VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 21, 2003

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

Serial No.	03-183
NL&OS/ETS	
Docket Nos.	50-338/339
	50-280/281
License Nos.	NPF-4/7
	DPR-32/37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
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

In an October 10, 2001 letter (Serial No. 01-623) Virginia Electric and Power Company (Dominion) submitted Reload Nuclear Design Methodology Topical Report, VEP-FRD-42 Revision 2, for NRC review. This topical report was revised to support the transition to Framatome-ANP Advanced Mark-BW fuel at North Anna. Revision 2 of VEP-FRD-42 addresses the restriction in the SER for Revision 1 that 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." Since the initial submittal of revision 2 to the topical report, additional information has been requested by and provided to the NRC staff in letters dated May 13, 2002 (02-280) and December 2, 2002 (02-662). The NRC Staff has requested additional information in a February 26, 2003 letter. The attachments to this letter provide the additional information to complete the NRC staff review of VEP-FRD-42, Revision 2.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

A001

cc: U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center, Suite 300
4201 Dominion Blvd.
Glen Allen, Virginia 23060

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.

BIT Mixing Model

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.

Technical Evaluation Report "Implications of these Limitations"

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.

DOMINION RESPONSE TO QUESTION 1b

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 $F_{\Delta H}$ (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

PREDICTED
MEASURED
PCT DIFFERENCE

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.246	0 275	0.242						
							0.249	0.280	0.246						
							1.0	1.8	1.6						
2					0.331	0.649	1.053	0.836	1.045	0.647	0.331				
					0.336	0.656	1.061	0.844	1.062	0.674	0.338				
					1.5	1 0	0.8	1.0	1.6	4.3	2.2				
3				0.365	1.179	1.299	1.170	1.301	1.168	1.299	1.183	0.368			
				0.381	1.200	1.313	1.177	1.301	1.175	1.312	1.196	0.379			
				4.3	1.8	1.1	0.6	0.0	0 6	1.0	1.1	3.0			
4			0.368	0.898	1.333	1.323	1.305	1.243	1.304	1.324	1.338	0.897	0.365		
			0.381	0.919	1.358	1.335	1.309	1.244	1.303	1.318	1.343	0.908	0 369		
			3.6	2.3	1.9	0.9	0.3	0.0	-0.1	-0.4	0 4	1.2	1.1		
5		0.356	1.204	1.346	1.221	1.262	1.168	1.236	1.168	1.261	1.220	1.339	1.197	0.355	
		0.367	1.259	1.370	1.228	1.265	1.168	1.233	1.164	1.258	1.222	1.358	1.212	0 361	
		3.1	4.6	1.8	0.6	0.2	0.0	-0.2	-0.3	-0.3	0.1	1.4	1.2	1.9	
6		0.665	1.321	1.334	1.266	1.017	1.145	1.071	1.143	1.017	1.265	1.332	1.319	0.665	
		0.668	1.334	1.338	1.252	1.012	1.139	1.065	1.133	1.012	1.261	1.333	1.330	0.677	
		0.5	0.9	0.3	-1.1	-0.5	-0.6	-0.5	-0.8	-0.4	-0.3	0.1	0.8	1.9	
7	0.258	1.070	1.186	1.317	1.174	1.145	1.038	1.005	1.038	1.148	1.174	1.316	1.187	1.074	0.259
	0.256	1.060	1.166	1.306	1.164	1.135	1.027	0.993	1.018	1.135	1.163	1.294	1.195	1.108	0.267
	-1.0	-0.9	-1.7	-0.8	-0.9	-0.9	-1.1	-1.2	-1.9	-1.1	-0.9	-1.7	0.7	3.2	3.1
8	0.284	0.853	1.320	1.255	1.243	1.074	1.005	0.996	1.005	1.074	1.243	1.255	1.320	0.853	0.284
	0.282	0.845	1.298	1.244	1.232	1.065	0.994	0.984	0.991	1.063	1.233	1.251	1.333	0.879	0.293
	-1.0	-1.0	-1.7	-0.9	-0.9	-0.9	-1.1	-1.2	-1.4	-1.0	-0.7	-0.3	1.0	3.0	2.9
9	0.259	1.074	1.187	1.316	1.174	1.148	1.038	1.005	1.038	1.145	1.174	1 317	1.186	1.070	0.258
	0.258	1.069	1.181	1.310	1.172	1.139	1.025	0.991	1.024	1.133	1.167	1.319	1.201	1.097	0.264
	-0.6	-0.5	-0.5	-0.5	-0.2	-0.8	-1.2	-1.4	-1.3	-1.1	-0.6	0.2	1.3	2.5	2.4
10		0.665	1.319	1.332	1.265	1.017	1 143	1.071	1.145	1.017	1.266	1.334	1.321	0.665	
		0.665	1.326	1.330	1.259	1.007	1.123	1.053	1.127	1.003	1.255	1.339	1.344	0.691	
		0.1	0.6	-0.1	-0.5	-0.9	-1.7	-1.7	-1.6	-1.4	-0.9	0.4	1.8	3.8	
11		0.355	1.197	1.339	1.220	1.261	1.168	1.236	1.168	1.262	1.221	1.346	1.204	0.356	
		0.356	1.203	1.337	1.211	1.246	1.147	1.211	1.146	1.229	1.198	1.361	1.235	0.366	
		0.3	0.5	-0.1	-0.7	-1.2	-1.8	-2.0	-1.9	-2.6	-1.9	1.1	2.6	3.0	
12			0.365	0.897	1.338	1.324	1.304	1.243	1.305	1.323	1.333	0.898	0.368		
			0.373	0.897	1.329	1.307	1.276	1.218	1.286	1.309	1.334	0.939	0.407		
			2.4	-0.1	-0.7	-1.3	-2.2	-2.0	-1.5	-1.1	0.1	4.5	10.6		
13				0.368	1.183	1.299	1.168	1.301	1.170	1.299	1.179	0.365			
				0.367	1.179	1.286	1.149	1.270	1.163	1.292	1.180	0.372			
				-0.3	-0.3	-1.0	-1.6	-2.4	-0.7	-0.5	0.1	1.9			
14					0.331	0.647	1.045	0.836	1.053	0.649	0.331				
					0.342	0.644	1.037	0.834	1.077	0.653	0.332				
					3.4	-0.4	-0.7	-0.1	2.3	0.6	0.2				
15							0.242	0.275	0.246						
							0.243	0.276	0.251						
							0.3	0.2	1.8						

STANDARD DEVIATION =1.235

AVERAGE PCT DIFFERENCE = 1.3

SUMMARY

MAP NO: N1-16-01

DATE: 10/10/01

POWER: 29%

CONTROL ROD POSITIONS:

D BANK AT 150 STEPS

F-Q(Z) = 2.108

F-DH(N) = 1.546

F(Z) = 1.283

BURNUP = 5.0 MWD/MTU

CORE TILT:

NW 1.0037	NE 1.0031
SW 0.9922	SE 1.0010

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	

SUMMARY

MAP NO: N1-16-02 DATE: 10/11/01 POWER: 74%

CONTROL ROD POSITIONS: F-Q(Z) = 1.848 CORE TILT:

D BANK AT 192 STEPS F-DH(N) = 1.451

NW 1.0014	NE 1.0039
SW 0.9933	SE 1.0014

F(Z) = 1.184

BURNUP = 24.0 MWD/MTU A.O. = 0.088

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

PREDICTED
MEASURED
PCT DIFFERENCE

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.283	0.330	0.279						
							0.283	0.334	0.281						
							0.1	1.3	0.9						
2					0.342	0.662	1.087	0.963	1.080	0.660	0.342				
					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			
				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		
			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	
		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	
		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	1.176	1.195	1.100	1.067	1.100	1.197	1.176	1.270	1.165	1.101	0.295
	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
	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
	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	
		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	
		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		
			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			
				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				
					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							0.279	0.330	0.283						
							0.279	0.330	0.287						
							0.2	0.0	1.4						

STANDARD DEVIATION =0.906

AVERAGE PCT DIFFERENCE. = 0.9

SUMMARY

MAP NO: N1-16-03 DATE: 10/23/01 POWER: 100%

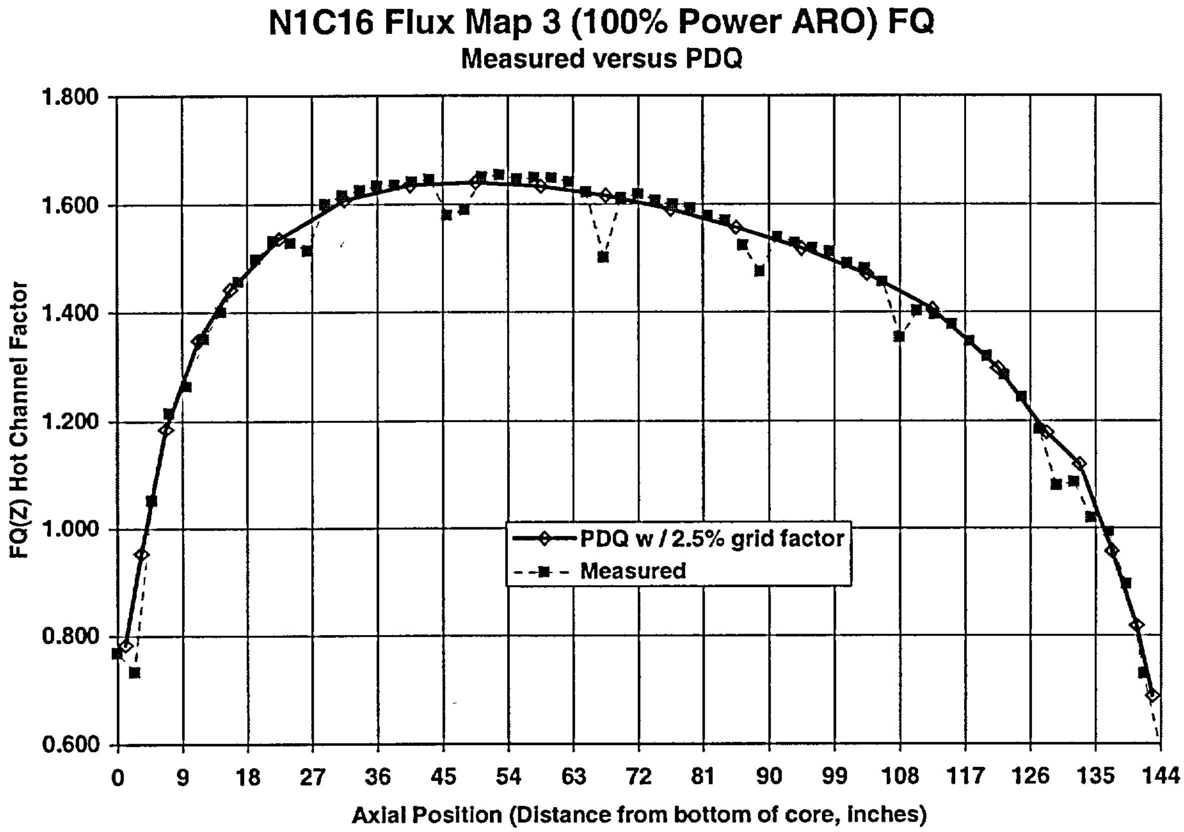
CONTROL ROD POSITIONS: F-Q(Z) = 1.786 CORE TILT:

D BANK AT 226 STEPS F-DH(N) = 1.405

F(Z) = 1.139

NW 0.9982	NE 1.0035
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 FΔH, 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
$F\Delta H$ (North Anna)	1479	0.1%	1.9%	No	N/A	60
$F\Delta H$ (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

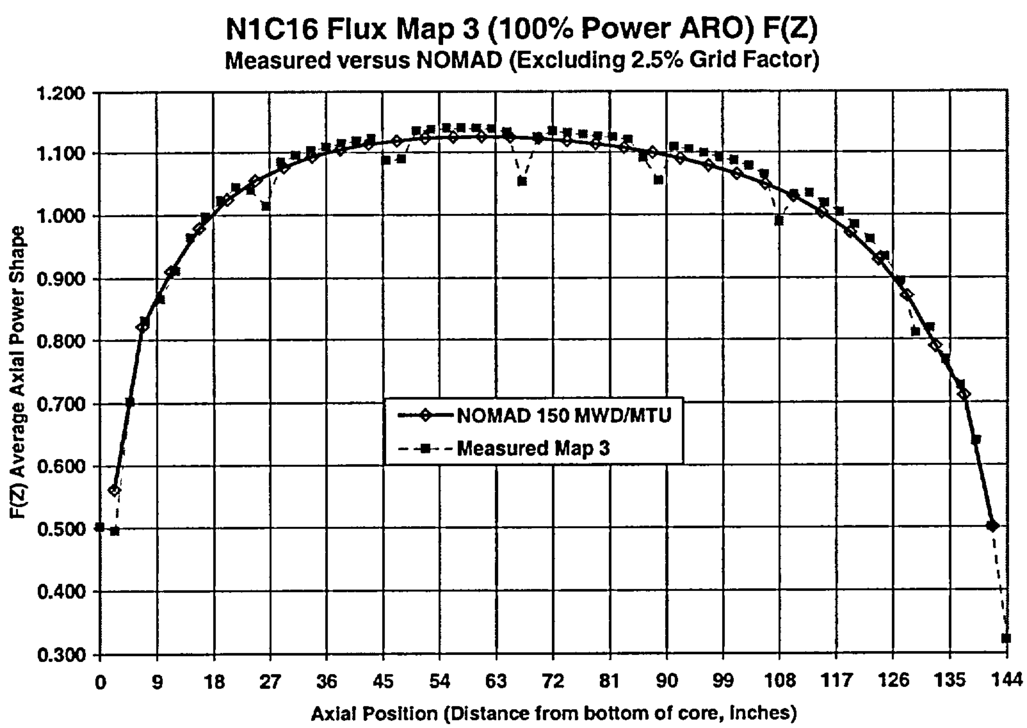
By letter dated December 2, 2002, VEPCO stated that the accuracy of the NOMAD model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the NOMAD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

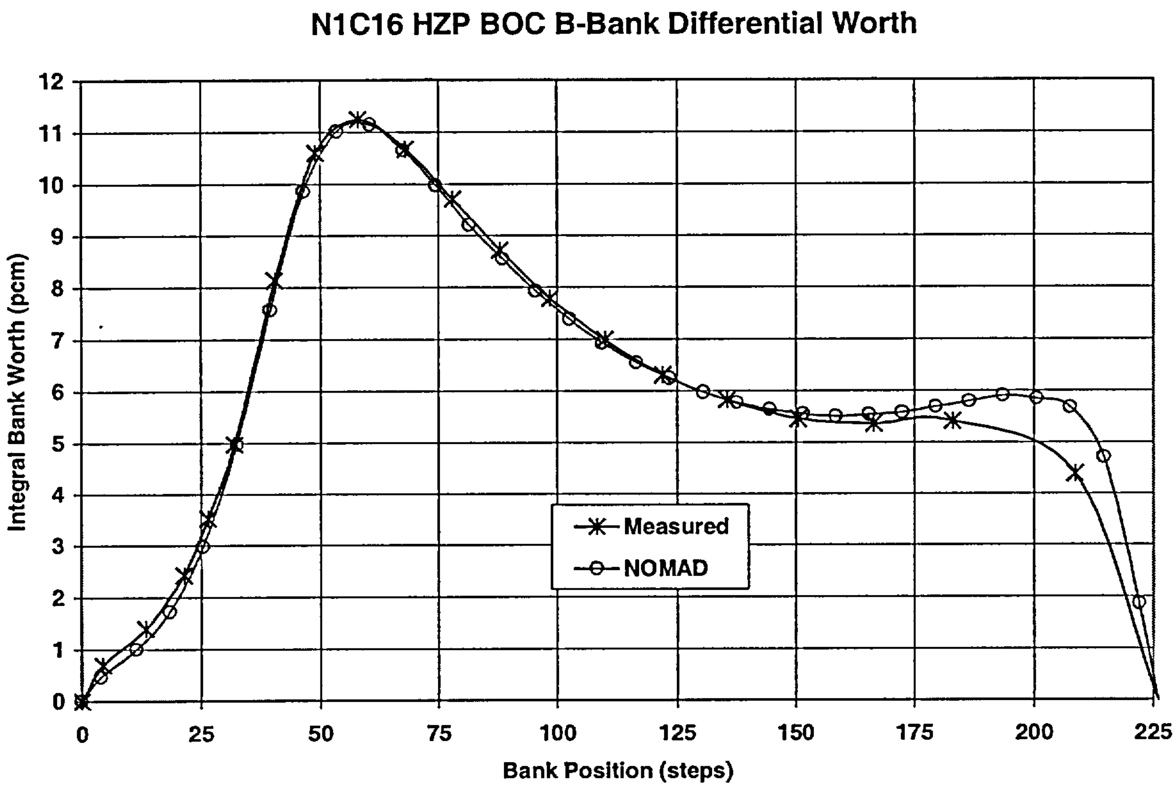
DOMINION RESPONSE TO QUESTION 1

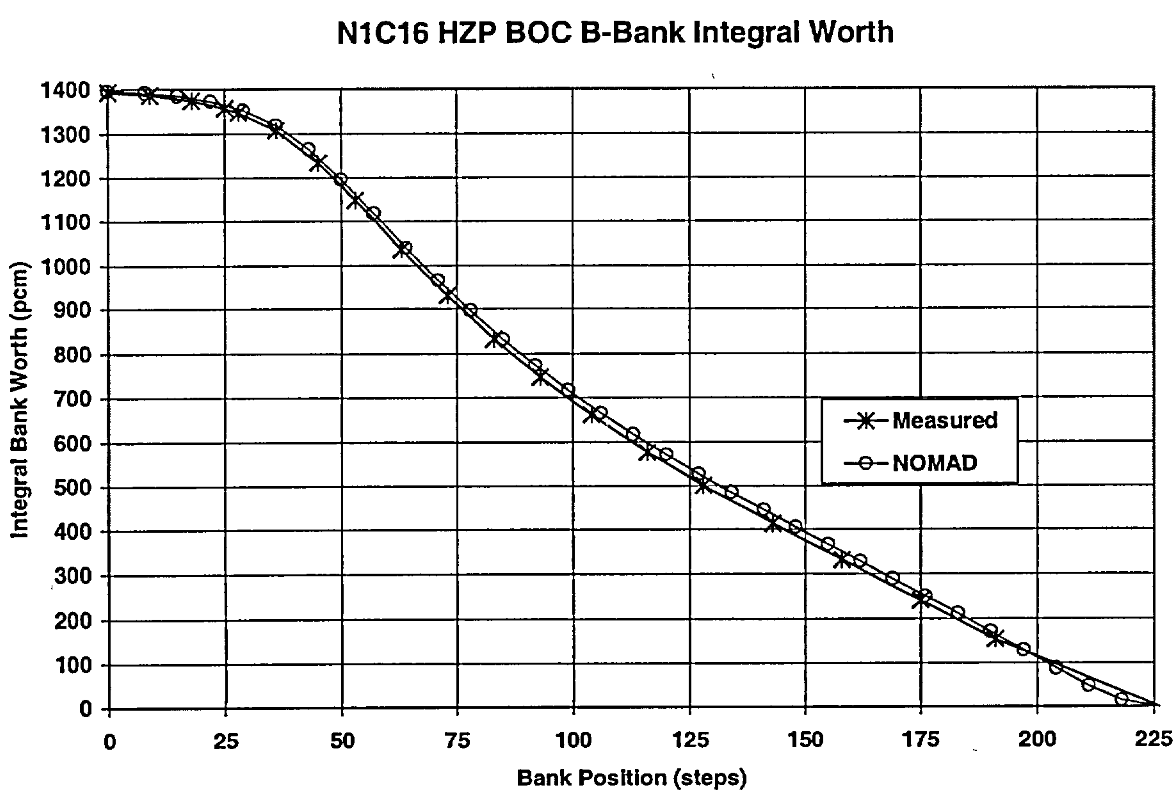
Verification of NOMAD accuracy comes primarily by extension through comparison to PDQ Two Zone model (Topical Report VEP-NAF-1) predictions during the NOMAD model setup process (see also the response to questions 3 and 7). The NOMAD model setup procedure provides specific power distribution and reactivity 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.

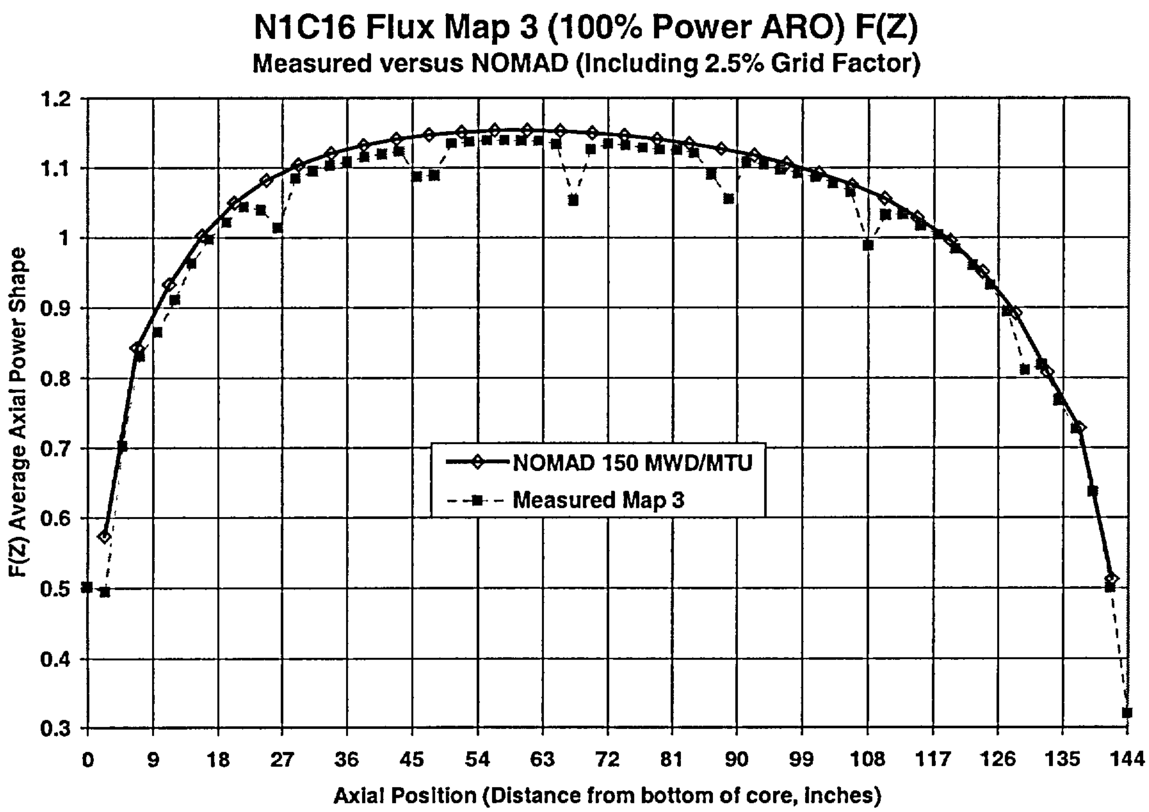
The final element of the control rod model is the ability to normalize bank worth to the PDQ Two Zone value. Although NOMAD was designed to produce acceptable control rod worth results without normalizing to PDQ, normalization is performed routinely for many design calculations to eliminate any difference 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.