the Dominion RPDC and Westinghouse, Combustion Engineering (CE), and Exxon-relaxed axial power distribution control methodologies is noted in DOM-NAF-5 and the VEP-NE-1 SER. DOM-NAF-5 also notes that the cooldown transient assumption differs between the RPDC methodology (20 °F) and the Westinghouse relaxed axial offset (RAOC) methodology currently used for KPS (30 °F); the larger value will be used unless a KPS-specific analysis demonstrates that a plant trip will occur before a 30 °F cooldown.

The VEP-NE-1 SER states that the RPDC approach is an acceptable methodology for use with reload cores similar to those of Surry Power Station (SPS) and North Anna Power Station (NAPS) because: approved methodologies are used for the analyses supporting RPDC; justification of uncertainties is provided; and the impact of cycle specific variations on the delta-I power domain, OPΔT and OTΔT trip setpoints, and other safety analyses are evaluated on a reload basis. In light of the similarity of the NSSS, RPS, and fuel design at KPS, SPS, and NAPS, the NRC staff finds that the RPDC approach is an acceptable methodology for use at KPS as documented in DOM-NAF-5.

3.3 Studsvik Core Management System Reactor Physics Methods

The Studsvik CMS reactor physics code package, detailed in TR DOM-NAF-1, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," (Reference 9), consists of CASMO-4, SIMULATE-3, and CMS-LINK. CASMO-4 is a multi-group two-dimensional transport theory code for depletion and branch calculations for a single assembly that is used to generate the lattice physics parameters, including cross sections, nuclide concentrations, pin power distributions and other nuclear data, which are used as inputs to SIMULATE-3. SIMULATE-3 is a two-group, three-dimensional modified coarse-mesh nodal diffusion theory code with coupled thermal-hydraulic and Doppler feedback. CMS-LINK is a linking code that processes CASMO-4 card image files into a binary formatted nuclear data library for use by SIMULATE-3.

Dominion uses the Studsvik CMS package for startup physics testing, RPDC, and licensing applications, including core reload design, core operation, and key core parameters for reload safety analyses.

The Studsvik CMS benchmarking data provided in DOM-NAF-1 was based on the 15x15 and 17x17 fuel designs used at SPS and NAPS respectively, while KPS currently uses 14x14 fuel. In addition, DOM-NAF-1 SER limits the use of DOM-NAF-1, prohibiting its application to "significantly different or new fuel designs." Since this restriction is not clearly defined and in light of the absence of benchmarking data to 14x14 fuel, the KPS CMS models have been validated by comparison to benchmarks from both higher order Monte Carlo neutron transport calculations and reactor measurements from 10 cycles of operation spanning transitions in fuel enrichment, fuel density, spacer grid design, fuel vendor, core operating conditions and burnable poison design. These benchmarking results, as provided in Attachment A of DOM-NAF-5, are consistent with the staff approved methodology described in DOM-NAF-1. Thus, the NRC staff finds that the Studsvik CMS methodology as detailed in DOM-NAF-1 and DOM-NAF-5 is applicable to KPS.

3.4 Reactor System Transient Analyses using RETRAN

Dominion uses RETRAN to perform transient thermal-hydraulic analyses of the NSSS for best-estimate (e.g. training simulator validation) and licensing applications (e.g. reload core safety analysis), as detailed in TR VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code" (Reference 10). RETRAN calculates general system parameters as a function of time and boundary conditions for input into more detailed calculations of departure from nucleate boiling (DNB) or other thermal and fuel performance margins.

The licensee performed transient analyses to confirm the adherence of reload core design limits to the bounds established by the reference analysis of record parameter values, as well as to verify that the core is acceptable from a safety operational point of view.

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

The Dominion KPS RETRAN models have been validated by selecting representative transient events and comparing the results of the KPS RETRAN models to the vendor RETRAN model that was used to perform the current USAR analyses. This approach is similar to the one taken in VEP-FRD-41. The results of this analysis, as provided in Attachment B of DOM-NAF-5, show that the Dominion KPS RETRAN model compares favorably to the vendor RETRAN model for the selected transients, and the differences can be understood based on differences in noding, inputs, or other modeling assumptions. The NRC staff finds that the RETRAN methodology as detailed in DOM-NAF-5 and VEP-FRD-41 is applicable to KPS.

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

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

If a reanalysis is performed, the results are compared to the appropriate analysis acceptance criteria. The reload evaluation process is complete if the acceptance criteria are met, and internal documentation of the reload evaluation is provided for the appropriate Dominion safety review. If the analysis acceptance criteria are not met, more detailed analyses and/or TS changes may be required to meet the acceptance criteria.

3.5 Statistical DNBR Evaluation Methodology

3.5.1 Analytical Methods

Topical Report DOM-NAF-5 details the events and analyses that will use the statistical DNBR evaluation methodology as well as those events that will use the deterministic models.

Topical Report VEP-NE-2, "Statistical DNBR Evaluation Methodology" (Reference 11), describes Dominion's methodology for statistically treating several of the important uncertainties in the DNBR analysis. The statistical DNBR evaluation methodology is used to determine a plant-specific and fuel-specific statistical DNBR limit. This limit DNBR combines the core heat flux (CHF) correlation uncertainty with DNBR sensitivities to uncertainties in key DNBR analysis input parameters. The statistical combination of some of these uncertainties permits a more realistic combination of the independent uncertainties and, thus, provides a more realistic evaluation of DNBR margin. The statistical DNBR evaluation methodology allows thermal-hydraulic evaluations to be performed using nominal operating conditions as opposed to deterministic initial conditions (nominal conditions plus evaluated uncertainty).

The statistical DNBR evaluation methodology is typically applied to all Condition I and II DNB events (except rod withdrawal from subcritical, RWSC), and to the LOFA analysis, the locked rotor accident and the single rod cluster control assembly withdrawal at power (SRWAP). The events modeled statistically (see Table 3.5.1 of Reference 1) are limited by the statistical design limits (SDLs) evaluated in the implementation of the statistical DNBR evaluation methodology for KPS, dated May 4, 2007 (Reference 4). In addition, there are events that will be evaluated with deterministic models. These events will be initiated from bounding operating conditions (nominal value), with appropriate uncertainty added to these nominal values. The events modeled deterministically are limited by the deterministic design limits (DDLs) stated in DOM-NAF-2 (Reference 12).

In its May 4, 2007, letter (Reference 4), Dominion submitted its KPS-specific statistical DNBR methodology analysis. The May 4, 2007, report supports the application of the NRC-approved TR stated above. In this plant-specific report, Dominion provided the technical basis and documentation necessary to evaluate the plant specific application of the VEP-NE-2-A methods to KPS. In its specific application analysis, Dominion used the VIPRE code with the Westinghouse WRB-1 CHF correlation for the thermal-hydraulic analysis of the Westinghouse 14x14 (422V+) fuel assemblies at KPS. The same report also provides documentation that the core safety limits and protection functions, such as the OTΔT, OPΔT, do not require revision as a consequence of this implementation.

3.5.2 Uncertainty Analysis

Consistent with VEP-NE-2-A, (Reference 11), various plant parameters were selected as the statistically treated parameters in the implementation analysis. The magnitudes and functional forms of the uncertainties for the statistically treated parameters were derived in a rigorous analysis of plant hardware, measurement and calibration procedures, and have been summarized in Table 3.2-1 of the May 4, 2007, submittal (Reference 4).

The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and inlet temperature were quantified using all sensor, rack, and other components of a total uncertainty and combined in a manner consistent with their relative dependence or independence. Westinghouse quantified these uncertainties for Kewaunee's transition to Westinghouse's 14x14 (422V+) fuel. Total uncertainties were quantified at the 2 sigma level, corresponding to two-sided 95% probability.

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

The magnitude and distribution of uncertainty on the enthalpy rise hot channel factor, $F_{\Delta H,}$ was quantified as a normal probability distribution with a magnitude of 3.0%. The statistical DNBR evaluation methodology (Reference 4) treats the $F_{\Delta H}$ uncertainty as a uniform probability distribution.

3.5.3 Verification of Nominal Set-points

Condition 1 of the NRC's SER for VEP-NE-2-A (Reference 11) requires that the nominal statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may be applied.

Consequently, in the May 4, 2007, submittal (Reference 4), the licensee provided analysis to demonstrate that S_{total} (the total DNBR standard deviation) as calculated in the TR is maximized for any conceivable set of conditions at which the core may approach the SDL. To this end, the licensee performed a regression analysis using as dependent variable the un-randomized DNBR standard deviations at each nominal statepoint (i.e. the raw MDNBR results obtained from a Monte Carlo simulation). The nominal statepoint pressures, inlet temperatures, powers and flow rates are used as the independent variable. The licensee stated that an evaluation of all the

data, linear fits, and regression coefficients indicated that there were no discernible trends in the database. Consequently, the licensee concluded that the total standard deviation had been maximized for any conceivable set of conditions at which the core may approach the SDL, and that the selected nominal statepoints provide a bounding standard deviation for any set of conditions to which the methodology may potentially be applied. The NRC staff finds the licensee's results and conclusion acceptable.

3.6 Reactor Core Thermal-Hydraulics using VIPRE-D Computer Code

The VIPRE (Versatile Internals and Components Program for Reactors) computer code is a reactor core thermal-hydraulics code developed by Battelle Pacific Northwest Laboratories. VIPRE is used to accurately calculate reactor coolant conditions to assure that the DNBR design limit is maintained.

The reactor core thermal-hydraulics code VIPRE-D, as described in TR DOM-NAF-2, is a Dominion-modified version of the VIPRE-01, MOD-02.1, which has been adapted to accommodate the various fuel designs used at Dominion nuclear power stations by incorporating vendor proprietary CHF correlations. The input and output has also been customized to incorporate it into the Dominion thermal hydraulic methodology.

VIPRE-D was approved by the NRC staff for pressurized-water reactor (PWR) licensing calculations up to the CHF using approved CHF correlations in accordance with the conditions and limitations listed in the SERs of DOM-NAF-2 and Electric Power Research Institute Report, NP-2511-CCM. In addition, VIPRE-D must be applied in a manner consistent with plant-specific and fuel-specific application conditions and limitations outlined in DOM-NAF-5. The NRC staff finds that the VIPRE-D thermal-hydraulics analysis methodology is applicable to KPS.

4.0 CONCLUSION

The NRC staff has reviewed Dominion's submittals and supporting documentation and finds the proposed use of Dominion nuclear core design and safety analysis methods at KPS to be acceptable. As such, TR DOM-NAF-5 is acceptable for use in licensing applications at KPS.

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 REFERENCES

1. Dominion Energy Kewaunee, Inc. (DEK) letter, Gerald T. Bischof to NRC, request for Approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” August 16, 2006.

2. DEK letter, Gerald T. Bischof to NRC, “Attachment A to Dominion Energy Kewaunee, Inc., Kewaunee Power Station Request for Approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” December 6, 2006.

3. DEK letter, Gerald T. Bischof to NRC, “Attachment B to Dominion Energy Kewaunee, Inc., Kewaunee Power Station Request for Approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” April 16, 2007.

4. DEK letter, Gerald T. Bischof to NRC, “Dominion Energy Kewaunee, Inc., Kewaunee Power Station Request for Approval of Topical Report DOM-NAF-5, “Implementation of the Dominion Statistical DNBR Methodology with VIPRE-D/WRB-1 at Kewaunee Power Station (KPS),” May 4, 2007.

5. Dominion letter, Leslie N. Hartz to NRC, “Virginia Electric and Power Company Responses to NRC questions regarding Kewuanee request for approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” June 12, 2007.

6. Dominion Topical Report VEP-FRD-42 Revision 2.0-A, “Reload Nuclear Design Methodology,” August 2003.

7. Westinghouse Topical Report WCAP-9272-P-A, 'Westinghouse Reload Safety Evaluation Methodology," (Proprietary), March 1978.

8. Dominion Topical Report VEP-NE-1, Rev. 0.1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," August 2003.

9. Dominion Topical Report DOM-NAF-I, Rev. 0.0-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," June 2003.

10. Dominion Topical Report VEP-FRD-41, Rev. 0.1-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code," June 2004.

11. Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

12. Dominion Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," September 2004.

Principal Contributor: A. Attard

Date: August 30, 2007