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January 4, 2018

Dominion Energy

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No. 17-476
NRA/DEA R0
Docket Nos.: 50-423
50-338/339
50-280/281
License Nos.: NPF-49
NPF-4/7
DPR-32/37

DOMINION NUCLEAR CONNECTICUT, INC.
VIRGINIA ELECTRIC AND POWER COMPANY
MILLSTONE POWER STATION UNIT 3
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
SUBMITTAL OF TOPICAL REPORT VEP-NE-1-A REVISION 0, MINOR REVISION 3, RELAXED POWER DISTRIBUTION CONTROL METHODOLOGY AND ASSOCIATED FQ SURVEILLANCE TECHNICAL SPECIFICATIONS AND TOPICAL REPORT VEP-FRD-42-A, REVISION 2, MINOR REVISION 2, RELOAD NUCLEAR DESIGN METHODOLOGY
FOR INFORMATION ONLY

Enclosed is one copy of topical report VEP-NE-1-A, Revision 0, Minor Revision 3, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," and one copy of VEP-FRD-42-A, Revision 2, Minor Revision 2, "Reload Nuclear Design Methodology."

The minor revision to VEP-NE-1-A includes the following:

- Addition of an alternate element of the methodology utilizing a direct xenon axial search method (in addition to the existing free xenon oscillation method) to create a sufficient number of axial power distributions to adequately bound the potential axial flux difference (delta-I) range allowed by normal operation.
- Addition of the applicability of the VEP-NE-1 methods to Millstone Unit 3 (in addition to North Anna and Surry Units 1 and 2), per prior NRC approval.
- Removal of explicit references to individual methodologies and replacement with a generic reference to an approved NRC neutronics code.

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- Administrative changes such as updating the record of revision, removing reference to specific plants where possible, regenerating figures with updated methodologies consistent with the above changes, and attaching the original RAI responses for VEP-NE-1-A, Rev. 0 consistent with current NRC guidance for publishing approved topical reports.

The change bars in VEP-NE-1-A, Revision 0, Minor Revision 3 reflect differences from the previous approved version of topical report VEP-NE-1-A, Revision 0, Minor Revision 2 (previously referenced as Revision 0.2-A).

The minor revision to VEP-FRD-42-A includes the following:

- Addition of Generic PWR CMS5 Core Physics methodology, as described in Topical Report SSP-14-P01/028-TR-P-A (Reference 2), as an acceptable analytical model for reload design and safety analysis.
- Addition of Millstone Unit 3 to the list of applicable Dominion Energy nuclear units for which VEP-FRD-42-A is applicable, per prior NRC approval.
- Removal of descriptions of the PDQ Two Zone and NOMAD analytical models (now obsolete).
- Conforming changes to the References section of VEP-FRD-42-A and addition of a note to other Dominion Energy topical report references that the current revision of any topical report is cited in the plant specific UFSAR/FSAR. Other minor editorial changes were also made.

The change bars in VEP-FRD-42-A, Revision 2, Minor Revision 2 reflect differences from the previous approved version of topical report VEP-FRD-42-A, Revision 2, Minor Revision 1 (previously referenced as Revision 2.1-A).

VEP-NE-1-A, Revision 0, Minor Revision 3 and VEP-FRD-42-A, Revision 2, Minor Revision 2 are both being submitted to the Nuclear Regulatory Commission for information only.

Should you have any questions in regard to this submittal, please contact Ms. Diane E. Aitken at (804) 273-2694.

Sincerely,

Craig D. Sly
Manager Nuclear Regulatory Affairs

Attachments:

1. VEP-NE-1-A, Revision 0, Minor Revision 3, Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications
2. VEP-FRD-42-A, Revision 2, Minor Revision 2, Reload Nuclear Design Methodology

Commitments made in this letter: None.

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ATTACHMENT 1

TOPICAL REPORT VEP-NE-1-A
REVISION 0, MINOR REVISION 3
RELAXED POWER DISTRIBUTION CONTROL METHODOLOGY
AND ASSOCIATED FQ SURVEILLANCE TECHNICAL SPECIFICATIONS

Millstone Power Station, Unit 3
North Anna Power Station, Units 1 and 2
Surry Power Station, Units 1 and 2

VEP-NE-1-A, Revision 0, Minor Revision 3

RELAXED POWER DISTRIBUTION CONTROL METHODOLOGY AND ASSOCIATED FQ SURVEILLANCE TECHNICAL SPECIFICATIONS

NUCLEAR ENGINEERING & FUEL STAFF

DOMINION ENERGY

RICHMOND, VIRGINIA

October 2017

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

Serial # 86-127

February 20, 1986

Rec'd. FEB 26 1986

Nuclear Operations
Licensing Supervisor

Mr. W. L. Stewart, Vice President
Nuclear Operations
Virginia Electric and Power Company
Richmond, Virginia 23261

Dear Mr. Stewart:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT VEP-NE-1, "VEPCO RELAXED POWER DISTRIBUTION CONTROL METHODOLOGY AND ASSOCIATED FQ SURVEILLANCE TECHNICAL SPECIFICATIONS"

We have completed our review of the subject topical report submitted by the Virginia Electric and Power Company (VEPCO) by letter dated December 10, 1984. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that VEPCO publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, VEPCO and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Standardization and Special
Projects Directorate
Division of PWR Licensing-B

Enclosure:
As stated

SAFETY EVALUATION REPORT

Report Title: Vepco Relaxed Power Distribution Control Methodology and Associated F Q Surveillance Technical Specifications

Report Number: VEP-NE-1

Report Date: October, 1984

INTRODUCTION

The Virginia Electric and Power Company (Vepco) has developed the relaxed power distribution control (RPDC) methodology to replace the constant axial offset control (CAOC) strategy currently employed at its Surry and North Anna reactors. Associated with the RPDC methodology is direct monitoring of the maximum peaking factor (F_Q) relative to plant limits; this replaces the present Fxy Technical Specifications. The analyses performed in support of relaxed power distribution control, and sample generic F_Q surveillance Technical Specifications are described in the subject report. Additional information considered in this review is given in Ref. 1.

SUMMARY OF TOPICAL REPORT

The constant axial offset control (CAOC) strategy currently employed by Vepco was developed by Westinghouse (W) in order to meet power peaking limits imposed by loss of coolant accident (LOCA) analyses. The CAOC procedure requires the maintenance of the axial flux difference (ΔI) within a specified, constant band about a target axial offset defined at equilibrium conditions. While maintenance of ΔI within these limits insures that the F_Q is bounded by a specified limit, CACO is unnecessarily restrictive, particularly below full power where significant margin to peaking limits exits. These restrictive ΔI limits have a negative impact on operational flexibility, especially in the ability to return to full power quickly following a reactor trip near end-of-cycle (EOC). The development of the relaxed power distribution control approach by Vepco was motivated primarily by this limitation.

ENCLOSURE

Under RPDC the ΔI vs. power operating domain is typically broader than that permitted under CAOC (even with band widening), with the width of the band increasing with decreasing power levels. (Similar variable width operating bands are employed by all three PWR vendors in their axial power distribution control procedures). The variable ΔI vs. power operating band takes advantage of the increased F_Q limits permitted at reduced power by maintaining a roughly constant margin to design limits at all power levels (vs. an increasing margin with decreasing power in CAOC)

The major elements of the RPDC methodology are:

1. Axial power distributions are generated with the Vepco one-dimensional NOMAD (Ref. 2) code which bound the potential ΔI operating band. The NOMAD analysis produces a spectrum of xenon distributions at selected burnups via a free-oscillation technique similar to that developed by Combustion Engineering (CE) (Ref. 3). The resulting xenon distributions are combined with rod insertions and power levels permitted by the power dependent rod insertion limit curve at the selected burnups to produce a range of power distributions (and associated ΔI's) at power levels between 50% and full power.

2. The axial power distributions from (1) are used in a 1D/2D/3D synthesis of F_Q (z) based on values of Fxy(z) generated by the Vepco FLAME (Ref. 4) and PDQ07 (Ref. 5) models. The synthesis includes an axial height dependent radial xenon redistribution factor calculated by FLAME, and uncertainty factors which account for the calculational uncertainty, and manufacturing variabilities.

3. Comparison of the resultant F_Q's to limits prescribed by LOCA analyses defines a preliminary ΔI vs. power operating domain.

4. The entire set of axial power distributions is also analyzed with the COBRA (Ref. 6) code relative to the 1.55 design axial power distribution for the loss of flow accident (LOFA). This analysis defines a second ΔI-power operating space that insures that the margin to the DNB design basis for LOFA is maintained.

5. The most restrictive ΔI-power domain (based on LOCA and/or LOFA) defines the permissible space for normal operation (Condition I). (For Vepco plants, the LOCA based band is usually more restrictive). Maintenance of ΔI within this operating space, coupled with adherence to control rod insertion limits, ensures that the margin to fuel centerline melt, DNB, and LOCA peak clad temperature design criteria are maintained during normal operation.

6. Three abnormal operation (Condition II) events are also considered in the analyses supporting RPDC: uncontrolled rod withdrawal, excessive heat removal, and erroneous boration/dilution. The purpose of these analyses is to confirm that the over-power delta-T (OPDT) and over-temperature delta-T (OTDT) trip setpoints have been conservatively calculated, and insures that required margins are maintained. The OPDT and OTDT trips provide transient and steady-state protection against fuel center-line melt and DNB, respectively. The initial conditions for the analyses of these events consist of the axial power distributions allowed by the Δ-I power operating domain determined in (5).

7. The maximum linear power density for each resulting Condition II distribution is determined by using the F_Q (z) synthesis techniques (with an allowance for densification) and compared to the design basis for fuel centerline melt. The OPDT f(ΔI) function is modified, if necessary, to insure that margin to the fuel center-line melt limit is maintained. The axial power distributions from the Condition II analyses are also evaluated to confirm that the OTDT trip function and its associated f(ΔI) term remain valid.

In conjunction with the implementation of the RPDC methodology, Vepco proposed to replace the current Fxy surveillance with direct monitoring of F_Q (z). In F_Q surveillance the measured F_Q at equilibrium conditions is augmented by a factor, N(z), which accounts for the maximum potential increase in F_Q (z) during normal opertion. The resultant augmented F_Q (z) is compared to the plant LOCA F_Q (z) limits to determine acceptability, or to initiate remedial actions. Sample Technical Specifications to be used with F_Q surveillance are given.

While the greatest benefit of relaxed power distribution control to Vepco is the ability to return to power quickly following a trip near EOC, institution of this methodology with its wider operating band is expected to yield additional operational benefits including reduced control rod motion and coolant system boration/dilution requirements.

SUMMARY OF TECHNICAL EVALUATION

All the analyses performed in support of RPDC employed codes which have been previously reviewed and approved by the staff (FLAME, PDQ07, NOMAD, COBRA). The approach used for generating bounding axial power distributions is based on the free xenon oscillation technique employed for a number of years by Combustion Engineering in their axial power distribution control methodology. (CE served as a consultant to Vepco in the implementation and application of this technique). Vepco has determined that this approach results in axial power distributions that sufficiently span the ΔI-power domain to ensure there is confidence that the most adverse conditions are available for subsequent analyses. In addition, relevant analyses performed by CE show that the sensitivity of the results obtained employing the free xenon oscillation methodology to variations in the impacting parameters are small, and are more than compensated for by the "bounding" nature of the approach, and the extreme distributions considered. This approach has been found acceptable for CE reactors for many years, and is acceptable for RPDC.

The calculation of F_Q via a 1D/2D/3D synthesis is similar to accepted approaches. Uncertainties associated with the calculation of F_Q are based on comparisons to measurements. The measurements included situations where azimuthal tilts spanning the range permitted by the technical specification limits were present. The combination of the FNU and FGR components of the uncertainty given in the report is greater than the 95/95 upper tolerance limit determined on the basis of comparisons to measurements. The magnitude of the uncertainty assigned to the calculated value of F_Q in the RPDC analyses is therefore acceptable.

Calculations of the radial xenon redistribution factor, Xe(z), component of the F_Q synthesis employed the FLAME code and considered a number of cycles, times in life and initiating conditions. The final Xe(z) was chosen such that it bounded all observed increases in Fxy(z). Even though this factor is now less than the previously used axially uniform value of 1.03, the analyses performed to justify the lower values are adequate.

The LOFA analyses performed with COBRA, and the Condition II events considered are similar to those included in the Westinghouse relaxed axial offset control (RAOC) methodology.

The over-power and over-temperature ΔT trip functions will be evaluated on a reload basis to assure protection against fuel center-line melt and DNB design basis limits. Other accident analyses will be reevaluated on a reload basis to insure that the assumption used in the RPDC analyses remain bounding.

Monitoring of adherence to operation within the permissible ΔI-power domain is accomplished by reliance on the ex-core detectors. The calculated ΔI domain will be reduced by 3% to accommodate the maximum excore detector calibration uncertainty permitted by the Technical Specifications. In addition, Vepco plans to further reduce the ΔI limits for the first-time analysis. The bounding nature of the RPDC approach provides further conservatism.

The Vepco RPDC methodology contains elements similar to those included in the W (Ref. 7) and CE variable-width ΔI band axial power distribution control strategies. Approved methods have been used in the analyses supporting RPDC and justification has been provided for the uncertainties assigned. These analyses and uncertainties are consistent with currently approved methods and practices. In addition, the impact of cycle specific variations on the ΔI - power domain, the over-power and over-temperature ΔT trip setpoints, and other safety analyses will be evaluated on a reload basis. Based on these considerations the RPDC approach represents an acceptable methodology for use with reload cores similar to those of the Surry and North Anna reactors.

The proposed F_Q surveillance is similar to the approach approved for W in conjunction with RAOC. The N(z) factor by which the measured F_Q (z) distribution is augmented to account for non-equilibrium normal operation is similar to the W(z) and V(z) functions used by W and Exxon, respectively, and approved for use with RAOC and PD II power distribution control strategies. The sample Technical Specifications given in the subject report replace Fxy surveillance with F_Q surveillance. This is acceptable because the F_Q surveillance is more appropriate for RPDC.

The sample Technical Specifications in the report acceptably implement RPDC with the following modifications:

- Specification 3/4.2, page 3/4 2-1

  The asterisk at the end of the APPLICABILITY line and the footnote should be deleted.

- Figure 3.2-1, 3/4 2-4

  This figure should be blank and contain the legend: "This curve is given in the Core Surveillance Report as per Specification 6.9.1.10."

- Specification 3.2.2, page 3/4 2-5

  Parenthetical comments should be added to the final three lines of action a as follows: "subsequent POWER OPERATION may preceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(z)$ exceeds the limit.

- Specification 6.9.1.10 (page unnumbered)

  After "initial criticality", add "unless otherwise approved by the Commission by letter", and change the end of the first paragraph to "approved by the Commission by letter".

A complete set of these revisions will be approved for North Anna Unit 2, Cycle 4 and could be used as a model.

CONCLUSION

We find the subject report suitable for reference as support for use of RPDC in licensing applications.

REFERENCES

1. Letter from W.L. Stewart (Vepco) to H.R. Denton (USNRC), "Virginia Electric and Power Company Relaxed Power Distribution Control (RPDC) Supplemental Information," (Oct. 21, 1985).

2. S.M. Bowman, "The Vepco NOMAD Code and Model," VEP-NFE-1A, Virginia Electric and Power Company (May 1985).

3. "C-E Setpoint Methodology," CENDP-199-NP Rev. 1-NP, Combustion Engineering Inc. (March 1985).

4. W.C. Beck, "The Vepco FLAME Model," VEP-FRD-24A, Virginia Electric and Power Co. (July 1981).

5. M.L. Smith, "The PDQ07 Discrete Model," VEP-FRD-19A, Virginia Electric and Power Co., (July 1981).

6. F.W. Silz, "Vepco Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT Computer Code," Vepco-FRD-33A, Virginia Electric and Power Co. (Oct. 1983).

7. R.W. Miller et al., "Relaxation of Constant Axial Offset Control," NS-EPR-2649 Part A, Westinghouse Electric Corp. (August 1982).

CLASSIFICATION/DISCLAIMER

PREFACE

VEP-NE-1-A, Revision 0, Minor Revision 3: Incorporates the axial xenon search methodology into the Axial Shape Generation section, eliminates reference to specific versions of production software and removes reference to out-of-date software, and removes reference to specific plants (North Anna and Surry) where applicable. Applicability to MPS3 is incorporated by Reference 21. This is a minor revision; therefore, change bars have been included to reflect differences from the previous version (VEP-NE-1-A, Revision 0, Minor Revision 2).

VEP-NE-1-A, Revision 0, Minor Revision 2: Modifies the FQ surveillance information in accordance with the approved Technical Specifications updates provided in NRC Correspondence Serial No. 16-416 [18] to address issues noted in Westinghouse Nuclear Safety Advisory Letter (NSAL) 09-5, Rev. 1 and Westinghouse NSAL-15-1, Rev. 0. Updates Figure 5.0-1 to align with the expanded axial region included in Dominion's FQ Surveillance program. This is a minor revision; therefore, change bars have been included to reflect differences from the previous version (VEP-NE-1-A, Revision 0, Minor Revision 1).

VEP-NE-1-A, Revision 0, Minor Revision 1: Presented a modified version of the Relaxed Power Distribution Control Methodology provided in VEP-NE-1-A, Rev. 0 published in March 1986. Updated the references to the current 3-D PDQ Two Zone and enhanced NOMAD models as well as outlined the use of the NRC approved Studsvik Core Management System in the RPDC methodology. Referred to the COLR section of the plant Technical Specifications for the applicable thermal-hydraulic codes(s) and correlation(s) for DNB analyses.

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SECTION 1 - INTRODUCTION

In response to Loss-of-Coolant Accident (LOCA) Emergency Core Cooling System (ECCS) criteria that imposed new requirements on local power peaking, Westinghouse developed the Constant Axial Offset Control (CAOC) power distribution control procedure [1]. The CAOC strategy restricts axial power skewing in the reactor core during normal operation to within a band of ±5% delta-I around a target value, determined at all-rods-out equilibrium conditions. Delta-I is defined as

$$Delta - I(\%) = 100 \times (Pt - Pb)$$

where Pt and Pb are the fractions of rated full-core power in the top and bottom halves of the core, respectively. This ±5% limit on axial power skewing reduces the magnitude of axial xenon oscillations which, in turn, decreases the magnitude of any power peaking during abnormal operation. A typical CAOC delta-I band is shown in Figure 1.0-1. The CAOC target value varies with burnup as the all-rods-out equilibrium delta-I changes.

Much of the low power operational flexibility of CAOC was originally centered around the use of the part length rods as a means for axial power distribution control [1]. Full length rods and boron were to be used mainly for reactivity control associated with changes in power. Since the requirement for removal of part length rods was imposed, full length rods have had to be used to help control the axial power distributions. As a result, it became more difficult to maintain the axial power distribution within the ±5% delta-I band at low powers. This is especially true near end-of-cycle when the soluble boron concentration has been reduced to a very low level to compensate for the effects of fuel depletion and fission product buildup. Should a trip occur during this portion of the cycle, a plant may not be able to return to full power easily because of difficulty in meeting the delta-I limits. There is insufficient reactivity available from boron dilution to allow the full length rod movement required to offset the buildup of xenon and, at the same time, maintain delta-I within its band. As a result, delta-I limits could be exceeded at low power levels, requiring the plant to remain below 50% power in order to meet the "one hour in twenty-four"[1] requirement in the plant Technical Specifications.

Some Westinghouse CAOC plants with available full power margin to their LOCA Overall Peaking Factor (FQ) license limits have transformed this margin into operating flexibility through delta-I "band widening." In the past [2], Surry had a delta-I band width of +6, -9% about the target value. This method of gaining operational flexibility does provide some additional full power delta-I operating space, but offers only minimal relief for post-trip return to power at end-of-cycle conditions.

[1] The CAOC Technical Specifications impose no operational limit on delta-I while a plant operates below 50% power. However, in order to ascend above 50% power, the plant must not have exceeded the delta-I bands for more than one penalty hour of the previous twenty-four.

This operational restriction on delta-I imposed by CAOC can be eased by the implementation of a variable delta-I band control strategy that takes credit for the full power delta-I margin available from standard band widening while also providing for an increasing delta-I band with decreasing power. The widened delta-I band is formed by maintaining an approximately constant analysis margin to the design bases limits at all power levels. This is in contrast to CAOC operation which has large amounts of margin available at reduced power. For plants which have LOCA-limited total peaking factors, this variable delta-I band would be selected such that the margin to the LOCA FQ*P*K(z) limit would remain approximately constant for all power levels. An example of a variable delta-I band is given in Figure 1.0-2.

The principal benefits of a variable band delta-I control strategy over CAOC operation are as follows:

1) The ability to return to power after a trip, particularly at end-of-cycle, is enhanced;
2) Control rod motion necessary to compensate for the CAOC ±5% delta-I band restrictions is reduced to only that motion needed to maintain operation within a much wider band;
3) The reactor coolant system boration/dilution requirements are decreased, due, in part, to the reduced control rod motion;
4) The plant has enhanced operational flexibility.

The concept of widened delta-I limits at reduced power levels is not a new one. Combustion Engineering [3] and Babcock and Wilcox [4] have supported increased axial skewing at reduced power levels for their reload cores for several years. Westinghouse [5] has also developed and licensed a variable delta-I control strategy called RAOC (Relaxed Axial Offset Control) for application to reload cores.

Dominion has combined some of the concepts from the Combustion Engineering methodology [3] with the current Dominion analysis techniques [1,6] to form an alternate methodology for variable band delta-I control. This methodology is called Relaxed Power Distribution Control (RPDC). The Sections that follow will discuss the Dominion procedure for generating the variable width delta-I band. They will also discuss the methods used to ensure that the margin to the design bases criteria, such as Departure from Nucleate Boiling (DNB), fuel centerline melt and Loss of Coolant Accident (LOCA) peak clad temperature is maintained.

This report also discusses the formulation of FQ Surveillance Technical Specifications. The CAOC radial peaking factor Fxy(z) surveillance is replaced by FQ(z) monitoring, using the measured value of FQ(z) augmented by a non-equilibrium operation multiplier, in order to verify compliance with the LOCA peaking factors. As will be seen in Section 5, FQ surveillance complements RPDC to form a consistent but more flexible plant monitoring scheme than that provided by the CAOC methods.

FIGURE 1.0-1 – TYPICAL CAOC LIMITS

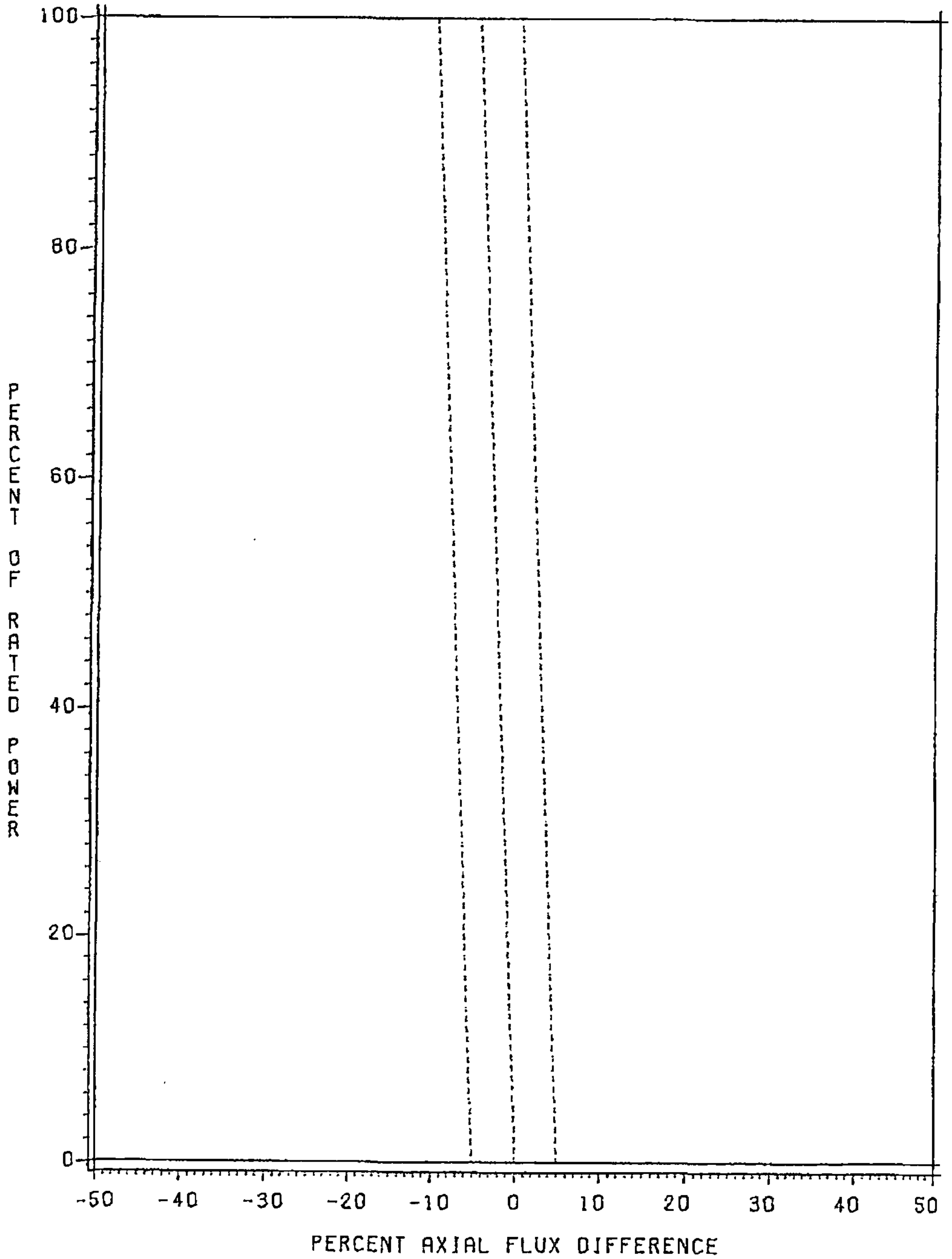
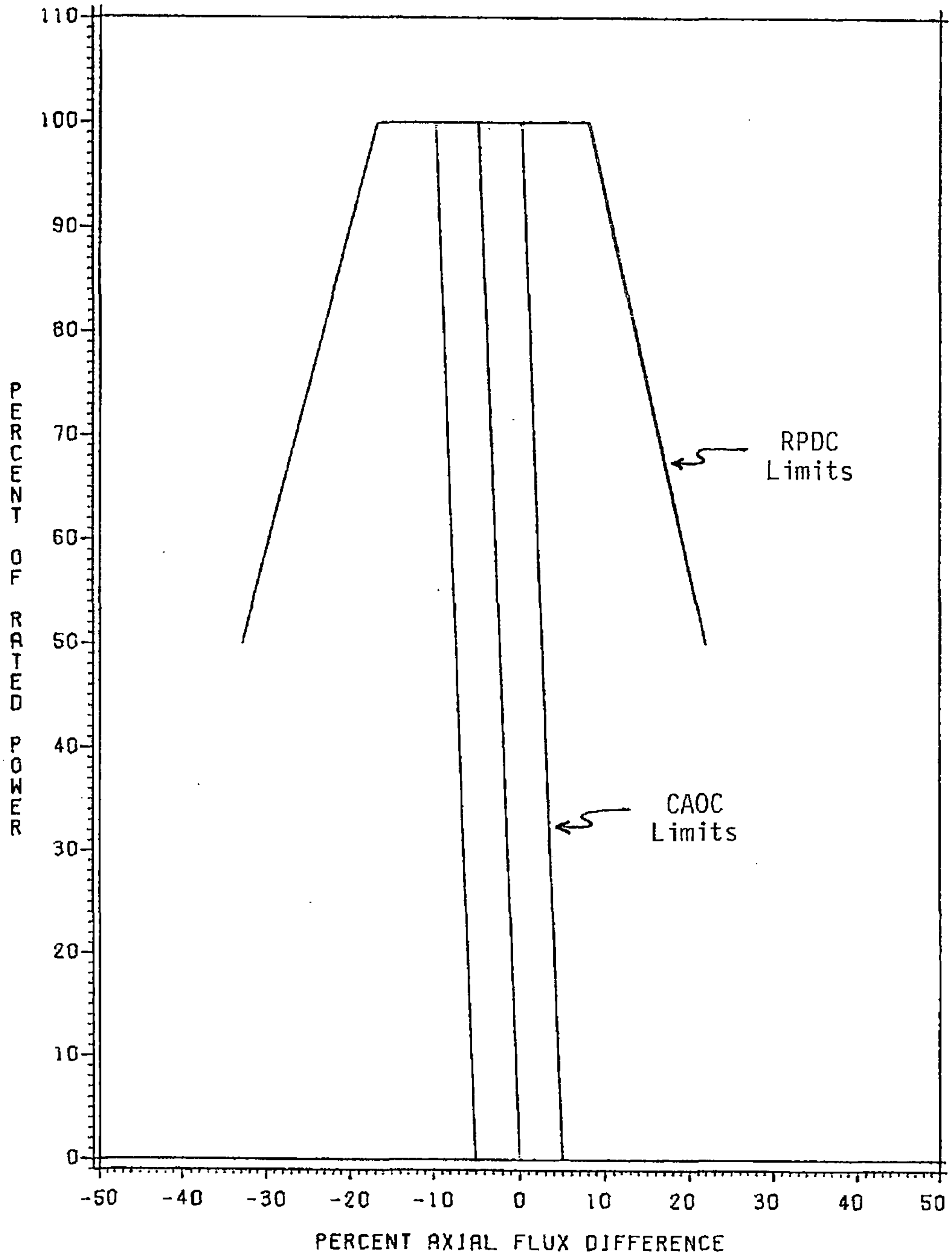


FIGURE 1.0-2 – TYPICAL VARIABLE AXIAL FLUX DIFFERENCE LIMITS



SECTION 2 – CONDITION I ANALYSIS

2.0 Analysis of Axial Shapes Which Result from Normal Operation

The objective of a RPDC analysis is to determine acceptable delta-I band limits that will guarantee that margin to all the applicable design bases criteria has been maintained and, at the same time, will provide enhanced delta-I operating margin over CAOC. Because the RPDC delta-I band is an analysis output quantity rather than a fixed input limit, as in CAOC, axial shapes which adequately bound the potential delta-I range must be generated. These axial shapes must include the effect of all potential combinations of the key parameters such as burnup, control rod position, xenon distribution, and power level. Dominion has developed the methodology of Section 2.1 to generate the large number of axial shapes included in RPDC.

After the axial power shapes have been created, two separate allowable delta-I limits for normal operation are established: one based on LOCA FQ considerations and the other one based on a Loss of Flow (the limiting DNB transient) thermal/hydraulic evaluation. The methods used are described in Sections 2.2 and 2.3, respectively. These two separate delta-I bands are combined to form a composite delta-I limit as discussed in Section 2.4.

All neutronics calculations are performed with NRC approved codes as listed in the COLR section of the plant Technical Specification, for example CMS [16].

2.1 Axial Shape Generation

The axial power distributions encountered during normal operation (including load-follow) are primarily a function of four parameters: power level, control rod bank position, burnup distribution, and the xenon distribution. For RPDC, reasonable incremental variations that span the entire expected range of values must be considered for each of these parameters. The following methods for each parameter are used to create the axial power distributions needed for the development of the RPDC normal operation delta-I limits. The combinations of burnups, power levels, rod configurations and xenon distributions typically evaluated on a reload basis are summarized in Table 2.1-1.

2.1.1 Power Level During Normal Operation

For the normal operation analysis, power levels spanning the 50% to 100% range are investigated to establish the RPDC delta-I limits. This range is consistent with the current CAOC Technical Specifications which do not impose axial flux difference limits or require CAOC operation below 50% of

full power.[2] The power levels used for RPDC analysis are selected at increments within the 50% to 100% range which are small enough to ensure an adequate number of power distributions are being analyzed; i.e. that all safety-related effects due to the power level are accounted for.

2.1.2 Control Bank Position During Normal Operation

During normal operation, the control rod bank insertion is limited by the cycle-specific Core Operating Limits Report (COLR) rod insertion limits. Figure 2.1-3 gives a set of typical rod insertion limits. The insertion limits are a function of reactor power, and the rods may be anywhere between the fully withdrawn position and the variable insertion limit. In order to adequately analyze the various rod positions allowed, control rod insertions versus power level are selected which cover the range of rod insertions allowed for each particular power. Due to the difference in rod insertion limits between plants, the number of rod insertion intervals necessary for each plant may be slightly different.

2.1.3 Cycle Burnup

The RPDC analysis is performed at several times in cycle life in order to provide limiting delta-I bands for the entire cycle. Typically, three or four cycle burnups, near beginning-of-cycle (BOC), near burnable poison burnout (Peak, if applicable), middle-of-cycle (MOC) and end-of-cycle (EOC), are chosen for the RPDC analysis. The MOC case is chosen to reflect the maximum middle-of-cycle radial peaking factors.

2.1.4 Axial Xenon Distributions During Normal Operation

The axial xenon distribution is a function of the core's operating history and, as a result, is constantly changing. In order to analyze a sufficient number of xenon distributions, two methods have been developed to ensure that all possible cases have been accounted for. The first method is a xenon "free oscillation" method similar to the one described in Reference 3. By creating a divergent xenon-power oscillation, axial xenon distributions can be obtained that will be more severe than any experienced during normal operation, including load follow maneuvers. The second method utilizes an axial search method that generates axial xenon distributions at specific conditions and delta-I intervals that bound the expected operating limits. Both of these methods are acceptable for generating a representative set of bounding axial xenon distributions and are described in greater detail in the following sections.

[2] The CAOC Technical Specifications impose no operational limit on delta-I while a plant operates below 50% power. However, in order to ascend above 50% power, the plant must not have exceeded the delta-I bands for more than one penalty hour of the previous twenty four.

2.1.4.1 Xenon "Free Oscillation"

To initiate a xenon-power oscillation in the "free oscillation" method, an equilibrium 3-D model of the reload cycle is perturbed. This perturbation will generally be in the form of a change in power, rod position, or both. However, since the core model may be inherently stable due to the presence of feedback mechanisms, these mechanisms must either be modified or bypassed to obtain a divergent oscillation. One way to accomplish this is to reduce the stability of the model by reducing the amount of Doppler (i.e., fuel temperature) feedback in the system. The divergent oscillation provides a spectrum of xenon distributions that will produce power distributions with delta-I values covering the expected delta-I range. The magnitude of the "free oscillations" should be such that the xenon distributions (when combined with normal operating conditions) produce axial power shapes with delta-I values that bound the expected operating limits.

The stability of the calculational model may vary with burnup or core loading. Therefore, the amount of perturbation and feedback modification necessary to achieve a divergent xenon oscillation may vary with cycle burnup or core loading. Typical examples are given in Figures 2.1-1 and 2.1-2 for beginning- and end-of-cycle, respectively. These particular oscillations were generated with CMS [16] using a reduced Doppler feedback and initiated by fully withdrawing D-Bank from an inserted position at equilibrium conditions.

The final power distributions generated using the free oscillation method used in the RPDC normal operation analysis result from combining axial xenon shapes, power levels, rod insertions and cycle burnups. At each selected time in cycle life, the xenon shapes are combined with each power level and rod configuration. A criticality search is then performed for each case using an approved neutronics code with normal feedback. Calculated axial power distributions are identified for use in the LOCA FQ and thermal/hydraulic evaluations discussed in Sections 2.2 and 2.3. The conditions result in a delta-I range of approximately -60% to +50%, bounding the expected final delta-I envelope at all power levels.

2.1.4.2 Axial Search

The axial search method achieves the same types of power distributions generated by the xenon free oscillation method but utilizes a different technique to provide shapes that bound the desired operating space. CMS has the ability to search to an explicit Axial Offset (AO) or delta-I by manipulating the xenon distributions applied to a particular statepoint. This allows criticality searches to be performed at well defined axial intervals that bound the expected operating limits. The axial search method can be performed directly at any core configuration (xenon shapes, power levels, rod insertions, and cycle burnups) and therefore does not require a recombination of statepoint variables as is required of the xenon free oscillation method. All axial power distributions produced are used in the LOCA FQ and thermal/hydraulic evaluations discussed in Sections 2.2 and 2.3.

FIGURE 2.1-1 – TYPICAL RPDC BOC XENON OSCILLATION

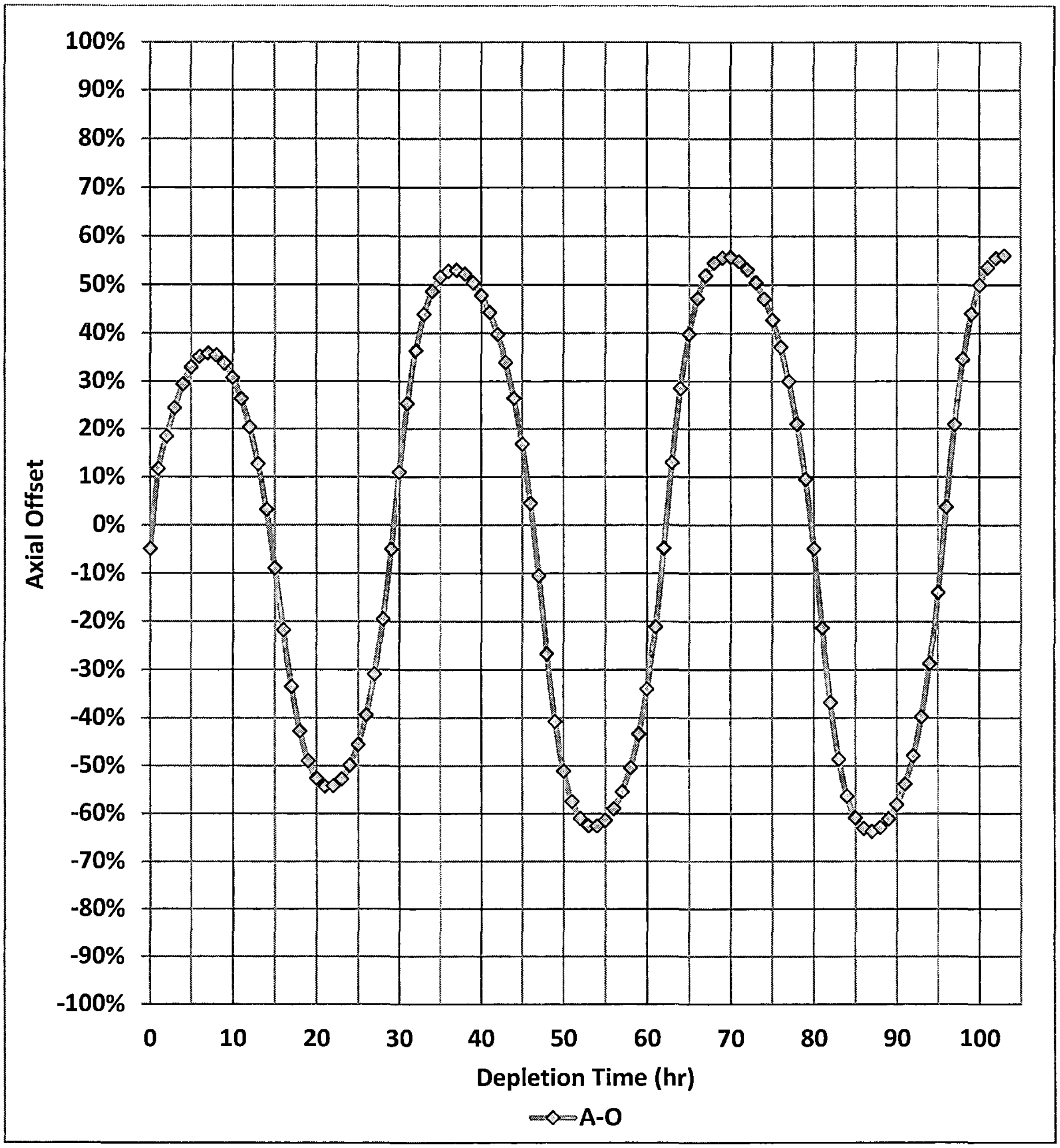


FIGURE 2.1-2 – TYPICAL RPDC EOC XENON OSCILLATION

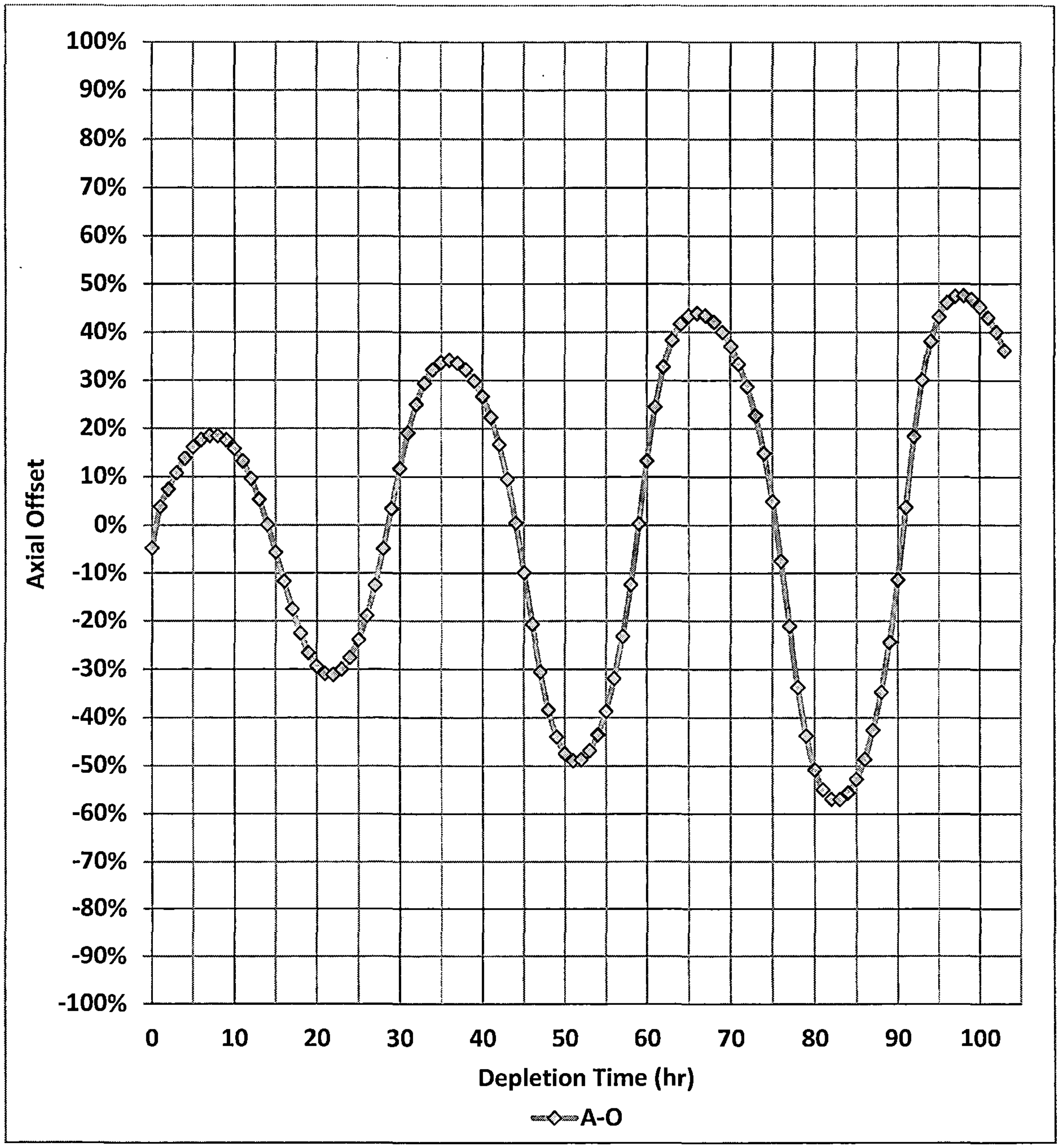


FIGURE 2.1-3 – TYPICAL ROD INSERTION LIMITS

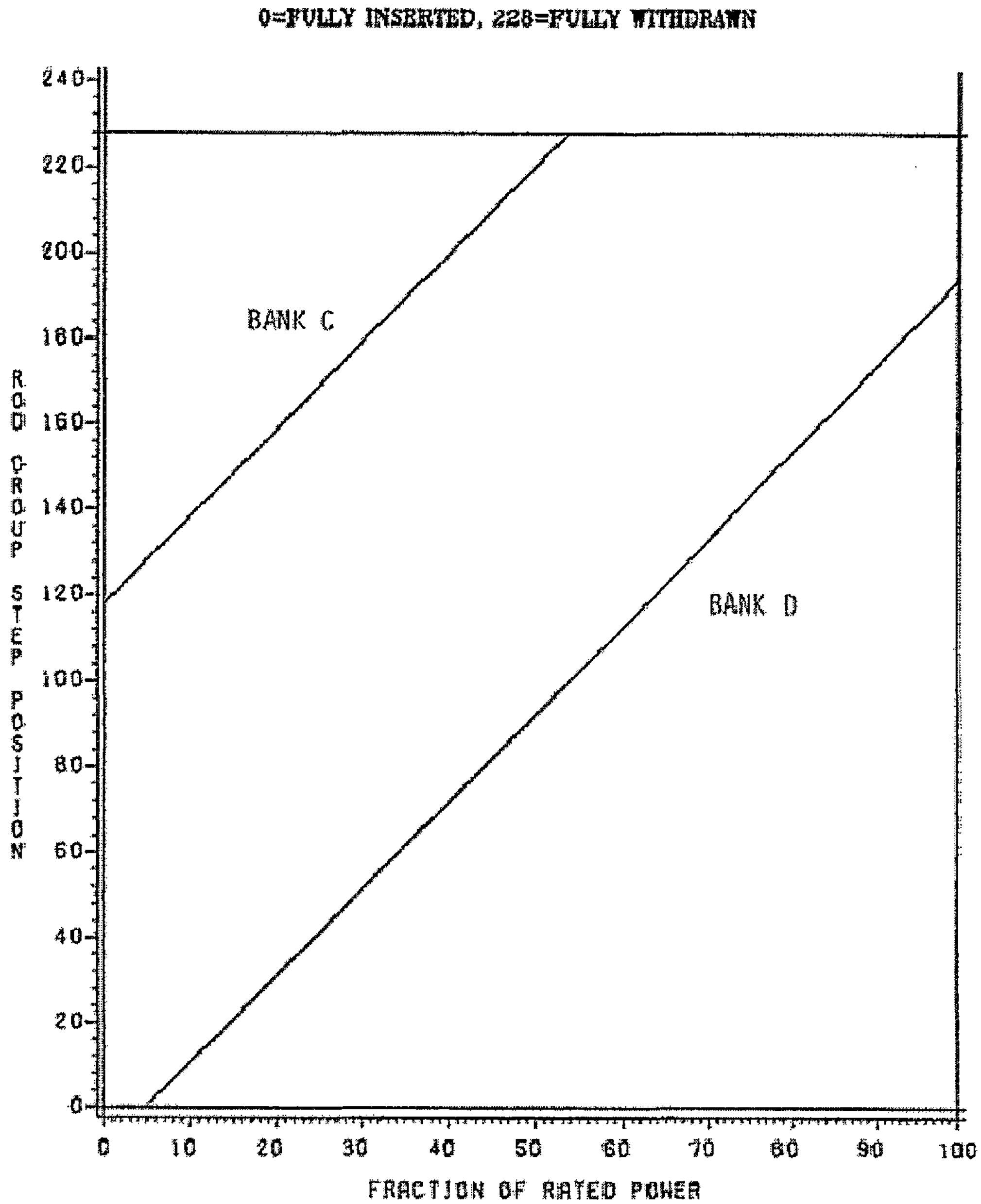


TABLE 2.1-1

TYPICAL CONDITIONS ANALYZED FOR
NORMAL OPERATION UNDER RPDC

Cycle Burnups	BOC, Peak (if applicable), MOC, EOC	
Burnup Window	Low, High	
Power Levels (%)	100, 90, 80, 70, 60, 50	
Rod Insertion Range Versus Power	Plant Specific (See Figure 2.1-3)	
Xenon Shapes	<u>Xenon Free Oscillation</u>	<u>Axial Search</u>
	50 for each burnup	Variable based on axial intervals and delta-I limits

2.2 LOCA Delta-I Limit Formation

The FQ*Power for each shape is compared to the LOCA FQ*Power*K(z) limit at each power level to determine which axial shapes approach the LOCA limit, thereby establishing a preliminary allowable delta-I versus power band. This comparison replaces the traditional CAOC FAC analysis [1] and ensures that the margin to the LOCA FQ*Power*K(z) envelope is maintained during the cycle as long as reactor operation remains within the delta-I limits. A typical LOCA delta-I limit is shown in Figure 2.2-1.

Modification of the LOCA delta-I limits can be used as a means of reducing or increasing the allowable FQ at a constant power. Cycle specific analysis is required to determine the relationship between FQ and the LOCA delta-I limits and core power to determine the allowable operating space.

2.2.1 FQ Using 3-D Model

The axial shapes created in Section 2.1 are used directly:

$$F_Q(z) = Fq(z) \times FNU \times FQE$$

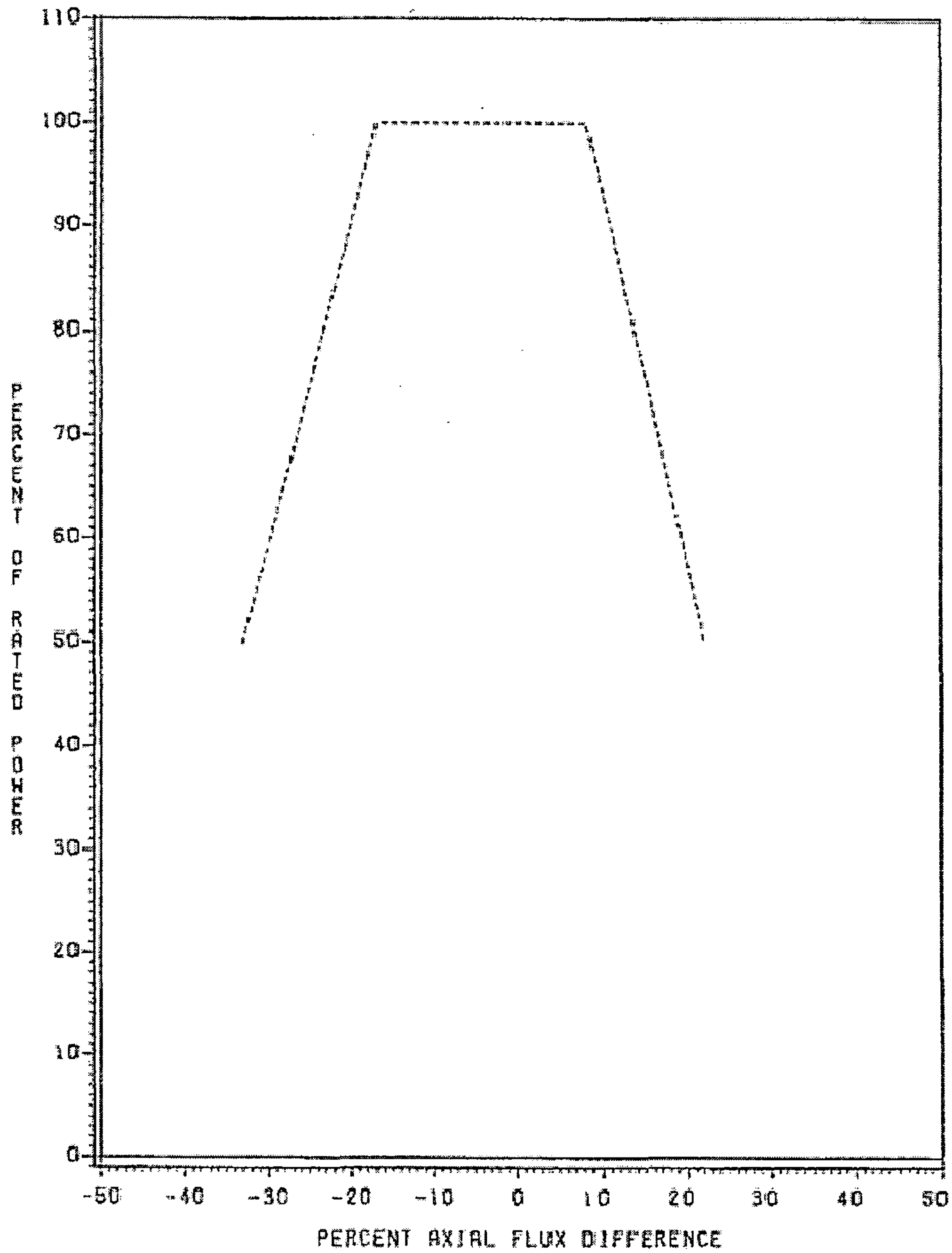
where the following are non-dimensional parameters:

$Fq(z)$ = Fq distribution generated using an approved 3-D neutronics code, dependent upon burnup, core height and rod position and power level. (Includes xenon redistribution and grid effects).

FNU = Nuclear uncertainty factor (consistent with the approved 3-D neutronics code)

FQE = Engineering heat-flux hot-channel factor [9, 10]

FIGURE 2.2-1 – TYPICAL LOCA DELTA-I LIMITS



2.3 Loss of Flow Thermal/Hydraulic Evaluation

The Loss of Flow Accident (LOFA) represents the most limiting DNB transient not terminated by the Overtemperature Delta-T trip. In order to ensure the applicability of the current LOFA analysis, the entire set of axial power distributions formed by the RPDC normal operation analysis are evaluated against the 1.55 cosine design axial power distribution for the Loss of Flow Accident analysis with the applicable thermal-hydraulic code(s) and correlation(s) that are listed in the COLR section of the plant Technical Specification. The thermal/hydraulic evaluation methods used in this LOFA evaluation are similar to those of the CAOC techniques. As a result of this LOFA comparison, a second set of delta-I versus power limits is formed. These delta-I limits delineate the allowable operating band which will ensure that the margin to the DNB design base for LOFA is maintained. The impact of RPDC on other DNB transient events is discussed in Section 3.

2.4 Final Normal Operation Delta-I Limit

The results of the LOFA delta-I limit generation are combined with the LOCA delta-I limits (Figure 2.2-1) to produce a set of limits which will ensure that the preconditions for both accidents are met. These generic limits will be verified on a cycle-by-cycle basis using the RPDC methods described in this report.

The LOCA FQ based delta-I limits are generally more restrictive than LOFA-based delta-I limits for Dominion's plants. This will allow the plant cycle specific COLR to take advantage of the FQ versus delta-I relationship discussed in Section 2.2.

SECTION 3 – CONDITION II ANALYSIS

3.0 Analysis of Axial Shapes Which Result from Condition II Events

One of the important features of any axial power distribution control strategy (RPDC, CAOC or any other) is the clear distinction between normal and accident conditions. The delta-I limits established in Section 2 and the cycle specific COLR control rod insertion limits (see Figure 2.1-3) define conditions of normal operation. If the axial power distribution (as measured by delta-I) remains inside the pre-established band during all normal operation, and the control rods remain within the cycle specific COLR limits, then the margin to the design criteria of fuel centerline melt, DNB and LOCA peak clad temperature, will be maintained.

This Section examines Condition II or Abnormal Operation events, which may be the result of system malfunctions or operator errors and create reactor conditions that fall outside the bounds analyzed in Section 2. The RPDC analysis examines the more limiting of these Condition II events and confirms that the Overpower Delta-T (OPDT) and the Overtemperature Delta-T (OTDT) setpoints[3] have been conservatively calculated and ensures that margin to the fuel design limits is maintained. These setpoints are verified on a cycle-by-cycle basis.

3.1 Determination of Accident Pre-Conditions

Initial condition parameters for Condition II analysis are determined from the core conditions allowed by the normal operation delta-I versus power envelope. These conditions are a function of rod control cluster (RCC) position, boron concentration, xenon distribution, burnup and core power level. Any set of these conditions which produce an axial power distribution within the normal operation delta-I envelope established in Section 2 (Figure 2.2-1) can be a potential starting point for a Condition II accident. Each set of valid normal operation conditions is considered in the RPDC Condition II analyses.

3.2 Condition II Accident Simulation

Three categories of credible accidents bound the range of abnormal operation events which must be considered in terms of their effect upon the axial power distribution or local power peaking. These three accidents are rod withdrawal, excessive heat removal and erroneous boration/dilution. The rod withdrawal and boration/dilution events [1] are the most limiting Condition II events with respect to the impact of control rod position on the axial power distribution or local power peaking. In the excessive heat removal event the impact of temperature is investigated.

[3] The OPDT and OTDT setpoints were designed primarily to provide transient and steady state protection against fuel centerline melt and DNB, respectively.

3.2.1 Uncontrolled Rod Withdrawal Event

The rod withdrawal event [6] is an erroneous control rod withdrawal starting from a normal operation condition with the control banks operating in their normal overlap sequence. To perform the analysis of this accident, the xenon distribution and boron concentration are fixed at values allowed by the normal operation analysis. The analysis is limited to those cases producing power levels between 50% of rated power and the high flux trip limit.

For the Xenon Free Oscillation method, the lead control bank is withdrawn in increments from the fully inserted to the fully withdrawn position. After each incremental movement a criticality search is performed with an approved neutronics code and the axial power distribution is identified for use in the Condition II evaluation of Sections 3.3 and 3.4.

For the Axial Search method, the lead control bank is withdrawn from the Condition I rod position to the fully withdrawn position and a criticality search is performed with an approved neutronics code. All axial power distributions are used in the Condition II evaluation of Sections 3.3 and 3.4.

3.2.2 Excessive Heat Removal Event

The Excessive Heat Removal (or cooldown) event, like the rod withdrawal event, is an overpower accident. The accident assumes a decrease in the reactor core inlet temperature as a result of a sudden load increase, steam-dump valve opening, excessive feedwater flow or a turbine valve opening [6]. Since the control rods are assumed to be in manual control for this event, they will remain at their original position, which allows the reactor power to increase.

Reduction of the inlet temperature is limited to 30°F, which has been shown to bound the results of the above accidents for North Anna, Surry, and Millstone [11, 12, 21]. Cases producing a power level greater than the high flux trip limit are excluded from consideration.

For the Xenon Free Oscillation method, the allowable normal operation xenon distributions, control rod positions and boron concentrations are provided as input to an approved neutronics code. The inlet temperature is reduced and a criticality search is performed. The axial power distribution from each case is identified for use in the Condition II evaluation of Sections 3.3 and 3.4.

For the Axial Search method, the allowable normal operation xenon distributions, control rod positions and boron concentrations are provided as fixed inputs to an approved neutronics code. The inlet temperature is reduced and a criticality search is performed. All axial power distributions are used in the Condition II evaluation of Sections 3.3 and 3.4.

3.2.3 Boration/Dilution

The Boration/Dilution event causes a movement in the control rods to compensate for the reactivity changes due to a change in soluble boron concentration as a result of inadvertent boration or dilution. In this analysis the control banks are assumed to be in automatic mode and to operate in a normal overlap sequence. The manual mode of operation could result in an overpower transient during a dilution incident. However, the consequences of this event are bounded by those of the rod withdrawal accident [6].

For the Xenon Free Oscillation method, to perform the boration/dilution analysis, an approved neutronics code reads each allowable xenon distribution from the Condition I analysis and a criticality search is performed with the rods inserted to the insertion limit. A rod position search is then performed to determine the amount of control rod insertion necessary to compensate for the reactivity associated with a dilution of fifteen minutes. All axial power distributions from the boration/dilution event are identified for the Condition II evaluation of Sections 3.3 and 3.4.

For the Axial Search method, to perform the boration analysis, the xenon distribution from the Condition I analysis is fixed. The lead control bank is withdrawn from the Condition I rod position to the fully withdrawn position and a critical boron search is performed. All axial power distributions are used in the Condition II evaluation of Sections 3.3 and 3.4.

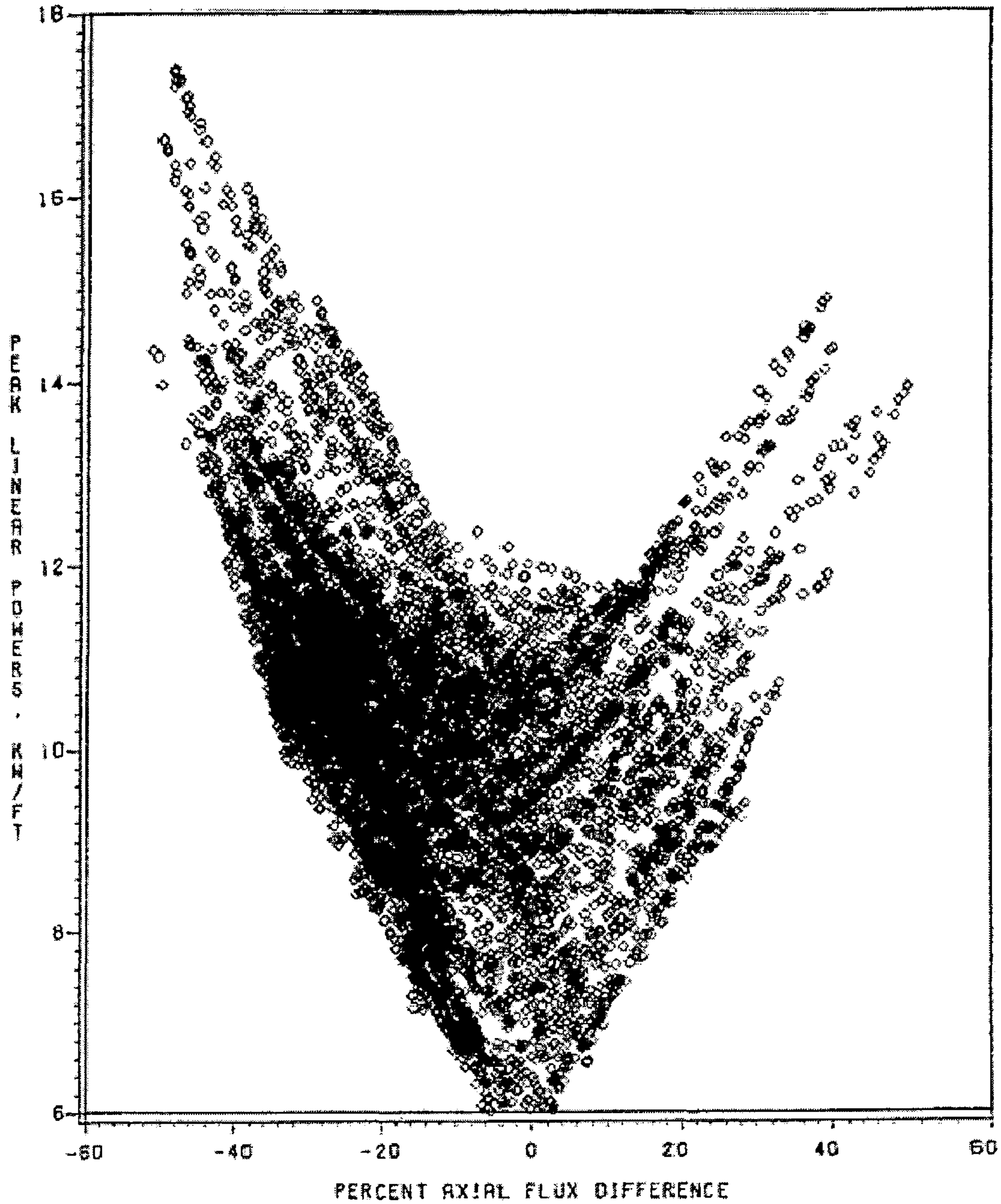
To perform the dilution analysis, the xenon distribution from the Condition I analysis is fixed. The lead control bank is inserted from the Condition I rod position to the insertion limit and a critical boron search is performed. A rod position search is then performed to determine the amount of control rod insertion necessary to compensate for the reactivity associated with a dilution of fifteen minutes. All axial power distributions are used in the Condition II evaluation of Sections 3.3 and 3.4.

3.3 Overpower Limit Evaluation

The maximum linear power density for each distribution produced by the Condition II accident simulations is determined using the 3-D model techniques as described in Section 2.2 (with the addition of the densification spike factor S(z)). The results may be plotted in the "flyspeck" format shown in Figure 3.3-1, which shows typical results for the three limiting Condition II accidents described in Section 3.2.

The peak power density "flyspeck" is compared to the design basis limit for fuel centerline melt. If necessary, the OPDT f(delta-I) function (which provides protection against this design limit) is modified to ensure that margin to the fuel centerline melt limit is maintained. If needed at all, this modification would be required only for very large values of delta-I. An alternative approach would be to maintain the margin to fuel centerline melt by restricting the OTDT f(delta-I) function beyond the DNBR requirement, effectively eliminating the need for the OPDT f(delta-I) function.

FIGURE 3.3-1 – TYPICAL MAXIMUM POWER DENSITY FLYSPECK



3.4 DNB Evaluation

The OTDT trip function and setpoints [13] provide DNB protection for Condition II accidents. Part of this function, the f(delta-I) term, responds to changes in the indicated delta-I created by skewed axial power distributions. The axial power distributions formed by the RPDC Condition II accident simulations are evaluated to confirm that the assumptions [13] used to form the f(delta-I) term and the rest of the OTDT trip function remain valid. If the RPDC power distributions for any subsequent reload should be more limiting than those previously used to establish the OTDT trip setpoints, the OTDT setpoints will be reformulated using standard techniques [13] and the appropriate RPDC power distribution parameters.

SECTION 4 – OTHER SAFETY ANALYSES

No changes are required to the other safety analysis methods described in Reference 6 to incorporate the effect of the widened delta-I band resulting from the RPDC methodology. The CAOC methods used by Dominion employ a conservative method for incorporating the effect of skewed axial power distributions. However, as is the practice with CAOC, the accident analyses will be evaluated on a reload basis for RPDC to ensure that the key input parameters remain bounding. Should an accident analysis be determined to be impacted by a reload design, that accident will be re-evaluated or reanalyzed, as appropriate.

SECTION 5 – FQ SURVEILLANCE

Dominion instituted FQ Surveillance Technical Specifications (TS) as part of the RPDC implementation process. FQ Surveillance Technical Specifications [14, 15, 18, 19] are a convenient method for overall power distribution monitoring during plant operation to ensure compliance with the specified LOCA FQ*K(z) limit. The Heat Flux Hot Channel Factor, $F_Q(z)$, shall be limited by the following relationships:

$$F_Q(z) \leq \frac{CFQ \times K(z)}{P} \quad for\, P > 0.5 \qquad [5-1]$$

$$F_Q(z) \leq \frac{CFQ \times K(z)}{0.5} \quad for\, P \leq 0.5 \qquad [5-2]$$

where the non-dimensional parameters are defined as:

CFQ = the plant LOCA FQ limit

$K(z)$ = the normalized LOCA $F_Q(z)$ limit as a function of core height

P = the fraction of rated thermal power

For TS surveillance and compliance, $F_Q(z)$ is approximated by $F_Q^E(z)$ [Equilibrium F_Q(z)] and $F_Q^T(z)$ [Transient F_Q(z)]. Thus both $F_Q^E(z)$ and $F_Q^T(z)$ must meet the preceding limits on $F_Q(z)$ and have separate required actions if the measurements are not within the limits.

$F_Q^E(z)$ is an excellent approximation for $F_Q(z)$ when the reactor is at the steady-state power at which the incore flux map was taken and is calculated by taking the measured $FQ(z)$ (from incore) and accounting for manufacturing tolerances and measurement uncertainties.

$$F_Q^E(z) = FQ(z) \times FNU \times FQE \quad [5-3]$$

where the non-dimensional parameters are defined as

$FQ(z)$ = the measured plant $FQ(z)$

FNU = Nuclear uncertainty factor (consistent with the approved 3-D neutronics code)

FQE = Engineering heat-flux hot-channel factor [9, 10]

$F_Q^T(z)$ represents the maximum potential $F_Q(z)$ by accounting for increases in localized power due to non-equilibrium normal operation within the allowable Condition I space (core power, delta-I limits, control rod positions, etc.). $F_Q^T(z)$ is calculated by applying a cycle dependent function, N(z), to $F_Q^E(z)$. The N(z) function represents the maximum possible increase in F_Q(z) that could result from normal

operation within a defined period, power range, and delta-I operating space. The impact of control rod insertion and xenon transients, both axial and radial, are included in N(z).

The expression for $F_Q^T(z)$ is:

$$F_Q^T(z) = F_Q^E(z) \times N(z) \qquad [5-4]$$

$$N(z) = \frac{Fq(z), Maximum\ Condition\ I}{Fq(z),\ Equilibrium\ Condition} \qquad [5-5]$$

The Fq(z)'s in equation $[5-5]$ are formed by the 3-D model techniques as described in Section 2.2. N(z) is similar to V(z) given in Reference 15 and W(z) given in Reference 14. A typical N(z) function is given in Figure 5.0-1.

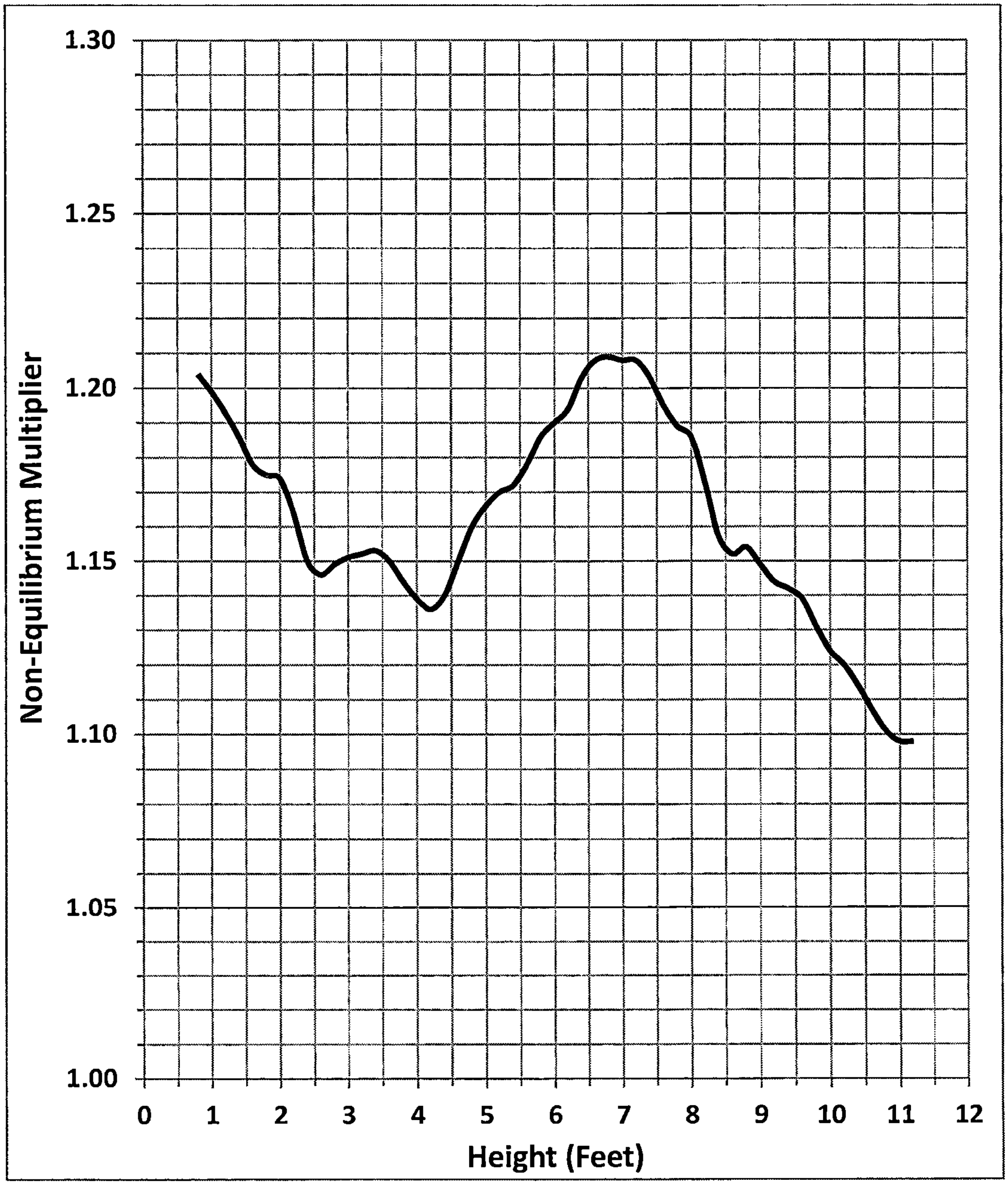
When $F_Q(z)$ exceeds the LOCA FQ*K(z) limits defined in equations $[5-1]$ and $[5-2]$, compensatory measures are needed in order to permit continued operation. There are separate required actions and completion times associated with $F_Q^E(z)$ and $F_Q^T(z)$ not meeting the limits, due to the severity of the condition and the amount of margin needed to restore compliance.

A cycle-specific analysis is performed to determine the relationship between $F_Q^T(z)$ and core power and LOCA delta-I limits. Results from this analysis are presented in the cycle-specific Core Operating Limits Report (COLR) in the form of operating space reductions (thermal power limit and delta-I band limits) that are the TS required actions to address the condition in which the $F_Q^T(z)$ is not within the limit.

In addition, a cycle-specific analysis is performed to determine the maximum change in $F_Q(z)$ over the required TS surveillance intervals. Results from this analysis are presented in the cycle-specific COLR in the form of $F_Q(z)$ surveillance penalties to be applied to the $F_Q(z)$ assessment for any of the following conditions:

1. Increase in measured maximum $\frac{F_Q^E(z)}{K(z)}$ from the previous surveillance,
2. Increase in measured maximum $\frac{F_Q^T(z)}{K(z)}$ from the previous surveillance,
3. Increase in predicted maximum $\frac{F_Q^E(z)}{K(z)}$ over the next surveillance period,
4. Increase in predicted maximum $\frac{F_Q^T(z)}{K(z)}$ over the next surveillance period.

FIGURE 5.0-1 – TYPICAL N(z) FUNCTION



N(z)'s IN THE AXIAL EXCLUSION ZONES AT THE TOP AND BOTTOM OF THE CORE ARE OMMITTED FROM FQ SURVEILLANCE PER TECHNICAL SPECIFICATION BASES

SECTION 6 – CONCLUSION

The RPDC methodology takes advantage of the large amounts of margin to the design bases limits available at reduced power levels in CAOC and forms wider delta-I limits at all powers. The RPDC methodology may be summarized as follows:

1. A full range of normal-operation axial power shapes is obtained by combining the key parameters upon which each shape is dependent: xenon distribution, boron concentration, core power level and control rod position. A xenon "free oscillation" or axial search method is used to create the many and varied axial xenon distributions required for this analysis.
2. These axial power profiles are analyzed to determine which shapes result in an approach to the LOCA and LOFA limits.
3. A final normal operation delta-I limit is established by conservatively bounding both the LOCA and the LOFA limits.
4. Conditions which yield shapes within the final normal operation delta-I limit are used as initial conditions for the bounding Condition II accident simulations.
5. The resultant transient shapes are analyzed and the overpower and overtemperature trip function/setpoints are specified to ensure that margin to fuel design limits is maintained.
6. A N(z) function is formulated based on calculated Condition I Fq's to support the implementation of Fq Surveillance Technical Specifications.

All neutronics calculations are performed with NRC approved codes. All DNBR calculations are performed using the applicable thermal-hydraulic code(s) and correlation(s) that are listed in the COLR section of the plant Technical Specification.

The RPDC methodology presented in this report allows the Dominion nuclear units to operate with additional operational flexibility while at the same time ensuring that the design bases limits are met with an appropriate margin.

SECTION 7 – REFERENCES

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4. Hanson, G.E.: "Normal Operating Controls," BAW-10122, Rev. 1., Babcock & Wilcox, Lynchburg, VA (April 1982).

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7. Deleted.

8. Deleted.

9. Letter from Virginia Electric and Power Company (L.N. Hartz) to U. S. Nuclear Regulatory Commission, "Virginia Electric And Power Company, North Anna Power Station Units 1 And 2, Proposed Technical Specifications Changes And Exemption Request Use Of Framatome ANP Advanced Mark-BW Fuel", Serial No. 02-167, March 28, 2002.

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11. North Anna Power Station Units 1 and 2 UFSAR.

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13. Ellenberger, S. L., et al.: "Design Bases for the Thermal Overpower delta-T and Thermal Overtemperature delta-T Trip Functions," WCAP-8745 (March 1977).

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18. Letter from United States Nuclear Regulatory Commission (V. Sreenivas) to Virginia Electric and Power Company (David A. Heacock), "North Anna Power Station, Unit Nos. 1 and 2– Issuance of Amendments to Revise Technical Specifications to Address Issues Identified in Westinghouse NSAL-09-5, Revision 1, and NSAL-15-1, Revision 0 (CAC. Nos. MF7186 and MF7187)", October 17, 2016.

19. Letter from Virginia Electric and Power Company (M. Sartain) to United States Nuclear Regulatory Commission, "Virginia Electric and Power Company (Dominion) North Anna Power Station Units 1 and 2 License Amendment Request to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1 and NSAL-15-1", December 10, 2015.

20. Letter from Virginia Electric and Power Company (M. Sartain) to United States Nuclear Regulatory Commission, "Virginia Electric and Power Company (Dominion) North Anna Power Station Units 1 and 2 Response to NRC Audit Regarding License Amendment Request to Address the Issues Identified in Westinghouse Documents NSAL-09-5, Rev. 1 and NSAL-15-1", June 15, 2016.

21. Letter from United States Nuclear Regulatory Commission (R. V. Guzman) to Virginia Electric and Power Company (David A. Heacock), "Millstone Power Station, Unit No. 3 - Issuance of Amendment Adopting Dominion Core Design and Safety Analysis Methods and Addressing the Issues Identified in Three Westinghouse Communication Documents (CAC NO. MF6251)", July 28, 2016.

Note: Current applicable revision of Dominion Topical Reports is maintained in the site-specific SAR.

October 21, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Cecil O. Thomas, Chief
Standardization & Special Projects Branch
U. S. Nuclear Regulatory Commission
Washington DC 20555

Serial No. 85-585
E&C/RTR:ap
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
RELAXED POWER DISTRIBUTION CONTROL (RPDC)
SUPPLEMENTAL INFORMATION

In our letter to you of December 10, 1984, Serial No. 683, we transmitted our Topical Report VEP-NE-1, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications". The Report, which was provided for review by your staff, is one of a series of Topical Reports supplying general information pertaining to the nuclear reload licensing and core follow support capabilities which have been developed at Virginia Electric and Power Company. It is our intention to use the RPDC methodology to replace the Westinghouse Constant Axial Offset Control (CAOC) which is presently used at both the Surry and North Anna stations.

On August 2, 1985, Mr. Cecil O. Thomas of the Standardization and Special Projects Branch of the Division of Licensing indicated by letter additional information was required for completion of the review. In order to expedite the review of VEP-NE-1, a meeting was held on August 8, 1985 to clarify the concerns and additional information required.

841629rtr065

Mr. Harold R. Denton
Page 2,

In response to this request as clarified in the August 8th meeting, the enclosed information is provided. Should you have any questions concerning the information, please contact us.

Very truly yours,

W. L. Stewart

Attachment

cc: Dr. J. Nelson Grace
Regional Administrator
Region II

Mr. Edward J. Butcher, Acting Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Mr. M. W. Branch
NRC Resident Inspector
North Anna Power Station

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

Mr. Harold R. Denton
Page 3,

WLB 10/21/85

bc: ~~Mr. R. H. Leasburg (w/o) - 21/OJRP~~
~~Mr. J. A. Ahladas - 4/EB~~
~~Mr. R. W. Calder - 3/EB~~
Mr. R. J. Hardwick - 5/OJRP
Mr. E. W. Harrell (w/o) - North Anna EWH by GK per 10/18/85 telecon with ETS
Mr. R. F. Saunders (w/o) - Surry
Mr. E. R. Smith, Jr. - North Anna ERS/DAH per 10/18/85 telecon with ETS
Mr. H. L. Miller - Surry HLM per telecon 10/16/85
Mr. R. F. Driscoll - Surry
Mr. D. B. Roth - North Anna
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Mr. R. W. Cross - 7/FB
NE File 2.5.8 - 2/EB
E&C Records Management NP-409.1 - 4A/EB
GOV 02 54-B
NOD Tech Library (bc original) JOH 10/21/85
Mr. L. J. Corfman LJC by JOH 10/16/85
Mr. E. T. Snow
Mr G. L. Pannell GLP 10-21-85

841629rtr065

I. Generation of Axial Xenon/Power Distributions and Delta-I vs Power Operating Domain

Question

1. What assurance is there that the axial xenon distributions via the free oscillation methodology with NOMAD result in axial power distributions that sufficiently span the delta-I power domain so that there is confidence that the most adverse power distribution/control rod conditions are available for evaluation relative to limiting Fq-LOCA, kw/ft and DNB criteria?

2. What is the sensitivity of the resultant allowable delta-I vs power domain to 1) the initiating perturbation for the free oscillation, including the power level at which the oscillation is followed, 2) uncertainties in control rod worths and reactivity coefficients, 3) control rod misalignment.

2a. How were the sensitivities determined, and how are they incorporated in the definition of the final allowable delta-I power domain?

Response

The objective of the free oscillation methodology used in RPDC is to generate a set of axial xenon distributions which vary more, as a function of core height, than could ever be expected from normal operation, including load follow maneuvers. These xenon distributions are used to perturb the axial power

distribution such that for a given power peaking limit a bounding delta-I may be obtained.

By using a divergent axial power-xenon oscillation, xenon distributions may be produced with xenon offsets (analagous to axial offset) spanning a range of +/- 25%. In contrast a load follow depletion assuming a constant axial offset control band of +/-10% in delta-I space (comparable to the usual RPDC operating bandwidth at 100% power) produces xenon distributions with offsets of approximately -8% to +6%.

These xenon distributions, when combined with the power levels and rod configurations outlined in VEP-NE-1 (Reference 1), produce approximately 9000 power distributions which have a range of delta-I values significantly greater than the allowable delta-I operating space as shown in Figure 1. Note that at 50% power, a delta-I of -40% is possible only if 90% of the core power is produced in the bottom half of the core. Figures 2 and 3 present histograms of the delta-I values calculated for a sample RPDC analysis. The histogram in Figure 2 includes all the delta-I values calculated regardless of power level while Figure 3 provides a histogram of the delta-I values generated corresponding to a specific power level (80% power in this case).

The free oscillation method is used only to generate a range of xenon distributions for use in the axial shape analysis. No data other than the xenon distributions are taken from the axial

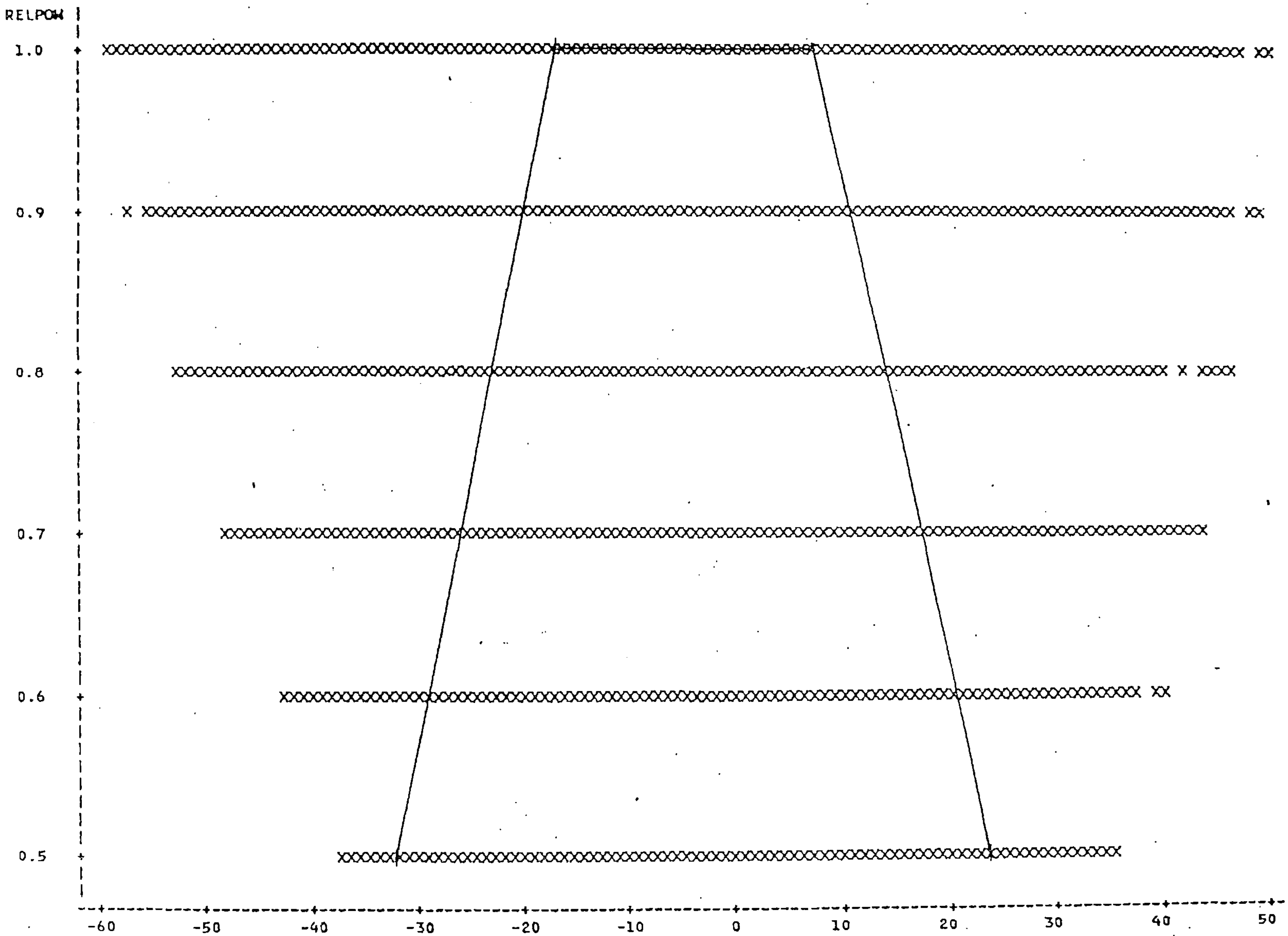
power-xenon oscillation. To initiate the axial power-xenon oscillation requires a perturbation of the 1-D model in some manner. There are a variety of ways in which to accomplish this including power level changes or rod motion. Studies performed by Combustion Engineering (Reference 2) using various perturbations to initiate the oscillation indicate the axial power shape analysis results show little sensitivity (on the order of one percent) to the initial perturbation.

In addition to the studies performed for the initiation of the power-xenon oscillation, Combustion Engineering has performed studies (Reference 2) to investigate the sensitivity of the shape analysis to variations in model parameters such as control rod worths and temperature coefficients. The results of these studies indicate very small variations in the results of the analyses, typically less than one percent. These small variations are more than compensated for by the inherent conservatism of the limits generated using extreme axial xenon distributions, core burnups with corresponding control rod worths and temperature feedbacks bounding those expected for the cycle, and a large number of axial power distributions for a wide range of delta-I values.

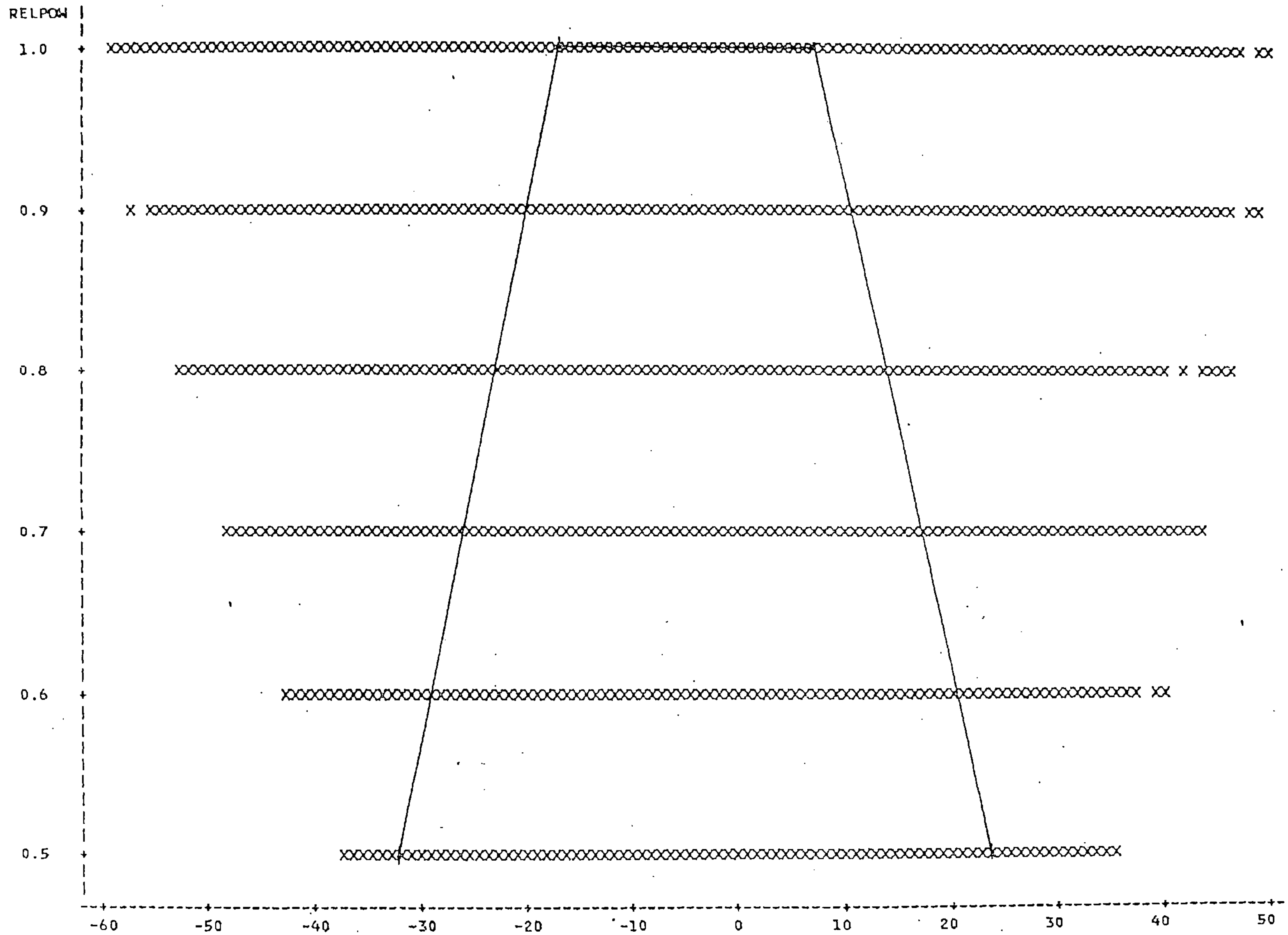
FIGURE 1

DELTA-I POINTS VS POWER LEVEL FOR S2C8 RPDC ANALYSIS

PLOT OF RELPOW*DELTAI SYMBOL USED IS X



PLOT OF RELPOW*DELTAI   SYMBOL USED IS X



% AXIAL OFFSET * RELATIVE POWER, (PT-PB)

NOTE: 4 OBS HAD MISSING VALUES OR WERE OUT OF RANGE 8378 OBS HIDDEN

FIGURE 2

HISTOGRAM OF DELTA-I FOR S2C8

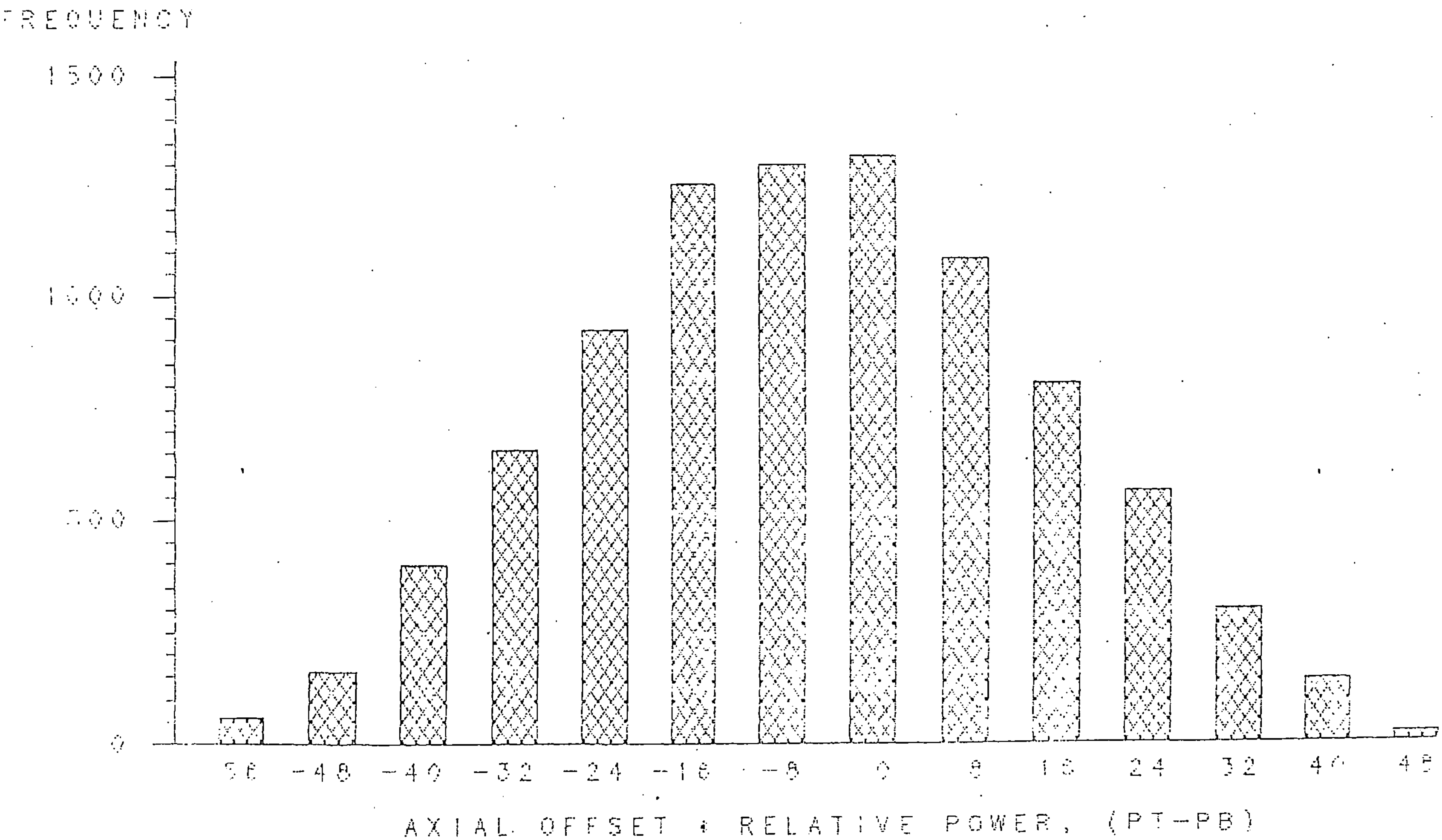
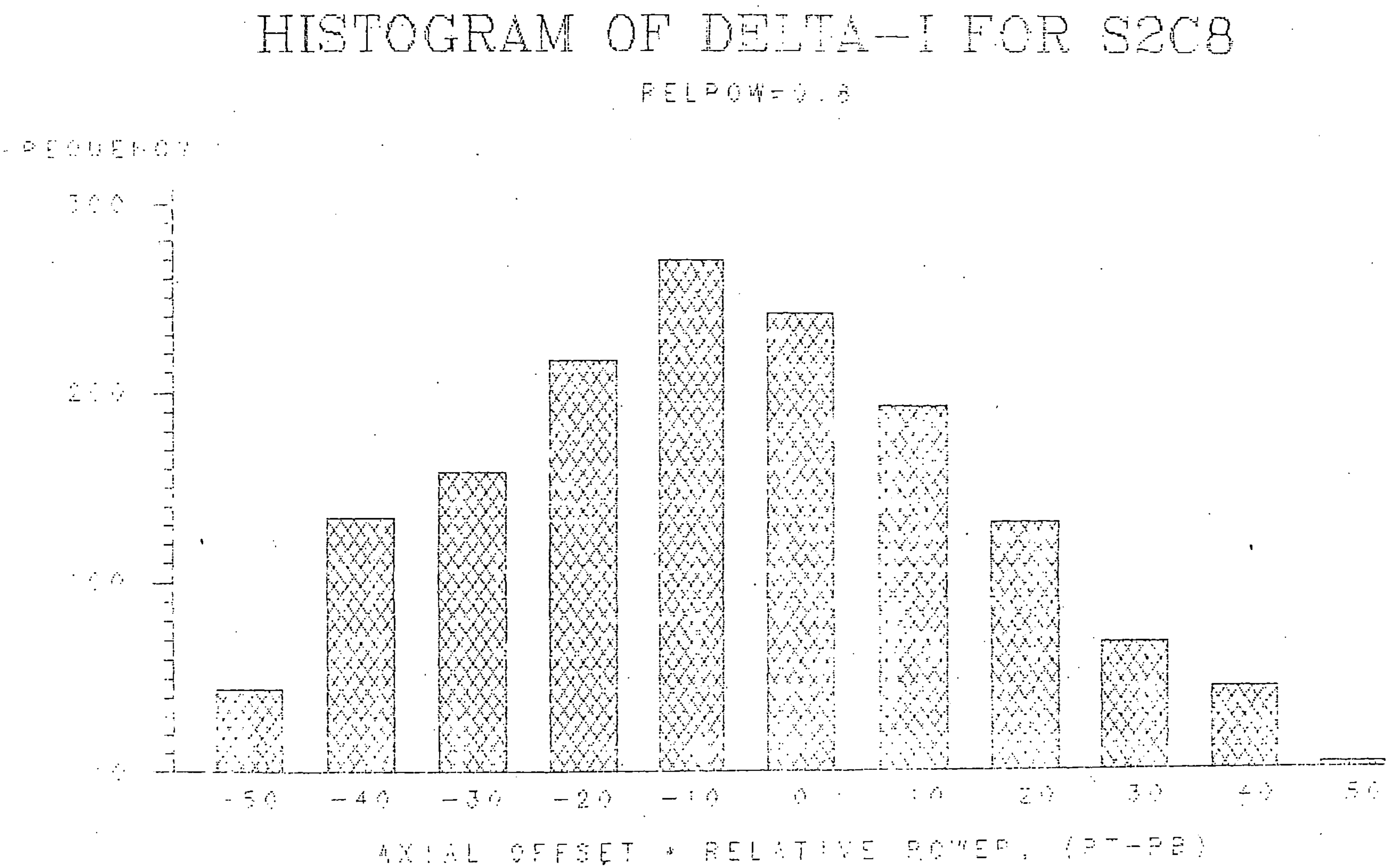


FIGURE 3


HISTOGRAM OF DELTA-I FOR S2C8
RELPOW=0.8
FREQUENCY
300
200
100
0
-50
-40
-30
-20
-10
0
10
20
30
40
50
AXIAL OFFSET * RELATIVE POWER, (PT-PB)

II. Calculation of Fq(z)

For any given axial power distribution, Fq(z) is calculated via a 1D/2D/3D synthesis approach (Eqn. (2-2))-

Fq(z) = Fxy(z) * P(z) * Xe(z) * FNU * FQE * FGR

In the above equation, Fxy(z), P(z), and Xe(z) are obtained from separate calculations, each of which have inherent uncertainties. The nuclear uncertainty factor, FNU, should properly account for all of these uncertainties when the above prescription is used. The reference for FNU in the topical report, however, refers to a Westinghouse (W) report on power peaking factors (WCAP-7912-P-A) which indicates that a 5% error should be applied to measured values of Fq. Use of this value by VEPCO for the uncertainty appropriate to a calculated Fq is not justifiable, a priori, for a number of reasons, including:

A) The 5% is based on an evaluation of measurement uncertainty and has not been justified for use as a calculational uncertainty.

B) Different methods and approaches are used by VEPCO and W.

C) The NRC review of the FLAME topical report (VEP-FRD-24A) indicates an 8% uncertainty for the calculated core average axial peaking factor. No uncertainties for P(z) calculated by NOMAD were given in the NOMAD topical report (VEP-NFE-1); however, since the adequacy of NOMAD was justified partially on the basis of comparisons to FLAME, it seems that an

uncertainty of this order should also be associated with the calculation of P(z).

D) The VEPCO PDQ-7 model used to obtain the peak rod FdH has maximum errors of -4% and tends to underpredict this quantity; Fxy(z) is based on the PDQ calculated FdH, and FLAME values for the relative nodal power and the core average axial power distribution.

Question

3) In view of the above considerations, justify the use of FNU =1.05 in the calculation of Fq(z).

4) How is the presence of azimuthal tilts allowable by the technical specifications accounted for in the determination of bounding values of Fxy(z)?

Response

Topical Report VEP-FRD-45A, "Nuclear Design Reliability Factors", (Reference 3) documents a statistical analysis performed to derive the calculational uncertainty for the calculation of Fq(x,y,z) using the Virginia Power FLAME model and a PDQ 2-D coarse mesh correction. These uncertainties were derived from a comparison of measurements and predictions based on one sided 95%/95% upper tolerance limits. This analysis resulted in uncertainty factors of 1.069 and 1.072 for Surry and North Anna, respectively, as compared to the 1.07625 value (FNU

* FGR) used in the RPDC analysis.

However, the RPDC analysis uses the NOMAD code (Reference 4) to generate the axial power shape in the calculation of Fq(z). In order to determine the calculational uncertainty associated with the use of NOMAD for the calculation of peak Fq(z) values, a new statistical analysis was performed.

The data base for the analysis consisted of flux maps from the last two completed cycles of operation for each of the Virginia Power nuclear units: North Anna 1 Cycles 3 and 4, North Anna 2 Cycles 2 and 3, Surry 1 Cycles 6 and 7, and Surry 2 Cycles 6 and 7. Eight to twelve flux maps from each cycle were used for a total of 86 maps, 43 for Surry and 43 for North Anna. It should be noted that the current NOMAD analysis compared nearly three times as many flux maps as the FLAME analysis in VEP-FRD-45A. Calculated Fq(z) versus measured Fq(z) were compared at five core heights per map for Surry and six for North Anna, as described in VEP-FRD-45A. This provided a data base of 215 Fq(z) values for Surry and 258 values for North Anna. This is smaller than the number of values in the original FLAME comparison (VEP-FRD-45A). However, in the NOMAD analysis, peak Fq(z) for the entire core is compared, whereas in the FLAME analysis, Fq(x,y,z) for each measured assembly was compared, resulting in approximately 40 times as much data per flux map.

Based upon the above comparison of predicted Fq(z) calculated using the methodology described in VEP-NE-1 versus the measured

Fq(z) produced by the INCORE code (Reference 5), uncertainty factors of 1.067 and 1.075 were calculated for Surry and North Anna, respectively. These values represent a 95%/95% upper tolerance limit based on a methodology similar to that used in Reference 5.

Virginia Power technical specifications allow azimuthal tilts of up to 2 percent. The analysis performed above used measured data taken on a full core basis which included any tilt effects occurring at the time of measurement. The comparison of measured and predicted values of Fq(z) is therefore a comparison of predicted versus measured values which include tilt effects. Since the tilts experienced by the units cover the range of the technical specifications limits, the uncertainty calculated above is sufficient to insure that bounding values of peaking factors are being generated.

III. Radial Xenon Redistribution Factor

The radial xenon redistribution factor Xe(z) appears to be based on a single 3-D transient xenon calculation using the FLAME code.

Question

5. What assurance is there that the results from this calculation are bounding for all applications?

6. What is the sensitivity of the results to different initial/ final, power/control rod configurations?

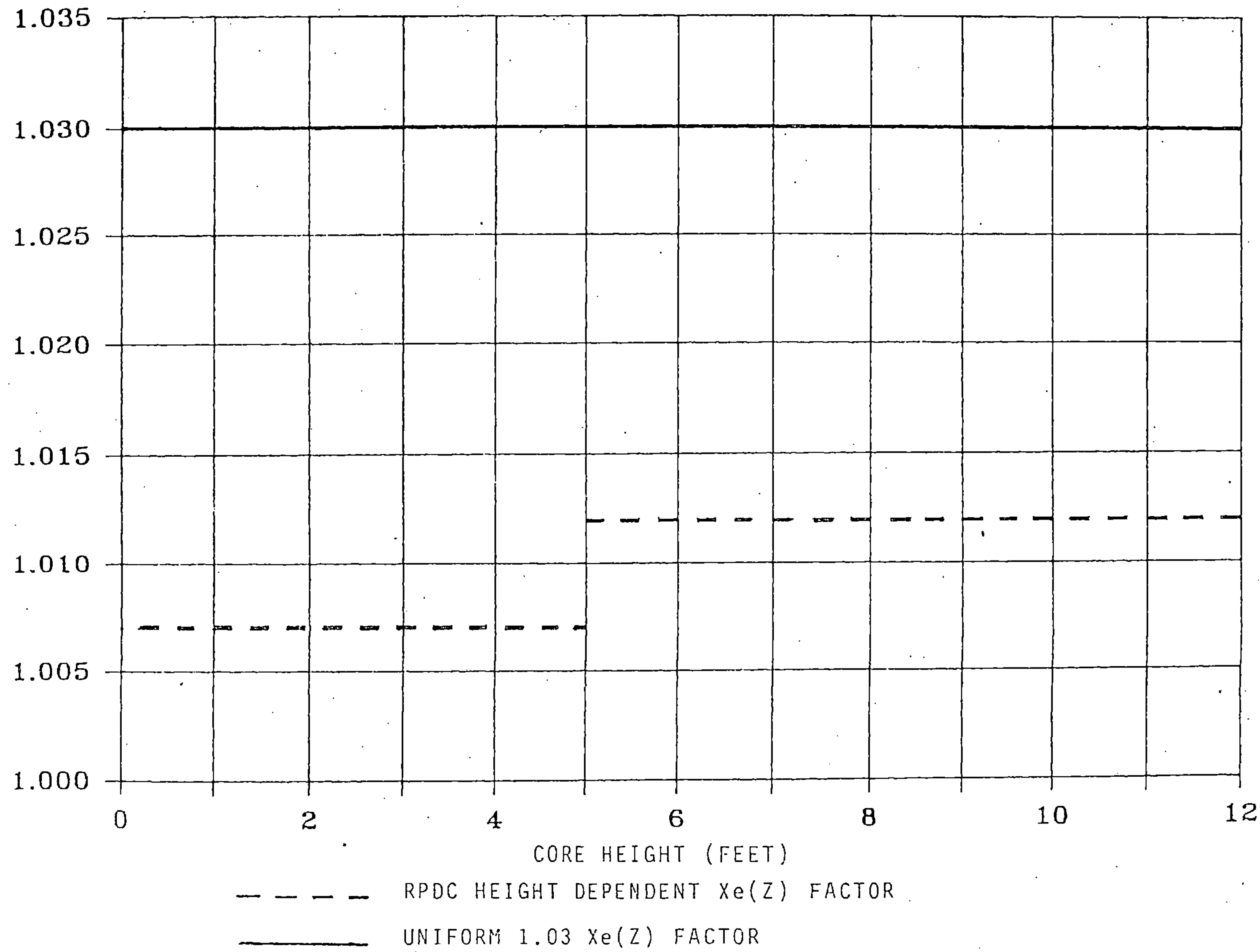
7. What is the typical magnitude of this factor relative to the previous uniform value of 1.03?

Response

The radial xenon redistribution factor is to account for increases in Fq(z) caused by radial redistribution of xenon due to power redistribution from rod movement. A bounding factor as a function of core height was determined using 3-D analyses by calculating the increase in Fxy(z) due to radial xenon redistribution for a number of cycles, times in cycle life, and different initiating conditions (power level, rod configuration), which may occur during normal operation. The radial xenon redistribution factor Xe(z) was chosen such that it bounded all increases in Fxy(z) determined in the above analyses. This factor is shown in Figure 4 relative to the

previous value of 1.03 which had been calculated using a 2-D model. As would be expected, the increase in Fxy(z) due to xenon redistribution showed the greatest sensitivity to the amount of control rod insertion during the precondition phase of the analysis as the control rods cause significant changes in the radial power distribution.

Figure 4
RADIAL XENON REDISTRIBUTION FACTOR XE(Z)



IV. Permissible Delta-I Power Domain

Question

8. What is the magnitude of, and basis for, the uncertainties applied in converting the NOMAD generated permissible delta-I power domain to the operating domain monitored by the excore detectors?

Response

The excore detectors are calibrated to the delta-I measured by the incore detectors to within a specified tolerance. To incorporate this into the allowable delta-I operating domain, the permissible delta-I domain generated in the RPDC analysis is reduced by the excore detector calibration uncertainty. For North Anna and Surry, the Technical Specification delta-I domain will allow for a maximum excore calibration uncertainty of 3%.

Question

9. A statement is made in connection with Tech. Spec. 3/4.2.2 and 3/4.2.3 that N(z) "was determined from expected power control maneuvers ...," while the text implies that N(z) is based on the Condition I analyses which employ the free oscillation methodology. Please clarify.

10. What is the uncertainty in the N(z) function?

11. What is the sensitivity of the N(z) function to possible variations in impacting parameters?

Response

The N(z) factor is determined using the condition I (normal operation) RPDC generated Fq(z) distributions within the allowable delta-I operating space plus the excore-incore measurement tolerance. These distributions are produced using xenon distributions which are more extreme than any which could be generated using load follow maneuvers. As such these distributions cover and bound the delta-I space much more adequately than could be done through the use of load follow maneuvers.

As the Fq(z) can only be measured at equilibrium conditions the N(z) factor is used to account for non-equilibrium effects the reactor may see under normal operation (i.e., power level changes, rod motion, xenon transients, etc.). The N(z) factor is calculated as the maximum potential increase in local peaking from the equilibrium value which could occur during transient, non-equilibrium operation within the allowed delta-I operating space plus the excore-incore measurement tolerance. Being the maximum increase in Fq(z), rather than a nominal value, the N(z) is a bounding function and the application of an uncertainty would be unnecessarily conservative.

The delta-I operating space generated using the RPDC methodology takes into account changes in power level, rod configuration, xenon distribution, and core burnup. By using the RPDC axial power distributions, the calculation of N(z) inherently includes the sensitivity to these parameters.

Question

12. Tech. Spec. 3/4.2.2 and 3/4.2.3 implies an FdH limit of 1.55/1.08 (=1.435). The calculation of Fxy(z) via Eqn.(3.5-1) of the NOMAD report, however, relies purely on calculated values (even if they are lower than 1.435). If FdH = 1.435 represents a limit, it should be accounted for in the generation of Fq(z) and the associated delta-I/power operating space.

    Please comment.

Response

Virginia Power technical specifications provide limits on both FdH (nuclear enthalpy rise hot channel factor) and Fq (heat flux hot channel factor). However, the limits are set to provide assurance of fuel integrity from different standpoints. The FdH limit is to ensure the minimum DNBR in the core is maintained at greater than or equal to 1.30 during normal operation and in short term transients. Limits on Fq are to provide assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit is not exceeded.

The RPDC reload analysis determines the impact of these parameters on the delta-I operating space independently of each other. The initial delta-I space is generated through the Fq LOCA analysis. This space is then verified to be acceptable from a DNB standpoint by performing analyses using the relative axial power distributions generated with RPDC and applying the technical specification delineated FdH limit (i.e., 1.550) to determine the minimum DNBR.

The calculation of Fq(z) is an estimate of a 3-D quantity by a 1-D/2-D synthesis procedure. Fq(z) is assumed to be the product of two components, the core average axial power density for plane z and the maximum ratio of peak power density to average power density for plane z:

$$Fq(z) = P(z) \times \text{maximum } Fxy(z)$$

The P(z) for this calculation is obtained from the 1-D code NOMAD which is normalized to the 3-D FLAME code. The calculation of the maximum Fxy(z) is performed using the 3-D FLAME code and applying a 2-D fine mesh correction to the coarse mesh solution for Fxy(z):

$$Fxy(z) = \frac{Fq(x,y,z)}{P(z)} \times \frac{FdH(x,y)}{RPD(x,y)}$$

where,

Fq(x,y,z) = relative power in node (x,y,z) from FLAME

P(z) = core average axial power in plane z from FLAME

FdH(x,y) = peak pin power in assembly (x,y) from PDQ discrete

RPD(x,y) = relative power in assembly (x,y) from FLAME

$$\frac{FdH(x,y)}{RPD(x,y)}$$ = coarse mesh correction forced to be greater than or equal to 1.0

The above Fxy(z) calculation is performed for each assembly in the core at each axial plane. The maximum Fxy(z) in each plane is then used in the synthesis calculation providing a conservative value for Fq(z).

As can be seen above the FdH(x,y) used in the above Fxy(z) synthesis is only a coarse mesh correction factor and should not be confused with the FdH used for DNB analysis which is analyzed independently of the Fq analysis. However, this method does result in the use of Fxy(z) values which, if integrated over the length of the core, would result in an FdH value greater than the maximum FdH calculated for the design.

Question

Possible Typo; TS4.2.2.2.f.1 (p 3/4 2-7)

P in denominator of second equation should be 0.5

Response

The P in the denominator of the equation should be 0.5. This error was corrected in our actual Technical Specification change request dated September 18, 1985.

REFERENCES

1. Basehore, K. L., et al. : "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications" VEP-NE-1 (October 1984).

2. "C-E Setpoint Methodology," CENPD-199-NP Rev. 1-NP, Combustion Engineering Inc., Windsor, CT (March 1982).

3. Miller, J. G., : "Vepco Nuclear Design Reliability Factors", VEP-FRD-45A (October 1982).

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

5. W. D. Leggett III and L. D. Eisenhart, "The INCORE Code", WCAP-7149, December 1967, (Westinghouse).

Serial No. 17-476
Docket Nos.: 50-280/281/338/339/423

ATTACHMENT 2

TOPICAL REPORT VEP-FRD-42-A
REVISION 2, MINOR REVISION 2
RELOAD NUCLEAR DESIGN METHODOLOGY

Millstone Power Station, Unit 3
North Anna Power Station, Units 1 and 2
Surry Power Station, Units 1 and 2

VEP-FRD-42-A, Revision 2, Minor Revision 2

RELOAD NUCLEAR DESIGN METHODOLOGY

By

NUCLEAR ENGINEERING & FUEL STAFF
DOMINION ENERGY
RICHMOND, VIRGINIA

OCTOBER 2017

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Prepared By:	J. T. Krizmanich	Nuclear Safety Analysis Design
Reviewed By:	D. T. Smith	Nuclear Core Design
Reviewed By:	T. N. Ho	Nuclear Safety Analysis Design
Recommended For Approval:	T. W. Schleicher	Supervisor – Nuclear Core Design II
Recommended For Approval:	A. H. Nicholson	Supervisor – Nuclear Core Design
Recommended For Approval:	S. A. Luchau	Supervisor – Nuclear Safety Analysis Design
Recommended For Approval:	C. B. Laroe (A.H. Nicholson for)	Manager – Nuclear Engineering and Fuel
Approved:	W. M. Adams	Director – Nuclear Engineering and Fuel

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

June 11, 2003 SERIAL # 03-381

REC'D JUN 19 2003

NUCLEAR LICENSING

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

SUBJECT: VIRGINIA ELECTRIC AND POWER COMPANY - ACCEPTANCE OF TOPICAL REPORT VEP-FRD-42, REVISION 2, "RELOAD NUCLEAR DESIGN METHODOLOGY," NORTH ANNA AND SURRY POWER STATIONS, UNITS 1 AND 2 (TAC NOS. MB3141, MB3142, MB3151, AND MB3152)

Dear Mr. Christian:

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

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

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

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

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

Sincerely,

Scott M. Moore

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

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

Enclosure: Safety Evaluation

cc w/encl: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

NORTH ANNA AND SURRY POWER STATIONS, UNITS 1 AND 2

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

1.0 INTRODUCTION

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

2.0 REGULATORY EVALUATION

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

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

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

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

3.0 TECHNICAL EVALUATION

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

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

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

3.1 Analytical Models and Methods

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

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

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

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

PDQ Two-Zone Model

By letter dated October 1, 1990, VEPCO initially requested approval of the PDQ Two-Zone model in order to support the use of axially zoned flux suppression inserts in Surry, Units 1 and 2. The PDQ Two-Zone model is a three-dimensional, coarse mesh model that was developed to replace the PDQ Discrete model described in Topical Report VEP-FRD-42,

Revision 1-A. The PDQ Two-Zone model is used to calculate three-dimensional power distributions, delayed neutron data, radial and axial peaking factors, assembly-wise burnup and isotopic concentrations, differential and integral rod worths, differential boron worth and boron endpoints, xenon and samarium worth, and core average reactivity coefficients such as temperature and power coefficients. In addition, PDQ is used to generate predicted power and flux distributions in order to translate thimble flux measurements into measured power distributions.

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

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

<u>NOMAD</u>

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

axial power peaking factors, differential control rod bank worth, integral control rod worth as a function of bank position, fission product poison worth, and reactivity defects.

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

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

RETRAN

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

In a letter dated August 10, 1993, VEPCO informed the NRC staff of various modifications and updates to its RETRAN model, and that these changes were to be implemented under the provisions of 10 CFR 50.59. This letter described several changes to the VEPCO RETRAN

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

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

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

<u>Core Thermal-Hydraulics and Nuclear Design Models</u>

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

changes associated with the Framatome ANP Advanced Mark-BW fuel are within the restrictions and limitations of the VEPCO core design and safety analysis codes. The NRC staff has reviewed this information provided by VEPCO and agrees that the Framatome ANP Advanced Mark-BW fuel features are within the modeling capability of the VEPCO core design analytical models. As such, the NRC staff finds that this modeling capability is applicable to both Westinghouse and Framatome ANP Advanced Mark-BW fuel types.

Analytical Methods

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

Analytical Model and Method Approval Process

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

3.2 Reload Design

The overall objective of core reload design is to determine fuel enrichment, feed batch size, and a core loading pattern that fulfills cycle energy requirements while satisfying the constraints of

the plant design basis and safety analysis limits. Topical Report VEP-FRD-42, Revision 2, provides a general description of the reload design methodology used for the North Anna and Surry Power Stations, and is largely consistent with the NRC-approved methodology of Topical Report VEP-FRD-42, Revision 1-A. This VEPCO methodology divides the reload design process into three phases: 1) core loading pattern design and optimization, 2) determination of core physics related key analysis parameters for reload safety analysis, and 3) design report, operator curve, and core follow predictions.

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

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

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

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

- analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits,
- use of bounding key parameter values in reference safety analyses,
- recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data,
- representation of key fuel features via detailed inputs in core design and safety analysis models, and
- fuel is modeled using approved critical heat flux correlations demonstrated to be applicable and within the range of qualification and identified in the plant COLR section of the TS.

4.0 CONCLUSIONS

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

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

5.0 CONDITIONS AND LIMITATIONS

Prior to the use of the Topical Report VEP-FRD-42, Revision 2, methodology for fuel types other than Westinghouse and Framatome ANP Advanced Mark-BW fuel, VEPCO must confirm that the impact of the fuel design and its specific features can be accurately modeled with the VEPCO nuclear design and safety analysis codes and methods as discussed in its submittal dated May 13,2002. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42, Revision 2, these proposed changes are required to be submitted for prior NRC review and approval.

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

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

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

6.0 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

13. Letter from C. O. Thomas (USNRC) to T. W. Schnatz (UGRA), "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, RETRAN - A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems, and EPRI NP-1850-CCM, RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," dated September 4, 1984.

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

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

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

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

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

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

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

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

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Date: June 11, 2003

CLASSIFICATION/DISCLAIMER

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PREFACE

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

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

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

Revision 2, Minor Revision 1 of this topical report adds CMS as an acceptable analytical model for reload design and safety analysis.

Revision 2, Minor Revision 2 of this topical report revises the presented methodology to include applicability to Millstone Unit 3 and to add CMS5 as an acceptable analytical model for reload design and safety analysis. Numerous grammatical and editorial corrections have also been made throughout the document.

Efforts of the following contributors to past versions of this document are hereby acknowledged:

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SECTION 1.0 - INTRODUCTION

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

The Dominion Energy methodology for designing a reload core at its nuclear units is an iterative process. The process involves determining a fuel loading pattern which provides the required total cycle energy and then demonstrating through analysis or evaluation that the plant will continue to meet all applicable safety criteria after considering the changes associated with the reload core. Should the characteristics of the proposed loading pattern cause any safety analysis criteria not to be met for operation within the current operating requirements, one of two remedies is selected. Either the loading pattern is revised or changes are made in the operating requirements (Technical Specifications or Core Operating Limits Report (COLR), as applicable). Such changes ensure that plant operation will satisfy the applicable safety analysis criteria for the proposed loading pattern.

The Nuclear Regulatory Commission has approved the use of Dominion's Reload Nuclear Design Methodology for Millstone Unit 3 in (Reference 18). The methodology described herein has been updated to reflect this approval. This report presents the methodology that may be employed by Dominion Energy for performing a nuclear reload design analysis at the North Anna, Surry, and Millstone Unit 3 Power Stations. It covers analytical models and methods, reload nuclear design, reload safety analysis, and an overview of analyzed accidents and key parameter derivations. This revision also incorporates generic reference to approved methodologies that are applicable to core thermal-hydraulic analyses. The COLR section of the plant Technical Specifications provides a listing of such applicable methodologies. The generic citation of these methodologies is intended to minimize duplicate NRC Staff review effort, since review and approval of any such methodologies would precede their listing in the COLR.

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

SECTION 2.0 - ANALYTICAL MODELS AND METHODS

2.1 Analytical Models

The major analytical models currently approved for use and utilized by Dominion Energy for reload design and safety analysis are:

1. Virginia Power PDQ Two Zone and NOMAD Models [Obsolete]
2. Studsvik Core Management System (CMS) – Casmo4 / Simulate3
3. Studsvik Core Management System 5 (CMS5) – Casmo5 / Simulate5
4. VEPCO RETRAN model
5. Core Thermal-Hydraulics models

Dominion Energy has replaced all previous PDQ models (PDQ Discrete and PDQ Two Zone / NOMAD, References 1-5) with two major implementations of the Studsvik Core Management System. The original CMS methodology, which consists primarily of the Casmo4 / Simulate3 codes, was first approved for use at North Anna and Surry power stations through Reference 16 and then approved for use at Millstone Unit 3 through Reference 18. The CMS5 methodology, which consists primarily of the Casmo5 / Simulate5 codes, was generically approved by the USNRC (Reference 19) for Pressurized Water Reactors (PWRs) and is adopted into the licensing basis for North Anna, Surry and Millstone Unit 3 through the 10CFR50.59 / Generic Letter 83-11, Supplement 1 process. Both the CMS and CMS5 core models were extensively validated by comparing calculated data to higher order calculations and to measurements from Surry and North Anna Units 1 & 2, Millstone Units 2 & 3, and Kewaunee. As described in References 16 and 19, each model has sufficient flexibility such that minor fuel assembly design differences (bounded by the Safety Evaluation for each methodology) can be adequately accounted for using model design input variables.

Use of the RETRAN Code and models, as originally approved for reference in licensing applications is documented in Reference 6. Supplemental details concerning the models used with the RETRAN Code were provided to the USNRC in an informational letter, Reference 7.

These models were implemented under the provisions of 10CFR50.59. The applicable thermal-hydraulic codes and models are listed in the COLR section of the plant Technical Specifications. These analysis models have been used successfully to model plant transient response for core reloads, as well as various changes to plant configuration including core uprate and steam generator replacement.

2.1.1 Virginia Power PDQ Two Zone and NOMAD Models

[DELETED – Obsolete Analytical Models]

2.1.2 Studsvik Core Management System (CMS and CMS5)

The CMS and CMS5 codes are very similar software packages; even to the extent that they are interoperable (e.g., Casmo4 cross sections can be read by either Simulate3 or Simulate5). The principal computer codes in CMS are Casmo4 and Simulate3. The principal computer codes in CMS5 are Casmo5 and Simulate5. In each case either the Cmslink or Cmslink5 code provides the coupling between Casmo and Simulate. More complete descriptions of each model and their auxiliary codes may be found in References 16 and 19. The following general descriptions of Casmo, Simulate, and Cmslink apply to all versions.

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

In order to generate a neutronic data library for Simulate a series of Casmo depletions and branch cases is required. This series of calculations is defined within CMS as the "Simulate Case Matrix." This case matrix consists of a series of depletions and instantaneous branch cases vs. exposure as a function of varied boron concentration, moderator temperature, fuel temperature,

and shutdown cooling time, as well as cases with control rods and without removable burnable poison in guide tube locations. Cmslink is a linking code that processes Casmo card image files into a binary formatted nuclear data library for use by Simulate.

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

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

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

2.1.3 VEPCO RETRAN Models

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

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

2.1.4 Core Thermal-Hydraulics Models

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

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

2.2 Analytical Methods

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

2.2.1 Core Depletions

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

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

2.2.2 Core Reactivity Parameters and Coefficients

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

2.2.2.1 Reactivity Coefficients and Defects

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

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

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

2.2.2.2 Differential Boron Worth

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

2.2.2.3 Delayed Neutron Data

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

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

2.2.2.4 Fission Product Poison Worths

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

2.2.3 Core Reactivity Control

The full length control rods control relatively rapid reactivity variations in the core. The control rods are divided into control banks and shutdown banks. The control banks are used to compensate for core reactivity changes associated with changes in operating conditions such as temperature and power level and are moved in a fixed sequential pattern to control the reactor over the power range of operation. The shutdown banks are used to provide shutdown reactivity.

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

2.2.3.1 Integral and Differential Rod Worths

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

2.2.3.2 Soluble Boron Concentrations

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

2.3 Analytical Model and Method Approval Processes

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

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

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

SECTION 3.0 - RELOAD DESIGN

3.1 Introduction

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

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

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

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

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

be met, the reload design may be altered or new operating limits may be specified in the COLR or Technical Specifications.

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

3.2 Core Loading Pattern Design and Optimization

3.2.1 Design Initialization

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

The design basis information to be reviewed includes:

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

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

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

This review will establish or define:

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

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

3. Reload design schedules.

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

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

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

7. Expected plant operating conditions.

3.2.2 Fuel Loading and Pattern Determination

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

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

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

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

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

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

3.3 Nuclear Design Aspects of Reload Safety Analysis

3.3.1 Introduction

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

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

3.3.2 Safety Analysis Philosophy

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

Plant safety is demonstrated by accident analysis, which is the study of nuclear reactor behavior under accident conditions. Accident analyses are usually performed in the initial stages of plant design and documented in the FSAR. The accident analyses for North Anna, Surry, and Millstone Unit 3 are typical in that the NSSS vendor performed the complete FSAR analysis. The four categories of plant conditions based on their anticipated frequency of occurrence and potential for public harm are described in References 10, 11, and 17. The accident analyses consider all relevant aspects of the plant and core including the operating procedures and limits on controllable plant parameters (Technical Specifications) and the engineered safety, shutdown, and containment systems.

There are two stages in the typical safety analysis process, and these stages are applicable to either initial plant design analyses or analyses that may be initiated during reload core design. First, steady state nuclear calculations are performed for the core conditions assumed in the accident analysis. The nuclear parameters derived from these calculations are called the core physics related key analysis parameters and serve as input to the second stage. The second stage is the actual dynamic accident analysis, which yields the accident results that are applicable for these key analysis parameter values. The accident analyses are transient calculations that usually model the core nuclear kinetics and those parts of the plant systems, which have a significant impact on the events under consideration.

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

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

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

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

Therefore, the overall process is as follows:

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

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

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

3.3.3 Non-Specific Key Parameters

Non-specific key parameters are derived by evaluating core characteristics for conditions bounding those expected to occur during the reload cycle to ensure that sufficiently limiting values of the parameter are determined. These conditions include conservative assumptions for such core parameters as xenon distributions, power level, control rod position, operating history, and burnup. These parameters are designated with '(NS)' in Table 2. Each non-specific key

parameter generally serves as safety analysis input to several accidents including the accidents that also require specific key parameters, such as rod ejection. In addition, numerical uncertainty factors that are appropriate to the models being used are applied to the calculated parameter.

3.3.3.1 Rod Insertion Limits

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

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

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

When tentative RIL lines have been selected by the method just outlined, they are then checked to see that they satisfy all of the other evaluation requirements. If any basis is not satisfied by the

tentative insertion limits, the insertion limits are raised until the most limiting basis is satisfied. These limits are then checked against the COLR. If these RIL lines exceed those in the COLR, the COLR is revised accordingly.

3.3.3.2 Shutdown Margin

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

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

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

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

3.3.3.3 Trip Reactivity Shape

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

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

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

3.3.3.4 Reactivity Coefficients

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

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

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

3.3.3.5 Neutron Data

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

The key analysis parameter is the β_{eff}, which is the product of the total β and the importance factor. The importance factor reflects the relative effectiveness of the delayed neutrons for causing fission. For some transients, it is conservative to use the minimum expected value of β_{eff}, while for others, the maximum expected value is more conservative. The use of conservatively

large versus small β_{eff} values is treated on an event by event basis. β_{eff} is calculated at the times in cycle life that would produce the bounding values for the cycle (typically BOC and EOC).

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

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

3.3.3.6 Power Density, Peaking Factors

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

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

3.3.4 Specific Key Parameters

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

3.3.4.1 Uncontrolled Control Rod Bank Withdrawal

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

The following assumptions and conservatisms are used:

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

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

3.3.4.2 Rod Misalignment

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

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

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

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

The dropped RCCA(s) event (dropped rod or dropped bank) is conservatively evaluated using the methodology described in WCAP-11394-P-A (Reference 15). Dominion Energy acquired the transient databases and methodology information necessary to perform the dropped rod analyses of Reference 15 from Westinghouse. Evaluations were performed which demonstrated the

applicability of the methodology, the correlations, and the transient database for the analysis of the dropped rod event for the North Anna, Surry, and Millstone Unit 3 Power Stations. This methodology for the evaluation of the dropped rod(s) event has been implemented for the North Anna, Surry, and Millstone Unit 3 Power Stations.

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

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

3.3.4.3 Rod Ejection

The rod ejection accident results from the postulated mechanical failure of a control rod mechanism pressure housing such that the coolant system pressure ejects the control rod and drive shaft to the fully withdrawn position. This results in rapid reactivity insertion and high

peaking factors. Rod ejections are analyzed at the beginning and end of the cycle at hot zero power and hot full power.

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

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

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

The rod ejection parameter derivation is performed in a conservative manner. Although the rod ejection accident is limited by Doppler feedback, the key analysis parameters are derived with all feedback frozen. Conservatism is ensured by calculating all physics parameters at steady state conditions using the "adiabatic assumption." This assumption asserts that any fuel damage which might occur during the transient takes place in a small time interval immediately following the ejection of the rod and before the thermal-hydraulic feedback effects of the core become

important. This freezing of the core feedback effects leads to larger values of the total power peaking factor and ejected rod worth than would otherwise be expected in the transient.

3.3.4.4 Steamline Break

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

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

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

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

First, a detailed nuclear calculation is performed at the limiting time in cycle, HZP power conditions with all rods fully inserted, except the highest reactivity worth stuck rod. These conditions are conservative initial assumptions for steambreak (see References 10, 11, and 17).

Next, conditions including power, non-uniform inlet temperature distribution, pressure, and flow (derived from the RETRAN code output data at the point where the minimum DNBR may occur) are input, and peaking factors and axial power distributions are generated. The stuck rod is assumed to occur in the coldest quadrant to maximize reactivity insertion.

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

3.3.4.5 LOCA Peaking Factor Evaluation

A loss of coolant accident (LOCA) is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system. The LOCA reload evaluation methodology that is employed by Dominion Energy is consistent with the fuel vendor methodology used for establishing and validating the operational limits for allowable core power distributions. A description of the reload validation methodology can be found in References 9 and 14.

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

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

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

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

3.3.4.6 Boron Dilution

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

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

3.3.4.7 Overpower Evaluations

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

3.3.5 Non-Nuclear Design Key Parameters

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

3.4 Reload Safety Evaluation Process

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

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

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

TABLE 1

EVALUATED ACCIDENTS

CONDITION II EVENTS

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

CONDITION III EVENTS

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

CONDITION IV EVENTS

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

TABLE 2

KEY ANALYSIS PARAMETERS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

4) Delayed neutron data and prompt neutron lifetime;

5) Relative power distributions at various core conditions;

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

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

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

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

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

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

SECTION 4.0 - SUMMARY AND CONCLUSIONS

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

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

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

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

Sometimes reanalysis will produce unsatisfactory results and other steps may have to be taken. Technical Specifications changes, COLR changes, or core loading pattern changes are typical adjustments that may be required. Raising the rod insertion limits, in order to reduce the ejected

rod Fq and worth, is an example of a COLR change. If Technical Specifications changes are necessary to keep key parameters bounded, these changes must be approved by the NRC in accordance with 10CFR50.59 prior to implementation at the plant. In addition, loading pattern adjustments may be required to bring some key parameters within the current limits or reduce the size of the deviation.

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Note: Current applicable revision of Dominion Topical Reports is maintained in the site-specific UFSAR/FSAR.

APPENDIX A

RAI Questions and Responses Set # 1

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

Total Pages : 27

Attachment

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

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

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

Question 1:

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

Response:

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

- Analysis framework in which safety analyses establish the acceptable values for reload core key parameters, while nuclear and fuel design codes confirm each core's margin to the limits

- Use of bounding key parameter values in reference safety analyses

- Recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data

- Representation of key fuel features via detailed inputs in core design and safety analysis models

- Fuel is modeled using approved critical heat flux (CHF) correlations demonstrated to be applicable and within the range of qualification

The Dominion reload design methodology focuses upon determining appropriately conservative values for two types of parameters: 1) the bounding value for key parameters assumed in the safety analyses and 2) the values for these same key parameters calculated for each reload core. The first parameter set constitutes the allowable limits for which the existing safety analyses remain valid. The reload values are determined for each specific core with the objective of confirming that they remain within the limit values. Application of this methodology to alternate fuel types would be accomplished in a fashion that preserves this fundamental approach. Prior to the use of

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

Question 2:

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

Response:

Core Design Models

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

displacement. The macroscopic cross section effect is transferred to the NOMAD model from PDQ. Similarly, cross sections in the PDQ model are a function of fresh fuel isotopic content; therefore, the density effects are also directly modeled.

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

RETRAN Models

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

Fuel properties

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

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

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

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

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

Core Geometry Input

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

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

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

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

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

Question 3:

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

Response:

General comment

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

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

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

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

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

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

Application Procedures

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

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

Quality Assurance/Change Control

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

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

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

The Framatome ANP overlays were developed from the following data:

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

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

Comparison Calculations

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

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

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

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

Reference:

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

Question 4:

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

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

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

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

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

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

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

Response to 4a:

<u>PDQ Two-Zone Model</u>

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

Reference 4.1 requested approval of the 3-D coarse mesh PDQ model (the two-zone model) by the end of the 1st Quarter, 1991 to support the use of axially zoned flux

suppression inserts (FSI's) in Surry Unit 1 Cycle 12.

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

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

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

The relevant discussion is as follows:

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

- Departures from methods of evaluation that are not described, outlined or summarized in the UFSAR (such changes may have been screened out as discussed in Section 4.2.1.3).

- Use of a new NRC-approved methodology (e.g., new or upgraded computer code) to reduce uncertainty, provide more precise results or other reason, provided such use is (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable SER. The basis for this determination should be documented in the licensee evaluation.

- Use of a methodology revision that is documented as providing results that are essentially the same as, or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of an SER".

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

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

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

Conservative vs. Nonconservative Results

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

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

Essentially the Same

Licensees may change one or more elements of a method of evaluation such that results move in the nonconservative direction without prior NRC approval, provided the revised result is 'essentially the same' as the previous result. Results are 'essentially the same' if they are within the margin of error for the type of analysis being performed. Variation in results due to routine analysis sensitivities or calculational differences (e.g., rounding errors and use of different computational platforms) would typically be within the analysis margin of error

and; thus, considered 'essentially the same.' For example, when a method is applied using a different computational platform (mainframe vs. workstation), results of cases run on the two platforms differed by less than 1%, which is the margin of error for this type of calculation. Thus, the results are essentially the same, and do not constitute a departure from a method that requires prior NRC approval.

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

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

NOMAD

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

Letter 96-319 (Reference 4.4) documented the NOMAD code and model update. These changes were necessitated by the transition to 3-D PDQ (see discussion above). The NOMAD flux solution and axial nodalization were not altered. The updated NOMAD model was qualified against plant data and its fidelity to the data was found to be as good as or better than that of the original code and model. The Nuclear Reliability Factors currently applied in reload analyses were shown to remain appropriate and reload results obtained with the updated model are essentially the same as those

obtained with the previous version. As such, the code and model updates do not constitute a change in the approved methodology of VEP-FRD-42 or the Code as described in VEP-NFE-1-A (see the discussion of NEI 96-07, Section 4.3.8, above).

TIP/CECOR

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

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

RETRAN

Dominion's reload methodology incorporates the RETRAN-02 code. RETRAN-02 was generically approved by the NRC in a letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

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

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

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

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

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

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

The VEP-FRD-41 SER further stated:

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

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

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

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

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

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

Response to 4b:

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

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

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

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

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

Response 4c:

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

Response 4d:

A. Production Codes

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

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

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

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

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

B. Models

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

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

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

References:

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

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

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

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

Question 5:

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

Response:

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

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

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

Question 6:

Regarding Section 2.2.2.1 - Reactivity Coefficients and Defects:

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

Response:

Parts a and b:

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

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

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

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

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

Part c:

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

Question 7:

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

Response:

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

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

This practice is analogous to that used for previous model updates prior to the issuance of NEI 96-07. For example, application of the 50.59 process to the PDQ model changes (and later the NOMAD and TIP/CECOR changes) was focused on the key issues of whether the change created an unreviewed safety question (USQ), maintaining the "margin of safety," and whether the change involved a change to a Technical Specification. The SER for prior model approvals were reviewed to ascertain the NRC basis for previous approval. In particular, the PDQ Two Zone model was found to be an equivalent replacement of the previous models used for the same purposes inside the existing reload methodology framework and hence the change was determined not to be a USQ. The validation process was at least as broad as for the earlier models, with

far more available data. Although the data supported reductions in some uncertainty factors, the existing uncertainty factors were maintained (no reduction in margin of safety). The process used is functionally equivalent to changing elements of the method under the current 50.59 process. This was an internal review process using the same criteria as the original review as described in associated NRC SERs and using appropriate screening techniques under 50.59. Finally, since PDQ was not directly referenced in the COLR, implementation of the model upgrades did not require a change to the Technical Specifications. As discussed in the response to Question 4b, PDQ is not listed among the analytical methods supporting the COLR in Technical Specifications since it is not used to determine values for core operating limits.

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

Question 8:

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

Response:

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

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

- For some reload cycles where small departures (a few percent) from an accident specific limit occur, these studies can be used to show that margin in another key parameter that influences the same accident offsets the departure. For example, the

end of cycle (EOC) least negative moderator temperature coefficient is a key safety parameter for the rod ejection accident, although its influence is relatively weak. For one recent cycle, a small departure from the limit for this parameter was shown to be offset by large margins in the calculated ejected rod worth, which strongly influences the accident analysis results. These sensitivities are documented in VEP-NFE-2-A.

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

Question 9:

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

Response:

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

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

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

Question 10:

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

Response:

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

Reference:

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

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

Question 11:

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

Response:

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

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

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

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

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

different way to the dropped rod event due to the use of Framatome ANP fuel.

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

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

Reference:

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

Question 12:

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

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

Please provide the technical basis for these changes.

Response:

Part a:

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

Part b:

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

APPENDIX B

RAI Questions and Responses Set # 2

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

Total Pages : 8

Attachment

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

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

Background

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

NRC Request for Additional Information:

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

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

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

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

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

Dominion Response:

PDQ and NOMAD Codes & Models

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

behavior of the core relative to the model predictions.

TIP/CECOR Code & Model

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

RETRAN Code & Model

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

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

Dominion's RETRAN models and capability were approved in Reference 6. As noted in the SER, the Virginia Electric and Power Company (Dominion) Topical Report was

supplemented in three subsequent submittals (References 7, 8, 9) prepared in response to NRC Requests for Additional Information.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- Dominion can implement or substitute, under 10 CFR 50.59(c)(2)(viii), NRC approved codes and methodologies for use in establishing core operating limits via

the processes outlined in Generic Letter 83-11 Supplement 1, without additional NRC review and approval of these methods.

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

References:

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

2. Letter from C. O. Thomas (NRC) to T. W. Schnatz (UGRA), Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

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

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

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

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

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

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

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

APPENDIX C

RAI Questions and Responses Set # 3

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

Total Pages : 56

Attachment 1

Responses to NRC
Questions on RETRAN

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

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

NRC RETRAN QUESTION 1

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

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

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

DOMINION RESPONSE TO QUESTION 1a

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

I. RETRAN 02/MOD002 Restrictions

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

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

Dominion Evaluation

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

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

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

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

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

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

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

Dominion Evaluation

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

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

Dominion Evaluation

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

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

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

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.

Technical Evaluation Report (TER) "Items Requiring Further Justification"

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

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

SUMMARY

MAP NO: N1-16-01	DATE: 10/10/01	POWER: 29%
CONTROL ROD POSITIONS:	F-Q(Z) = 2.108	CORE TILT:
D BANK AT 150 STEPS	F-DH(N) = 1.546	NW 1.0037 | NE 1.0031
	F(Z) = 1.283	SW 0.9922 | SE 1.0010
	BURNUP = 5.0 MWD/MTU	A.O. = -6.233

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

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

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

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

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

SUMMARY

POWER: 100%

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

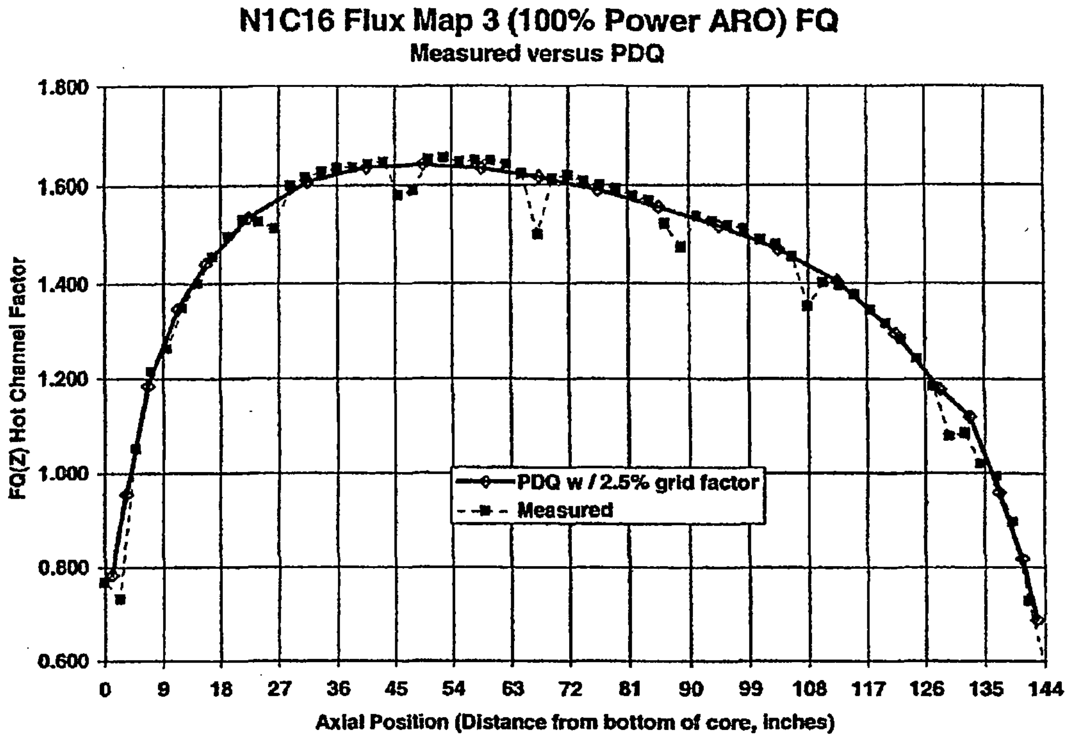
D BANK AT 226 STEPS F-DH(N) = 1.405

F(Z) = 1.139

BURNUP = 436.4 MWD/MTU A.O. = -2.537

CORE TILT:

NW 0.9982	NE 1.0035
SW 0.9954	SE 1.0029



NRC PDQ QUESTION 2

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

DOMINION RESPONSE TO QUESTION 2

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

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

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

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

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

NRC PDQ QUESTION 3

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

DOMINION RESPONSE TO QUESTION 3

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

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

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

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

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

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

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

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

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

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

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

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

axially) by specifying a volume weighted value for the crod nuclide. For insertion into the top half of a node, the crod nuclide number density is set to 0.5 in that node. Because the crod number density and cross sections are non-physical for a microscopic model, the crod nuclide is specified as non-depleting.

NRC PDQ QUESTION 4

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

DOMINION RESPONSE TO QUESTION 4

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

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

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

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

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

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

The Kolmogorov-Smirnov test (the D test) was used to assess the assumption of normality for the percent difference data. The assumption of normality was found to be acceptable for the

pooled data for each of the three cycles based on the results of the D test. A one-sided upper tolerance limit was defined as:

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

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

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

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

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

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

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

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

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

NUF Derivation Methods

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

NUF Derivation Methods (Continued)

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

Additional Information for Statistically Derived NUF Data

Parameter	Number of observations	Mean	Standard Deviation	Normality assumed?	Standard Deviation Multiplier (K)	N^{th} value (n)
Control Rod Worth – Integral worth, individual banks (raw data)	157	1.0%	4.5%	Yes	1.88	N/A
Critical Boron Concentration	54	6.3 ppm	20.0 ppm	Yes	2.05	N/A
Differential Boron Worth (raw data)	30	-0.3%	4.4%	No	N/A	N/A
Isothermal Temperature Coefficient	57	-0.8 pcm/°F	0.96 pcm/°F	No	N/A	1
$F\Delta H$ (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

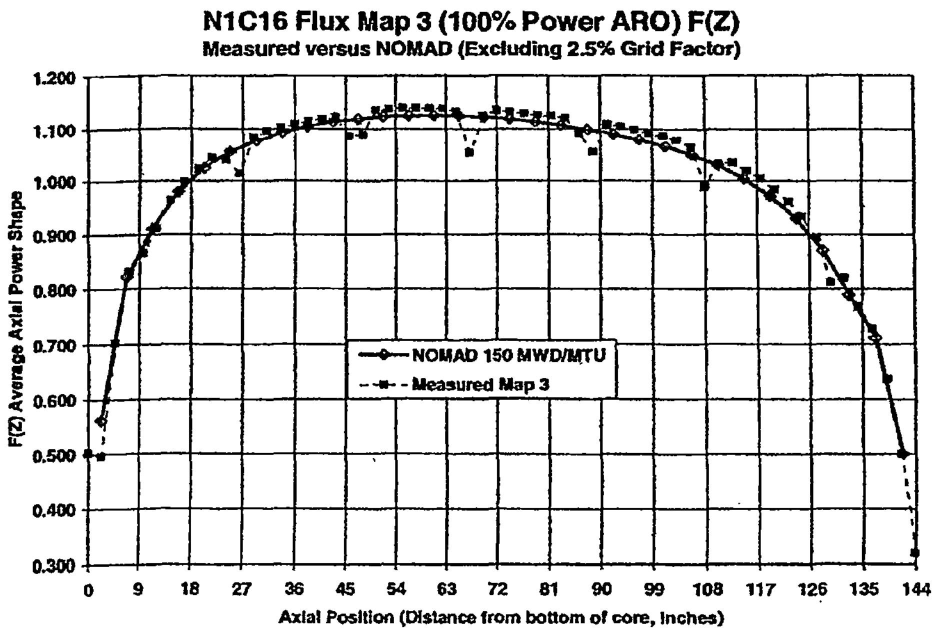
By letter dated December 2, 2002, VEPCO stated that the accuracy of the NOMAD model is verified each cycle during startup physics testing and during routine core follow. Please provide representative results from a recent refueling outage (comparisons between the startup physics test data and the NOMAD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

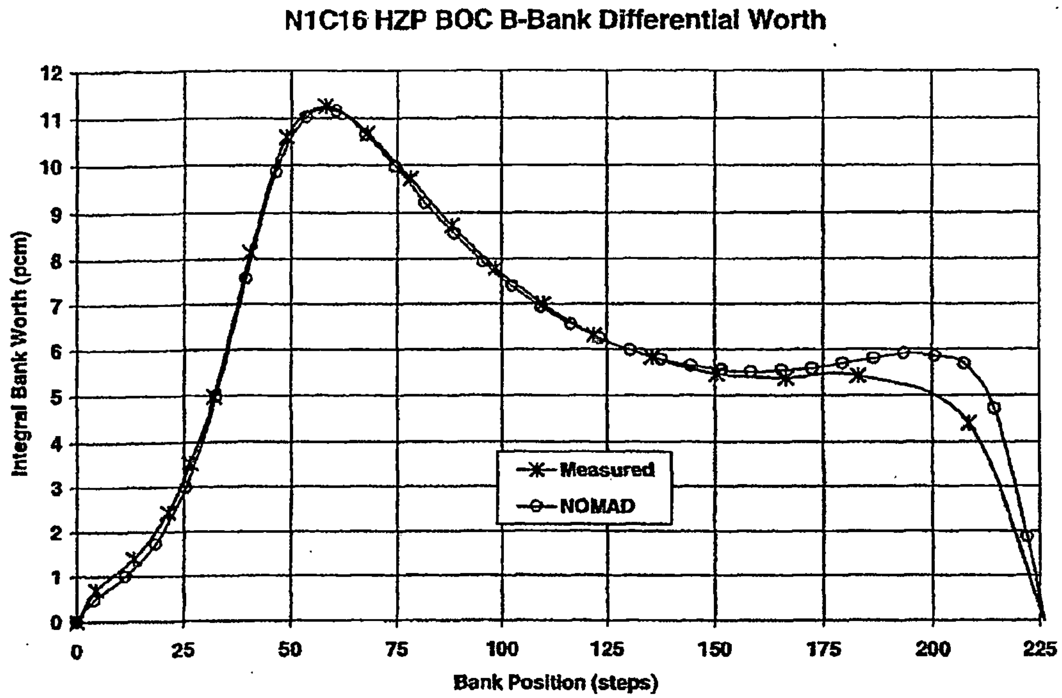
DOMINION RESPONSE TO QUESTION 1

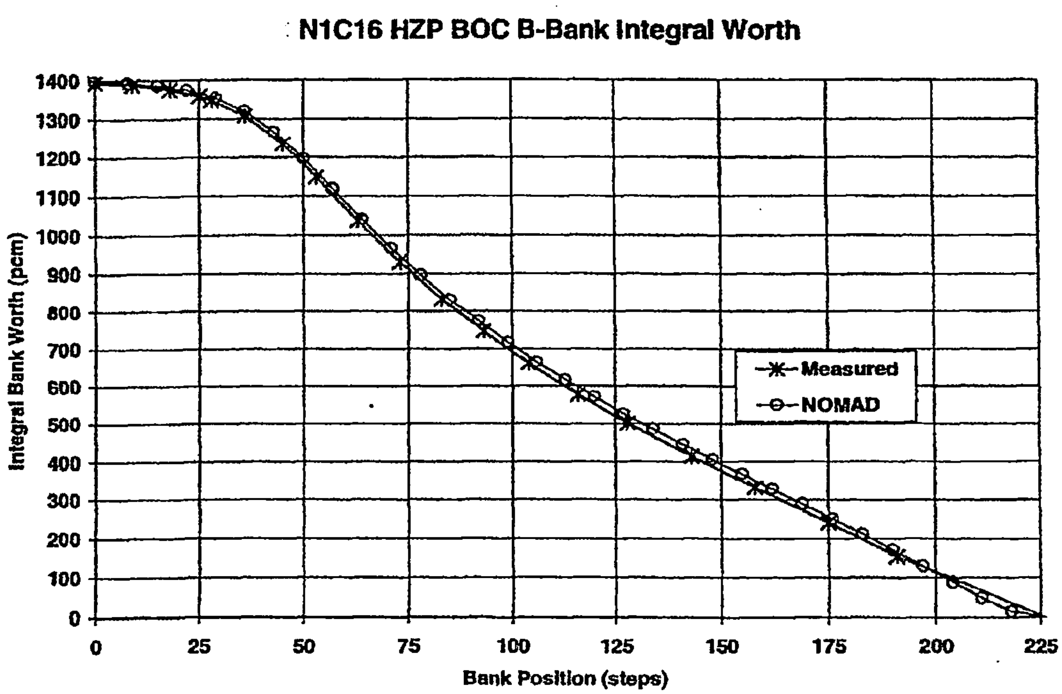
Verification of NOMAD accuracy comes primarily by extension through comparison to PDQ Two Zone model (Topical Report VEP-NAF-1) predictions during the NOMAD model setup process (see also the response to questions 3 and 7). The NOMAD model setup procedure provides specific power distribution and reactivity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

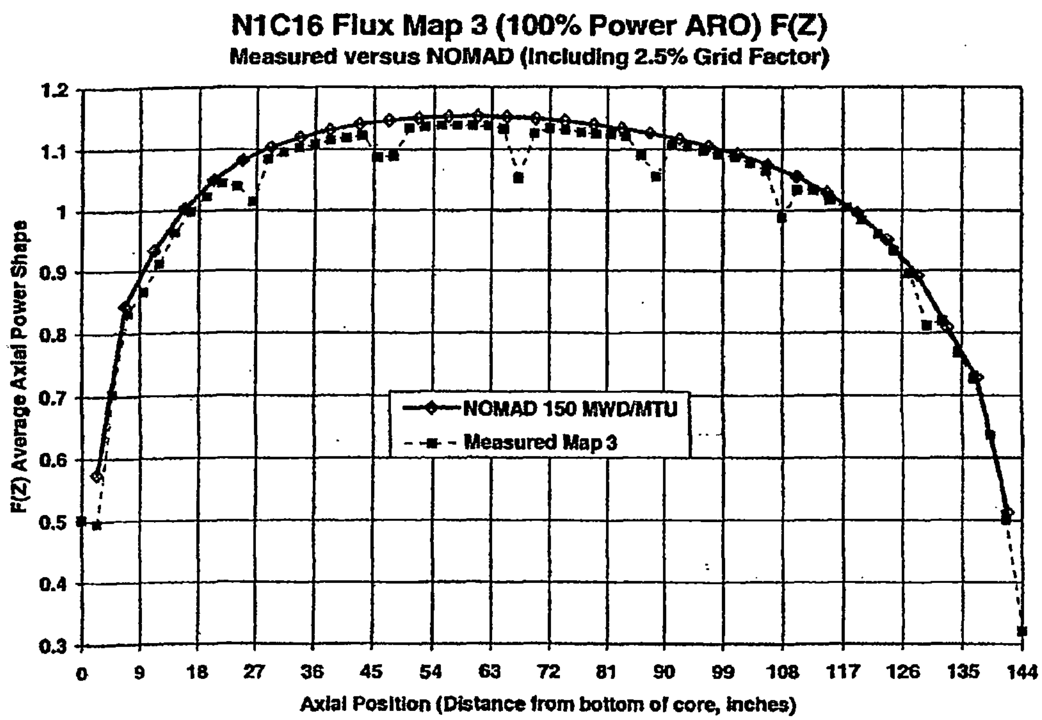
The final element of the control rod model is the ability to normalize bank worth to the PDQ Two Zone value. Although NOMAD was designed to produce acceptable control rod worth results without normalizing to PDQ, normalization is performed routinely for many design calculations to eliminate any difference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

APPENDIX D

VEP-FRD-42, Revision 1 RAI Questions and Responses Set

NRC Letter Serial No. 86-152 dated March 4, 1986
Dominion Letter Serial No. 86-152 dated May 2, 1986

Total Pages : 40

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

NOTED MAR 1 2 198 R.T.R.

March 4, 1986

Serial # 86 - 152

Rec'd. MAR 1 2 1986

Nuclear Operations
Licensing Supervisor

Mr. W. L. Stewart, Vice President
Nuclear Operations
Virginia Electric and Power Company
Richmond, Virginia 23261

Dear Mr. Stewart:

SUBJECT: REQUEST NUMBER ONE FOR ADDITIONAL INFORMATION ON VEP-FRD-42, REVISION 1

We are currently reviewing the Virginia Electric and Power Company Licensing Topical Report, VEP-FRD-42, Revision 1, entitled "Reload Nuclear Design Methodology".

The initial review reveals the need for the additional information indicated in the enclosure. In order to complete this review within the currently scheduled time, responses to all questions should be received by NRC by March 21, 1986. Please advise Richard Emch at (301) 492-7750 if you cannot meet this date.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Standardization & Special
Projects Directorate
Division of PWR Licensing-B

Enclosure:
As stated

Request for Additional Information on Virginia Power
"Reload Nuclear Design Methodology (VEP-FRD-42),"
August, 1985

S/D 1. Please indicate and discuss the main areas of difference, if any, between VEP-FRD-42 and the Westinghouse Reload Safety Evaluation Methodology (WCAP-9272). Is the Virginia Power methodology intended to apply as well to reactors and fuel other than those manufactured by Westinghouse?

S/D 2. Are the codes indicated in Section 2.1 the only codes used by Virginia Power in their reload safety analyses? Have all codes that are used in the reload safety analyses been reviewed and approved by the NRC?

S/D 3. Have all the initial current limits (as defined in Section 3.3.2) been verified using the Virginia Power methodology? If not, justify the validity of comparing values of key analysis parameters determined by using two distinct methodologies.

S/D 4. How are the results of the safety evaluations of reload safety analysis (as described in Section 3.4) incorporated in the limiting conditions of operation, limiting safety system setpoints, and technical specifications for a reload cycle? For a typical reload which are the limiting conditions of operation, limiting safety system setpoints, and technical specifications that are expected to change?

S 5. Please provide a set of key analysis parameters of each accident included in Table 1. For each accident, and each key analysis parameter, indicate the direction of variation that would increase the severity of the accident. Specify the criteria that are used to determine whether a given variable is a key analysis parameter for a particular accident.

S/D 6. Discuss the full scope of the review of design basis information (Section 3.2.1). Indicate the steps that are taken (comparison to checklists, e.g.,) to ensure that the review is complete.

7. Provide the calculational uncertainty and bias in all key analysis parameters, and indicate how these uncertainties are conservatively accounted for in the reload analysis.

8. The concept of bounding analysis proposed in Section 3.1 assumes the existence of a reference cycle safety analysis (e.g. the FSAR) to which the safety analyses of later cycles is compared. Is the methodology of VEP-FRD-42 intended to produce such a reference safety analysis? If so, justify the methods Virginia Power intends using to perform neutronic, thermal-hydraulic, fuel performance, transient and accident analysis calculations for the reference safety analysis of a mixed core containing non-Westinghouse fuel.

9. The concept of bounding analysis assumes (1) a monotonic dependence of the accident consequence on each key analysis parameter (i.e., the limiting direction of the key analysis parameter does not change over the entire range of the key analysis parameter) and (2) the effects of the different key analysis parameters on the accident consequence are not coupled (i.e., the effect of simultaneous changes in key analysis parameters may be determined by varying these parameters separately). Since these assumptions are not valid in general, provide the range of validity for each key analysis parameter over which bounding analysis is applicable.

10. Describe in detail the simple quantitative evaluation that may be made instead of an actual reanalysis (Section 4.0). If the simple quantitative evaluation makes use of available sensitivities to key analysis parameters, indicate how they are obtained and whether these sensitivities are generic or plant specific. Specify quantitative criteria that need to be satisfied if a simple quantitative evaluation is to be made instead of a complete reanalysis of the accident.

11. Describe the Virginia Power methodology for determining fuel performance key analysis parameters such as densification power spike, axial fuel rod shrinkage, fuel rod internal gas pressure, fuel stored energy and decay heat.

12. How is a fuel rod census curve determined for the single RCCA withdrawal accident (Section 3.3.4.2)? What is the conservative shape assumed for the fuel rod census curve?

13. Are the delayed neutron fraction, prompt neutron lifetime and Doppler weighting function included in the list of key analysis parameters for the rod ejection accident (3.3.4.3)? If not, justify their exclusion.

14. Since the use of a maximum prompt neutron lifetime is not conservative for the analysis of the rod ejection accident, justify the use of a maximum value of the prompt neutron lifetime as they key analysis parameter for transients (Section 3.3.3.5).

15. Discuss the relationship of a conservative trip reactivity shape (Section 3.3.3.3) to axial flux distributions realizable under CAOC and RPDC.

16. How does Virginia Power determine the six biases and constants, k_1 through k_6, the seven time constants, t_1 through t_7, and the trip reset function $f(\Delta I)$ associated with the overpower and overtemperature ΔT trip for a relaod cycle? Has Virginia Power established the adequacy of the presently used $f(\Delta I)$ function under RPDC?

17. Provide a discussion of how Virginia Power intends to account for the effects of fuel rod bowing, power spiking and fission gas release in their reload safety analysis.

18. How does Virginia Power intend re-analyzing the Loss Of Coolant Accident (LOCA) for a reload core?

19. How would a set of conservative initial conditions be determined when a reanalysis of a specific accident is necessary?

May 2, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. C. E. Rossi, Assistant Director
Division of PWR Licensing-A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 86-152
NO/EJL:vlh
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION
NORTH ANNA POWER STATION
RELOAD DESIGN METHODOLOGY-ADDITIONAL INFORMATION

On September 19, 1985 we submitted Revision 1 of the topical report VEP-FRD-42, "Reload Nuclear Design Methodology", to you for review and approval. This report describes the current methodology and code models that we use to perform nuclear reload design and safety analyses. In a letter dated March 4, 1986 from Mr. Herbert N. Berkow, we received a request for additional information on VEP-FRD-42, Revision 1. On April 4, 1986 we met with members of your staff to review the information that was requested. Enclosed are our reponses to the information request.

Very truly yours,

W. L. Stewart

Enclosure

1. Responses to the Information Request on VEP-FRD-42, Revision 1.

cc: Dr. J. Nelson Grace
Regional Administrator
NRC Region II

Mr. Lester S. Rubenstein, Director
PWR Project Directorate No. 2
Division of PWR Licensing-A

Mr. Leon B. Engle
NRC North Anna Project Manager
PWR Project Directorate No. 2
Division of PWR Licensing-A

Mr. Chandu P. Patel
NRC Surry Project Manager
PWR Project Directorate No. 2
Division of PWR Licensing-A

Mr. Larry King
NRC Resident Inspector
North Anna Power Station

NRC Senior Resident Inspector
Surry Power Station

Mr. Marv Dunenfeld
Senior Nuclear Engineer
Reactor Systems Branch
Division of PWR Licensing-A

WL 5/21/86

bc: Mr. R. H. Leasburg - 21/OJRP
Mr. J. A. Ahladas - 4/EB
Mr. ~~R. W. Calder~~ - 3/EB LWC 4-13-86
~~Mr. R. J. Hardwick, Jr. - 5/OJRP~~ RJH 4/28/86
~~Mr. J. L. Wilson - 5/OJRP~~ JLW 4/25/86
~~Ms. N. E. Clark - 5/OJRP~~ NEC
Mr. E. W. Harrell - NAPS EWH per telecon with JHL 4/18/86.
Mr. R. F. Saunders - SPS
Mr. E. R. Smith, Jr. - NAPS ERS per telecon with JHL 4/18/86.
Mr. H. L. Miller - SPS HLM per telecon with JBL 4/15/86.
~~Mr. J. W. Ogren - 5/OJRP~~ JO 4/25/86
Mr. G. L. Pannell - 5/OJRP for SLT 4/25/86
Mr. D. J. VandeWalle - 5/OJRP DJV 4/18/86
Mr. D. A. Sommers - 5/OJRP DAS 4/18/86
Dr. E. J. Lozito - 5/OJRP EJL 4/15/86
Mr. C. T. Snow - 5/OJRP C
Mr. A. F. Yaros - 8/OJRP
Mr. R. M. Berryman - 2/EB RMB 4/11/86
Mr. K. L. Basehore - 2/EB KLB 4/11/86
Mr. D. Dziadosz - 2/EB DD 4/11/86
Mr. R. T. Robins - 2/EB RTR 4-11-86
Dr. B. L. Shriver - RP5
Mr. R. W. Cross - 5/FB
Mr. I. B. Choate - 5/FB
NOD Tech Library (bc original)
GOV 02-54B
NE File 2.5.8 - 2/EB
E&C Records Management NP-409.1 - 4A/EB

Enclosure

Responses to the Information Request

On VEP-FRD-42, Revision 1

RESPONSE TO INFORMATION REQUEST ON VEP-FRD-42 REVISION 1

Question 1: Please indicate and discuss the main areas of difference, if any, between VEP-FRD-42 and the Westinghouse Reload Safety Evaluation Methodology (WCAP-9272). Is the Virginia Power methodology intended to apply as well to reactors and fuel other than those manufactured by Westinghouse?

Response: As stated in VEP-FRD-42, Rev. 1, the methods we use to determine and evaluate the key parameters for a reload safety analysis are consistent with those outlined in WCAP-9272, and Reference 1.

The main area of difference between WCAP-9272 and VEP-FRD-42 Rev. 1 is in the calculation and verification of axial flux difference bands for reactor operation. WCAP-9272 describes the process by which the constant axial offset control method is verified. Our analogous method for verifying a constant axial offset band is described in Reference 2. However as indicated in VEP-FRD-42 Rev. 1, we have developed a relaxed power distribution control (RPDC) mode of operation for our reactors as an alternative to axial offset control. The method of shape verification for RPDC is described in Reference 1.

The methodology described in VEP-FRD-42 Rev. 1 is currently being used for our nuclear units utilizing Westinghouse fuel. The methodology described in VEP-FRD-42 Rev. 1 would be applicable to other reactors of Westinghouse design. However, use of this methodology for non-Westinghouse reactors would have to be justified based on the specific reactor type. The basis for this justification would be submitted for NRC review prior to application. The methodology described in VEP-FRD-42 Rev. 1 would be

applicable to fuel from other vendors as long as it was determined that the design basis accidents and their associated key parameters remained the same. In the event that we utilize fuel from other vendors, the appropriate documentation will be submitted for NRC review.

Question 2: Are the codes indicated in Section 2.1 the only codes used by Virginia Power in their reload safety analyses? Have all codes used in the reload safety analyses been reviewed and approved by the NRC?

Response: In general, all of our reload analyses are performed with the codes listed in Section 2.1 of VEP-FRD-42 Rev. 1, which have been reviewed and approved by the NRC. The exceptions to this are as follows:

a) Data handling routines, designed to summarize and/or transfer data between approved codes are not submitted for specific NRC approval. They are, however, subject to the same general verification and configuration control requirements as the approved analysis codes.

b) Current evaluations of the dropped rod transient for North Anna (a negative rate trip protected plant) are based on the Westinghouse interim methodology discussed in Reference 3 and as such make use of the Westinghouse LOFTRAN code. LOFTRAN is an NRC-approved code and is routinely used for transient safety analysis of most Westinghouse units. We have a new dropped rod methodology under development which will use the Section 2.1 codes. Qualification documentation for this methodology will be submitted for NRC review.

c) Our Loss of Coolant Accident (LOCA) capability is also based on NRC-approved Westinghouse codes. For further discussion, see the response to Question 18.

d) For those transients for which we have received NRC approval to apply the Westinghouse statistical DNB methodology (Improved Thermal Design Procedure-ITDP), the Westinghouse THINC code, an NRC-approved core thermal-hydraulics code, is being used for DNB calculations. We have submitted, in Reference 4, a topical report describing an in-house statistical DNBR evaluation methodology. Upon approval of this methodology, we plan to use our approved COBRA model for these transients.

Question 3: Have all the initial current limits (as defined in Section 3.3.2) been verified using the Virginia Power methodology? If not, justify the validity of comparing values of key analysis parameters determined by using two distinct methodologies.

Response: As indicated in Section 3.3.2 of VEP-FRD-42 Rev. 1, the significance of the initial current limits lies not in the codes which generated the physics parameter values, but in the fact that they were assumed in the transient analysis. Thus for the transient analysis to remain bounding for a given reload cycle, it must be demonstrated that parameter values for that reload fall within the limits established by the safety analysis. The list of key safety parameters is a "living document", i. e., whenever a key parameter is violated and a transient reanalysis is required, the list is updated to reflect the new key safety parameter values assumed in the analysis.

The governing consideration for each reload, therefore, is to ensure that the key safety parameter values predicted by the physics analysis are 1) less limiting than the current limits (otherwise a transient reanalysis may be required) and 2) bound the range of values expected for the cycle. The former constraint is met simply by comparing the cycle specific results to the limits, which in turn are set by the transient analysis assumptions, and not by physics calculations. The second constraint is met by qualifying the physics models against data as described in References 2,5,6, and 7 and by applying the appropriate design calculational uncertainty factors as discussed in the response to Question 7. Thus direct comparison of cycle-specific core physics results to previous results, whether generated by the same or a different methodology, is not required. For further discussion on comparing analyses generated with different transient models, see the response to Question 5.

Question 4: How are the results of the safety evaluations of reload safety analysis (as described in Section 3.4) incorporated in the limiting conditions of operation, limiting safety system setpoints, and technical specifications for a reload cycle? For a typical reload which are the limiting conditions of operation, limiting safety system setpoints, and technical specifications that are expected to change?

Response: During the design and safety analysis initialization process for each reload cycle, the Technical Specifications are reviewed to determine whether changes to the limiting conditions of operation and limiting safety system setpoints have occurred since the previous reload which could impact the results of the safety analysis of the reload core. Similarly, if changes to the limiting conditions for operation or limiting safety system setpoints are found to be required for operation of the

reload core, the required changes and the associated technical justification are incorporated into the Reload Safety Evaluation (RSE), which is reviewed and approved by the our Safety Evaluation and Control staff and the Station Nuclear Safety and Operating Committee. The RSE and associated Technical Specifications changes are then submitted to the NRC for approval prior to startup.

The reload design and analysis philosophy for our units has been such that changes to the limiting safety system settings have not been requested to support operation of specific reload cores. Those changes which have been requested (e.g., changes to the core thermal limits and associated overtemperature/overpower delta-T setpoints following steam generator replacement, change of the part-power multiplier on the local enthalpy rise hot channel factor (F-delta-H) limit equation) have been outside the reload licensing process.

Historically, the sections of the Technical Specifications for our nuclear units which have been most likely to change from cycle-to-cycle have been:

a. The control rod bank insertion limits. These limits influence the shutdown reactivity margin, ejected rod worths and peaking factors and normal operation radial peaking. In the past, the normal operation peaking at lower powers has typically driven requests for revisions to the insertion limits. Subsequent to these requests, we obtained NRC approval for increasing the normal operation power peaking factor (F-delta-H) limits at part power. As a result reload-related requests to change the rod insertion limits are not anticipated in the future.

b. The axially-dependent radial peaking factor (Fxy) limits. For North Anna, current reloads involve reporting of cycle-specific Fxy limits as part of a core surveillance report. Once the Relaxed Power Distribution Control (RPDC) methodology is implemented, the core surveillance report will include cycle-specific axial flux difference (AFD) limits and cycle-specific values for the nonequilibrium power peaking factor discussed in detail in Reference 1.

c. The minimum power for which frequent local power peaking (FQ) surveillance is required (the FQ surveillance threshold power level). In some previous Surry cycles these changes occurred when the reload calculations performed under the ECCS Final Acceptance Criteria showed transient local power peaking factors (FQ) in excess of the limit established by the LOCA/ECCS analyses. This situation was aggravated by the relatively low FQ limits which were established for the units when high degrees of steam generator tube plugging (25% or more) were present. Following replacement of the steam generators, LOCA reanalyses resulted in an FQ limit which is high enough that frequent FQ surveillance is not anticipated for future reloads.

Question 5: Please provide a set of key analysis parameters of each accident included in Table 1. For each accident, and each key analysis parameter, indicate the direction of the variation that would increase the severity of the accident. Specify the criteria that are used to determine whether a given variable is a key analysis parameter for a given accident.

Response: For a complete correlation of key safety parameters with the various accidents as well as limiting directions, the reviewer is referred to WCAP-9272, which has been submitted to the NRC for review.

In order to ensure that the transient analysis results conservatively bound the expected response, the various accidents must be correlated with those parameters which have a significant impact on them and the limiting direction of variation for those parameters. This is done in an integrated manner for each accident, drawing primarily upon physical reasoning and past analytical experience.

Physical reasoning, although not adequate in and of itself to establish the limiting direction for all parameters, provides an important starting point for assessing the impact of various parameters. For example, for cooldown accidents the limiting DNBR is expected to occur for the most negative moderator temperature coefficient, since this maximizes core power. Accidents which reach their most severe condition very rapidly, such as rod ejection, are expected to be much less sensitive to the moderator coefficient, etc.

Sources of past analytical experience for the various accidents are also drawn from. For many transients, the identification and limiting direction of key safety parameters are presented in the units' UFSARs. A more formal

presentation of key safety parameters for the various accidents along with indication of limiting directions is given in WCAP-9272. We have confirmed the applicability of these key parameters and limiting directions for a broad selection of the UFSAR transients using the models and methods referred to in Section 2.0 of VEP-FRD-42 Rev. 1. The fact that this information is valid for both the Vendor's methodology and our methodology is not coincidence, but reflective of the fact that the basic physical characteristics of the accidents being modeled remain fixed.

This process of confirming the applicability of the WCAP-9272 conclusions to our models and methods involved the execution of numerous sensitivity studies using the RETRAN computer code. As a result, a considerable base of experience has been compiled over several years. Some of the results of these studies have been submitted to the NRC. Reference 8, for example, presented the results of sensitivity studies for the rod ejection event using our methodology. These studies covered variations in eight different neutronics parameters as well as various thermal hydraulics and power-distribution related factors.

As part of the NRC review of our RETRAN transient analysis topical report (Reference 9), we submitted sensitivity studies (Reference 10) which covered a spectrum of parameters for transients involving reactivity addition, changes in primary to secondary heat transfer (both increase and decrease) and decrease in RCS flow rate. In summarizing those studies, it was concluded that our RETRAN models showed the same general sensitivities as discussed in the Surry and North Anna UFSAR's.

In addition to these externally published sensitivity studies, we have performed and documented internally numerous other studies which have

confirmed the limiting directions of key parameters for those accidents which have been shown historically to be most subject to reanalysis as a result of core reloads.

Question 6: Discuss the scope of the review of design basis information (Section 3.2.1). Indicate the steps that are taken (comparison to checklists, e.g.,) to ensure that the review is complete.

Response: Prior to initiation of the reload design safety analysis, we perform a review of the pertinent design basis information applicable to the reload. This review is governed by our internal procedures, which provide for the review of the following:

1. Procedures controlling the performance of, review of, approval of, and documentation of all production calculations.
2. The unit's previous cycle documentation including;
   a. Safety Analysis,
   b. Nuclear Design Analysis,
   c. Operating History,
   d. Design or Technical Specification changes made since startup.
3. Applicable correspondence with vendor and NRC.
4. Applicable sections of the UFSAR.
5. Current Technical Specifications.
6. Technical Specifications changes expected to be made before or during the cycle.
7. Current Safety Related NSSS Parameter Checklist.
8. Current Key Safety Parameters Checklist.
9. Current Fuel Assembly and Insert Component Restricted List.
10. Current Fuel Management Scheme.

To ensure that all applicable information is current and being properly utilized a series of meetings between the design and safety engineers is held. A design initialization checklist provides the schedule for these meetings and indicates the information reviewed, a summary of any design requirements or restrictions unique to the cycle, and design assumptions to be used for the cycle resulting from operational or licensing

uncertainties. In addition, this checklist provides the necessary operating and energy requirements for the cycle. The checklist is reviewed and approved by management.

The NSSS Parameter Checklist contains the current NSSS parameters (fluid volumes, system flow rates, safety and relief valve capacities, etc.) necessary for the accident analyses. This checklist is reviewed prior to use by the safety analysis group, power station personnel, the nuclear operations support group, and other appropriate personnel to ensure the accident analysis input assumptions are consistent with the current system parameters. Note that this is a double check since all approved design changes which potentially impact the safety analyses will have been previously reviewed by Nuclear Engineering in accordance with the appropriate procedures for the design change approval process.

During the reload safety analysis phase the design engineer calculates the parameters identified by the Key Safety Parameters Checklist. This checklist contains the limiting nuclear and thermal hydraulic parameters and the values used in the current safety analyses. If a reload parameter is not bounded by the current limit the safety engineer is notified. At the end of the analysis phase for the design engineer, the calculations are documented and the results are formally summarized by memorandum to the safety engineer. This memorandum takes the form of a comparison between the calculated reload values and the limits specified by the Key Safety Parameters Checklist. Prior to the finalization of the design engineer's calculation of the key parameters, the design engineer and safety engineer will have formally reviewed the Key Safety Parameters Checklist at least twice: once prior to the initiation of the calculations to review the limits and the second time at the conclusion of the design

calculations to ensure the reload values are bounded by the current limits and/or to identify areas where accident reanalysis or reevaluation is required.

Question 7: Provide the calculational uncertainty and bias in all key analysis parameters, and indicate how these uncertainties are conservatively accounted for in the reload analysis.

Response: As indicated in Table 2 of VEP-FRD-42 Rev. 1 the key analysis parameters to be determined for each reload cycle may be divided into three categories. These are non-specific, specific, and fuel performance and thermal hydraulic related parameters.

a) Non-specific Key Parameters

The non-specific parameters are generated to allow evaluation of the general core characteristics of the reload cycle. These parameters are calculated for conditions bounding those expected to occur during the reload cycle to ensure the limiting values of the parameter are determined. These conditions include conservative assumptions for such parameters as xenon distributions, power level, control rod position, operating history, and burnup. Table 1 provides a list of the non-specific key parameters generated in the reload safety analysis along with the associated conservatisms used in calculating the parameters.

b) Specific Key Parameters

Specific key parameters are generated by statically simulating an accident with the physics models. These parameters are (or are directly related to) rod worth, reactivity insertion rate, or peaking factors. The static conditions for the accident are the most conservative conditions for the

accident and include conservative assumptions for variations in such parameters as power level, control rod position, xenon distribution, previous cycle burnup, and current cycle burnup. Table 2 provides a list of the specific key parameters generated in the reload safety analysis and the conservatisms applied in calculating the parameters.

c) Fuel Performance and Thermal/Hydraulic Parameters

The core thermal limits are generated assuming the design uncertainties discussed in Reference 11. In that report, detailed discussions are presented on hot channel factors, including radial and local power factors, channel geometry, local flow starvation, etc. DNB correlation uncertainties are also discussed.

Fuel performance related information, including initial fuel temperatures (both core average and hot spot) the effects of fuel densification on core average power and on local power, fuel rod internal gas pressure, fuel stored energy and decay heat are currently generated by the fuel vendor and transmitted to us for each reload core for use in performing the reload evaluations. The vendor calculations are performed using the latest approved version of the Westinghouse PAD code, documented in References 12, 13, and 14. All of these documents have been reviewed and approved by the NRC.

Question 8: The concept of bounding analysis proposed in Section 3.1 assumes the existence of a reference cycle safety analysis (e.g. the FSAR) to which the safety analysis of later cycles is compared. Is the methodology of VEP-FRD-42 intended to produce such a reference safety analysis? If so, justify the methods Virginia Power intends using to perform neutronic, thermal-hydraulic, fuel performance, transient and accident analysis calculations for the reference safety analysis of a mixed core containing non-Westinghouse fuel.

Response: The methodology presented in VEP-FRD-42, Rev. 1 and its referenced documents is our integrated methodology for ensuring that existing or "reference" safety analyses bound the conditions realised for reload cores and for producing new reference analyses as required. It should be pointed out that in the development and maintenance of a list of key safety parameters for a given plant, the reference safety analyses performed for various UFSAR chapter 15 accidents may have been originally performed at different points in the history of the unit as various cycle designs have required the reanalysis of different accidents. The key safety parameters list is therefore a "living document" which reflects the most recent analyses of the various events.

In general, we expect the methods presented in VEP-FRD-42 Rev. 1 to be valid for both Westinghouse/Non-Westinghouse fuel mixes as well as cores designed by other vendors for use in Westinghouse designed plants. The codes that we use are currently being used by other design organizations to analyze a variety of fuel and NSSS types. However, we do not have models at the present time which reflect non-Westinghouse fuel. In the event that we utilize fuel from other vendors, the appropriate documentation will be submitted for NRC review.

Question 9: The concept of bounding analysis assumes (1) a monotonic dependence of the accident consequence on each key analysis parameter (i.e., the limiting direction of the key analysis parameter does not change over the entire range of the key analysis parameter and (2) the effects of the different key analysis parameters on the accident consequence are not coupled (i.e., the effect of simultaneous changes in key analysis parameters may be determined by varying those parameters separately). Since these assumptions are not valid in general, provide the range of validity for each key analysis parameter over which bounding analysis is applicable.

Response: The assumption of monotonic dependence of accident consequences on each key analysis parameter (assumption (1) above) can be established for most parameters for most events by the application of physical reasoning. This intuitive approach is backed up by numerous sensitivity studies performed by us and our fuel vendor which confirm the trends for the various key parameters. Expressed in terms of a response surface, this combination of physical reasoning plus sensitivity studies is used to demonstrate with a high level of confidence that for every dependent/independent variable plane, there are no inflection points in the response curve. For those combinations of parameters and events where inflection points have been identified (non-monotonic dependence), a spectrum of values of the parameter in question is assumed in performing the analysis. A case in point is the analysis of the uncontrolled control rod bank withdrawal at power, where a range of reactivity insertion rates, corresponding to varying rod speeds and differential rod worths, is analyzed.

The second condition (no coupling between the key analysis parameters) is not required for the bounding analysis concept to be valid as long as the first condition (monotonic dependence in every plane) is rigorously satisfied. In other words, even if the effect of simultaneous changes in key parameters cannot be determined by varying the parameters separately, as long as there are no inflection points in the surface and all parameters are within their respective limits, then the analysis results are bounded by the reference analysis. The only thing that cannot be determined is the exact degree of margin between the cycle-specific results and the reference analysis results.

The range of validity for each key parameter is therefore established by the parameter limits themselves. As accidents are reanalyzed to envelop new reload cycles, the limits and therefore the ranges of validity are extended.

Question 10: Describe in detail the simple quantitative evaluation that may be made instead of an actual reanalysis (Section 4.0). If the simple quantitative evaluation makes use of available sensitivities to key analysis parameters, indicate how they are obtained and whether these sensitivities are generic or plant specific. Specify quantitative criteria that need to be satisfied if a simple quantitative evaluation is to be made instead of a complete reanalysis of the accident.

Response: Quantitative evaluation of a small violation of parameter limits may be made in one of several ways. First, if the interplay between the various key safety parameters in determining accident response is well defined, margin in one parameter may be used to offset a small violation

in another parameter. This process is best defined by presenting some examples:

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

2) For small violations in the normalized trip reactivity shape curve, the safety analyst may use the minimum total trip reactivity calculated for the cycle to show that at the points on the curve where the violations occur, the actual integral trip reactivity is greater than that assumed in the safety analysis.

A second method of quantitative evaluation involves using tradeoffs of known sensitivities which have been generated either specifically for our plants using our methods or, in one case, using generic three-loop results generated by Westinghouse using approved methods. The specific case involves generic sensitivities published in Reference 15 which show the trade-off between allowable post-ejection peaking factor (FQ) and ejected rod worths. For some reload cycles where small violations (a few percent) of the FQ limits occur, these studies can be used to show that when there is appreciable margin to the ejected rod worth limit, the reference safety analysis remains bounding. The validity of performing this type of

evaluation is based on the use of available sensitivities which have been generated with the same transient methodology as used in performing the reference analysis.

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

Question 11: Describe the Virginia Power methodology for determining fuel performance key analysis parameters such as densification power spike, axial fuel rod shrinkage, fuel rod internal gas pressure, fuel stored energy and decay heat.

Response: Fuel performance key analysis parameters for our reloads are determined by the fuel vendor using their fuel rod performance analysis methodology. Fuel region specific and fuel rod specific input for this analysis is based on information provided by our reload design models for each reload design. This specific input includes estimates of the achievable fuel region burnups for the reload cycle and the associated individual fuel rod power histories, fluxes and fluences. Results of the fuel rod performance analysis are evaluated by the vendor to insure that fuel performance design criteria will not be violated for the reload design. The results are also provided to us for comparison with fuel performance related key safety parameters.

Question 12: How is a fuel census curve determined for the single RCCA withdrawal accident (Section 3.3.4.2)? What is the conservative shape assumed for the fuel rod census curve?

Response: The single RCCA withdrawal accident is classified as an ANS Condition III event. For Condition III events a small amount of fuel failures are allowed. The fuel rod census is performed to determine whether the percentage of rods which may enter into DNB and exceed the thermal limits of the fuel is greater than the amount assumed in the safety analysis.

The fuel rod census for the single rod withdrawal accident is performed using a 1D-2D synthesis technique to determine the F-delta-H for each rod in the core. Full core 2-D pin by pin radial power distributions are calculated for both all rods out and D-bank in less a single RCCA. The 1-D model is then used to determine the axial power sharings for the rodded and unrodded planes associated with D-bank to the insertion limits less a single RCCA. Using the power sharings from the 1-D calculations, the pin by pin radial power distributions from the 2-D calculations are synthesized to obtain F-delta-H's for each pin which are then tabulated to determine the percentage of rods with an F-delta-H greater than the value assumed to produce DNB. This analysis is performed at various times in life to account for cycle burnup effects on the power distribution. In addition axial xenon variations which may occur are accounted for in the 1-D model used for generation of power sharings by artificially skewing the axial power distributions in a conservative manner.

Question 13: Are the delayed neutron fraction, prompt neutron lifetime and Doppler weighting function included in the list of key analysis parameters for the rod ejection accident (3.3.4.3)? If not, justify their exclusion.

Response: The delayed neutron fraction, post ejected rod peaking factor, ejected rod worth and Doppler defect are included in the list of key analysis parameters for the rod ejection accident. The prompt neutron lifetime is not included (see response to Question 14). The Doppler weighting function is not included because it is a generic parameter coupled to the post-ejection maximum power peaking factor. Additional discussion of our methodology for analyzing the rod ejection accident may be found in Reference 8.

Question 14: Since the use of a maximum prompt neutron lifetime is not conservative for the analysis of the rod ejection accident, justify the use of a maximum value of the prompt neutron lifetime as the key analysis parameter for transients (Section 3.3.3.5).

Response: Instead of the maximum value of the prompt neutron lifetime, a generic value is used as described in Reference 8 which describes our rod ejection accident methodology. Analyses described in Reference 8 showed a low sensitivity of the methodology to this parameter and were used to justify the use of a generic value.

Question 15: Discuss the relationship of a conservative trip reactivity shape (Section 3.3.3.3) to axial flux distributions realizable under CAOC and RPDC.

Response: A conservative trip reactivity shape is produced in the reload safety analysis by minimizing the initial worth of the tripped rods through the use of the most bottom peaked axial power distribution realizable under CAOC or RPDC operation. As indicated in VEP-FRD-42 Rev. 1, the trip reactivity shape is produced using the 1-D NOMAD code (Reference 2). At the most limiting cycle burnup the axial power distribution is artificially skewed to produce an axial flux difference consistent with the negative side of the operating band (CAOC or RPDC). The trip shape is then generated by determining the reactivity versus position for the trip banks using the above core conditions.

Question 16: How does Virginia Power determine the six biases and constants k1 through k6, the seven time constants, t1 through t7, and the trip reset function f(delta-I) associated with the overpower and overtemperature delta-T trip for a reload cycle? Has Virginia Power established the adequacy of the presently used f(delta-I) function under RPDC?

Response: The coefficients of the steady state overtemperature and overpower delta-T protection equations (k1,k2,k3,k4,k6) as well as the f(delta-I) function are developed in accordance with the methodology of Reference 16. The basis for the current protection equations is verified on a reload basis and would normally remain unchanged. Previous changes to the protection equations have been the result of a modification to the thermal design flowrate, core power uprate, or the implementation of a F-delta-H part power multiplier of 0.3.

The constants in the dynamic term of the overpower delta-T equation (k5 and t3) are generic Westinghouse values chosen based on studies documented in Reference 16. The time constants t1,t2 and t4-t7 are selected during the initial plant design phase and are governed by such considerations as compensation for thermal and transport delays in the RCS and in the temperature sensors, and by minimizing the frequency of spurious noise-related trips. These considerations will not change from reload to reload and the time constants are therefore expected to remain fixed. The time constants are reflected in our transient analysis models.

As noted in Section 3.4 of Reference 1, the adequacy of the F(delta-I) function is demonstrated on a cycle-specific basis for RPDC. Evaluations to date have shown that ample margin exists in the existing f(delta-I)

function to accomodate the wider range of axial power shapes inherent in RPDC.

Question 17: Provide a discussion of how Virginia Power intends to account for the effects of fuel rod bowing, power spiking and fission gas release in their reload safety analysis.

Response: The effects of fuel rod bow are quantified in Reference 17. Appropriate penalties derived from this analysis are fully compensated for in our thermal hydraulic analysis methodology by the generic retained DNBR margin for each type of fuel as documented in References 18 and 19.

Power spiking effects are accounted for by the application of a conservative power spiking factor to applicable safety analyses. This factor is calculated by the fuel vendor using their fuel rod performance methodology, (see the response to Question 11 above). The effects of fission gas release is likewise considered by the fuel vendor in their analysis of the fuel rod design for the reload cycle as described in the response to Question 11 above.

Question 18: How does Virginia Power intend re-analyzing the Loss of Coolant Accident (LOCA) for reload cores?

Response: The LOCA key safety parameters for each reload are compared to the current limiting values assumed in the most recent plant specific LOCA analysis to confirm that the limits remain bounding. As discussed previously in the response to Question 4, we attempt to perform any accident reanalysis on a schedule that would allow NRC review and approval well in advance of the Reload Safety Evaluation. For LOCA reanalysis, we

use the Westinghouse LOCA-ECCS evaluation models which have been developed and approved for use with 10CFR50 Appendix K applications. This was discussed with the NRC staff during a May 16, 1984 NRC audit (Reference 20). The NRC staff has previously reviewed and approved several LOCA analyses (References 21 and 22) performed by our staff with Westinghouse codes and methods.

Question 19: How would a set of conservative initial conditions be determined when a reanalysis of a specific accident is necessary?

Response: Most accidents exhibit some small sensitivity to the initial conditions assumed. For accident evaluation, the initial conditions are obtained by adding or subtracting, as appropriate, maximum steady-state errors to or from rated values. Steady-state errors which are applied are:

a)	Core power	+2 percent allowance for calorimetric error
b)	Average RCS temperature	+/- 4°F allowance for deadband and measurement error
c)	Pressurizer pressure	+/- 30 psi allowance for operational fluctuations and measurement error

In general, errors are chosen in the directions which minimize core margins or margins to other plant design criteria (e.g., overpressure) and are therefore dictated by the type of analysis being performed. Similar to the application of uncertainties to the key analysis parameters, the limiting directions for the application of errors to the initial conditions are determined based on a) physical reasoning and a conceptual understanding of the type of accident being analyzed; b) vendor experience and insight as documented in the accident analysis chapters of the FSARs and c)in some cases, sensitivity studies performed with our transient analysis models and methods.

For transients involving increase or decrease in secondary heat removal, the initial steam generator mass has some influence on transient results also. We use a 10% uncertainty about the best estimate initial steam generator mass.

The initial conditions for transient analyses are chosen and justified on a case by case basis. These assumptions are identified in the licensing submittal for each reanalysis.

A discussion of the statistical treatment of errors for certain proposed DNB applications is presented in Reference 4 which is currently awaiting NRC review and approval.

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3. NRC Meeting Report, "Summary of Nov. 19, 1979 Meeting with Westinghouse and Licensees Regarding Dropped Rod Protection," January 18, 1980.

4. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), "Statistical DNBR Evaluation Methodology", Serial No. 85-688, October 8, 1985.

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

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13. W. J. Leech, "Improved Analytical Models Used in Westinghouse Westinghouse Fuel Rod Design Computations Application for Transient Analysis", WCAP 8720 Addendum 1, (September 1979).

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16. S. L. Ellenberger, et al., "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions", WCAP 8745, (March, 1977).

17. J. Skarita, et al., "Fuel Rod Bow Evaluation", WCAP 8691 Rev. 1, (July, 1979).

18. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), "Amendment to Operating Licenses NPF-4 and NPF-7 North Anna Power Station Unit Nos. 1 and 2 Proposed Technical Specification Change", Serial No. 731, February 14, 1985.

19. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), "Virginia Power Reduction in Rod Bow DNBR Penalty for Surry Power Station Unit Nos. 1 and 2", Serial No. 85-064, March 21, 1985.

20. Letter from J.R. Miller (NRC) to W.L Stewart (Vepco), "NRC Audit for VEPCO Utilization of Westinghouse Computer Codes - Surry 1&2 and North Anna 1&2", June 19, 1984.

21. Letter from L.B. Engle (NRC) to R.H. Leasburg (Vepco), April 13, 1982.

22. Letter from L.B. Engle (NRC) to R.H. Leasburg (Vepco), January 27, 1983.

23. J. G. Miller, "Vepco Nuclear Design Reliability Factors", VEP-FRD-45A, (October, 1982).

TABLE 1 - NON-SPECIFIC KEY PARAMETERS

PARAMETER	CONSERVATISM APPLIED		COMMENTS
1. Moderator Temp. Coef.			
a) HZP	Surry	2.6 pcm/°F	The HZP conservatisms are based on the maximum difference in the conservative direction between the measured and predicted values plus a 1 pcm/°F bias. The HFP North Anna conservatism is based on the maximum difference in the conservative direction between the measured and predicted values. The most limiting positive and negative coefficients for the cycle are calculated.
	North Anna	1.8 pcm/°F	
b) HFP	Surry	3.0 pcm/°F	
	North Anna	2.2 pcm/°F	
2. Doppler Temp. Coef.		---	Calculated for range of conservative operating conditions including HZP-HFP, rodded conditions, cycle burnups. Range of values used in safety analyses.
3. Doppler Power Coef.		---	
4. Delayed Neutron Frac.		1.05	Calculated at conservative cycle burnups and operating conditions to determine limiting values for reload.

TABLE 1 - NON-SPECIFIC KEY PARAMETERS

PARAMETER	CONSERVATISM APPLIED	COMMENTS
5. Prompt Neutron Lifetime	1.05	Calculated for cycle burnup which provides limiting value. For further discussion see question 14.
6. Boron Worth	---	Accident analysis assumes value of 16 pcm/ppm. Reload values calculated are in 7-8 pcm/ppm range.
7. Control Bank Differential Worth	1.10	Conservative xenon and power distributions used in calculation.
8. Shutdown Margin	---	Assumes most reactive rod stuck. Calculated rod worth is reduced by 10%. Impact of rod insertion limits is accounted for. Calculation performed for range of cycle burnups to determine most limiting value.

TABLE 1 - NON-SPECIFIC KEY PARAMETERS

PARAMETER	CONSERVATISM APPLIED	COMMENTS
9. Boron Concentration	Surry 36 ppm North Anna 18 ppm	Conservatisms applied are based on maximum difference in the conservative direction between measured and predicted values for HZP. Values calculated for cycle burnup which provides most limiting value.
10. Trip Shape	---	Calculated using cycle burnup, power distribution, and xenon distribution which minimizes initial reactivity insertion.
11. Trip Magnitude	---	Calculated assuming most reactive rod stuck. Calculated rod worth is reduced by 10%. Impact of rod insertion limits for near full power operation is accounted for.
12. Fuel Power Census		
a) Normal Operation	1.08	Calculated for range of cycle burnups to determine most limiting value. Control bank insertion limits are accounted for.

TABLE 2 - SPECIFIC KEY PARAMETERS

PARAMETER	NRF[1]	CONSERVATISM APPLIED	
I. Integral Rod Worth	1.10		
1. Dropped Rod		1.10	1. Conservative cycle burnup
			2. Conservative axial power distribution
2. Ejected Rod		1.10	1. No feedback
			2. Conservative axial burnup distribution
			3. Conservative axial xenon distribution
II. Differential Rod Worth	2pcm/step		
1. Reactivity Insertion Rate due to Rod Withdrawal		1.10	1. Applied to maximum value calculated
			2. Conservative axial xenon distribution
			3. 10% translates to 3-10 pcm/step (HFP-HZP)
III. Radial Peaking Factor	1.05		
1. Normal Operation		1.08	1. Conservative cycle burnup
			2. Conservative versus power level
			3. Conservative axial xenon distribution
2. Rod Misalignment		1.11	1. Conservative cycle burnup
			2. Conservative axial xenon distribution

TABLE 2 - SPECIFIC KEY PARAMETERS

PARAMETER	NRF[1]	CONSERVATISM APPLIED		
IV. Power Peaking Factor	1.075			2D/3D synthesis
	1.0815[2]			1D/2D/3D synthesis
1. F_QxP vs Core Height		(CAOC)	1.0815[3]	1. Base load operation
				2. Load follow operation
				3. Impact of previous cycle burnup
		(RPDC)	1.0815[3]	1. Limiting cycle burnups
				2. Conservative axial xenon distributions
				3. Large number of axial power distributions analyzed
				4. Axial power distributions bound expected operating range
2. Ejected Rod Hot Channel Factor			1.10	1. No feedback
				2. Conservative axial xenon distribution
				3. Conservative axial burnup distribution
3. Overpower Peak Kw/ft			1.0815[4]	1. Conservative axial xenon distribution
				2. Conservative axial burnup distribution

1. Reference 23
2. Reference 1
3. An engineering uncertainty is also applied
4. An engineering uncertainty and densification spike factor are also applied

APPENDIX E

Acceptance for Referencing of Licensing Topical Report VEP-FRD-42 Revision 1, "Reload Nuclear Design Methodology"

(Sent by Nuclear Regulatory Commission letter Serial No. 86-516 to NRC dated Jul 29, 1986)

Total Pages : 10

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

JUL 29 1986

Serial # 86-516

Rec'd. AUG 04 1986

Nuclear Operations
Licensing Supervisor

Mr. W. L. Stewart, Vice President
Nuclear Operations
Virginia Electric and Power Company
Richmond, Virginia 23261

Dear Mr. Stewart:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT VEP-FRD-42 REVISION 1, "RELOAD NUCLEAR DESIGN METHODOLOGY"

We have completed our review of the subject topical report submitted by the Virginia Electric and Power Company (VEPCO) by letter dated September 19, 1985. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that VEPCO publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, VEPCO and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Charles E. Rossi

Charles E. Rossi, Assistant Director
Division of PWR Licensing-A

Enclosure:
As stated

SAFETY EVALUATION REPORT

Topical Report Title: Reload Nuclear Design Methodology

Topical Report Number: VEP-FRD-42 Revision 1

Topical Report Date: August 1985

INTRODUCTION

This topical report describes Virginia Power's methodology for designing reload cores and performing reload safety analyses. Virginia Power has had access to Westinghouse reload design and safety analysis codes since 1981, when a transition program aimed at enabling Virginia Power to progressively assume design and safety analysis responsibilities was initiated. Virginia Power's reload safety analysis methods are, consequently, similar to the Westinghouse reload safety analysis methodology[1].

2. SUMMARY OF TOPICAL REPORT

The analytical models used by Virginia Power are described in Sections 2.1.1 through 2.1.5 of the topical report. The analytical models for nuclear design calculations utilize the PDQ07, FLAME and NOMAD codes. Each of these models has previously been approved by the staff[2-5]. Neutron spectrum generation and calculation of few group constants for the nuclear design models is performed with the B&W NULIF[6] code. The PDQ07 model is used for standard two-dimensional diffusion-depletion calculations utilizes either a discrete mesh (one mesh block per fuel pin) or coarse mesh.The nodal FLAME model used for three-dimensional calculations utilizing 32 axial nodes. The NOMAD model utilizes one-dimensional, two-group diffusion theory with 32 axial mesh points and is used for load follow and power distribution control calculations. The RETRAN model employs point kinetics and plant specific representations of components and systems such as pumps, safety and relief valves and control systems. The RETRAN model is used in reactor coolant system transient analyses, while the COBRA model is used in detailed thermal-hydraulic analyses. Both models have been approved by the NRC staff.[7,8]

The nuclear design methods employed by Virginia Power are described in Sections 2.2.1 through 2.2.3 of the topical report. The analytical methods used in transient and thermal-hydraulic analysis are described in referenced topical reports. The nuclear design methods described are the usual methods employed for core depletion calculations and determination of core reactivity parameters and coefficients. In addition to the codes mentioned above, Virginia Power has indicated that they use the Westinghouse LOFTRAN[9] code for the dropped rod control cluster assembly (RCCA) event,and the Westinghouse LOCA code package for the analysis of the loss of coolant accident (LOCA). Fuel performance analyses are performed by Westinghouse on receipt of expected operational data for the cycle from Virginia Power.

The overall reload design process is described in Section 3.0 of the topical report. The process is carried out in three phases. In the initial phase, a core loading pattern is selected and optimized on the basis of cycle energy requirements and operational constraints. In the second phase key analysis parameters are determined for the optimized reload core, and the key analysis parameters are shown to be bounded by the limiting values of these parameters assumed in a reference safety analysis, or a reanalysis or reevaluation of the affected accidents is performed. The second phase, therefore, demonstrates that the reload core can be operated safely. In the last phase physics design predictions necessary for the support of plant operations are determined and documented.

Design and optimization of the core loading pattern is discussed in Sections 3.2.1 and 3.2.2 of the topical report. The design process is initiated by a review of design basis information such as operational requirements, safety criteria, operational and technical specification limits, and reload safety analysis parameters. The fuel loading pattern is shuffled and optimized to meet the requirements of maximum permissible radial peaking factor, minimum permissible shutdown margin, and the technical specification limits on the moderator temperature coefficient.

Reload safety methods used by Virginia Power are discussed in Sections 3.3.1 through 3.3.4.7 and in Section 3.4 of the topical report. The methodology used is similar to the Westinghouse "bounding analysis" method. It assumes the existence of a valid conservative safety analysis, the reference analysis, and a set of key analysis parameters that fully describe the accident under study. If all key analysis parameters for a reload core are conservatively bounded by the values of these parameters for the reference analysis, the reference safety analysis applies, and further analysis is unnecessary. When a key analysis parameter is not bounded, further analysis is considered necessary to ensure that the required safety margin is maintained. This last determination is made either through a complete re-analysis of the accident, or through a simpler though conservative evaluation process. The key analysis parameters are determined from conservative static calculations. Discussions of key analysis parameters such as rod insertion limits, shutdown margin, trip reactivity shape, reactivity coefficients, delayed and prompt neutron data, and power peaking factors are presented in Sections 3.3.3 through 3.3.3.6 of the report. Specific accidents such as uncontrolled control rod bank withdrawal, rod misalignment error, rod ejection, steam line break, LOCA, boron dilution and overpower transients are discussed in Sections 3.3.4 through 3.3.4.7. A list of evaluated condition II, III and IV accidents are presented in Table 1, while Table 2 presents a list of key analysis parameters used in the safety evaluation process. Preparation of the nuclear design report for use during startup physics tests and in the operation of the reactor cycle is described in Section 3.5 of the topical report.

3. SUMMARY OF EVALUATION

The evaluation of VEP-FRD-42 was based mainly on an assessment of the scope and applicability of the proposed methods and the general methodology presented. The following sections address these topics.

3.1 Scope and Applicability

The purpose of the topical report is two-fold: (i) to provide a description of the determination of nuclear safety analysis parameters, and (ii) to provide a discussion of the use of the calculated safety analysis parameters (nuclear, thermal-hydraulic and fuel performance) in performing the "bounding analyses" and establishing the safe operation of the reload core. The fuel performance safety analysis parameters are supplied by the fuel vendor. Virginia Power's methods for transient and thermal-hydraulic analyses have been described in separate topical reports[7,8] that have been reviewed and approved by the NRC staff. In response to our request, Virginia Power has discussed the incorporation of the results of the safety evaluation in the limiting conditions of operation, limiting safety system setpoints, and technical specifications for a reload cycle (Reference 10, responses to Questions 4 and 16). Virginia Power has also described their review of design basis information to ensure that all information provided is current and complete before the safety evaluation process is initiated (Reference 10, response to Question 6). With the incorporation of this additional information discussed above, we find that the two main objectives of the topical report have been served.

Although Virginia Power expects the methods presented in VEP-FRD-42 to be, in principle, valid for both Westinghouse/non-Westinghouse fuel mixes as well as cores designed by other vendors for use in Westinghouse designed plants, 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.

3.2 Methodology

All codes used by Virginia Power in the physics and thermal-hydraulics analyses of the reload core have been reviewed and approved by the NRC staff (Reference 10, response to Question 2). In addition, Virginia Power's utilization of Westinghouse computer codes in selected areas of safety evaluation was the subject of an NRC audit in 1984.[11] Based on the results of this audit and the present review we find Virginia Power's calculational methods for physics and thermal-hydraulic analysis of reload cores acceptable. VEP-FRD-42 provides descriptions in some detail of core depletion calculations, determination of core reactivity parameters and coefficients,and calculations of control rod and soluble boron worth. These calculational procedures follow conventional methods using approved codes, and are therefore acceptable.

In the safety evaluation process, Virginia Power proposes to use a bounding analysis concept (Reference 10, response to Question 1). This approach employs a list of key analysis parameters and limiting directions of the key analysis parameters for various accidents (Reference 10, response to Question 5). The bounding analysis approach is a perturbation approach in which the impact of the perturbations from the reference core are evaluated in place of a complete new safety analysis of each reload core. If all key analysis parameters are conservatively bounded, the reference safety analysis is assumed to apply, and no further analysis is necessary.If one or more key analysis parameters is not bounded, further analysis or evaluation of the accident in question is performed.

The validity of the bounding analysis concept depends on several aspects of the key analysis parameters. Chief among these are: completeness of the set of key analysis parameters with respect to a given accident, the assumption of a monotonic dependence of an accident consequence on the

value of a given key analysis parameter, and the assumption that the effects of two or more key analysis parameters are decoupled. The correlation of the key analysis parameters and their limiting directions with the various accidents (Reference 10, response to Question 5) have been reviewed and were found acceptable. The assumptions of monotonicity and decoupling of the key analysis parameters are generally valid provided the parameters do not differ largely from their reference values. For cases in which the reference analysis is bounding, the key analysis parameters show only small variations from the reference values, and the assumptions of monotonicity and decoupling are not of concern. In cases where the reference analysis is not bounding, and a full reanalysis is made, the assumptions indicated are not required. It is only in cases where a reevaluation rather than a reanalysis is made that these assumptions need to be justified. Virgina Power has not established quantitative criteria to determine the point at which a re-evaluation rather than a complete reanalysis becomes permissible. However, Virginia Power has indicated that in each case an evaluation is performed documentation containing the exact numerical values pertaining to the violation including a detailed discussion of the reasoning and approach used will be submitted in the Reload Safety Evaluation Report. Given these conditions, we find the use of quantitative evaluations, based on known sensitivities in cases where a small violation of parameter limits exists, acceptable.

Since Virginia Power uses a different set of codes than Westinghouse to determine the values of the key analysis parameters, there is a concern that the existence of systematic biases between values of key analysis parameters calculated by Westinghouse and Virginia Power would impact the current limiting values of the parameters assumed in the safety evaluation. In response to this concern, Virginia Power has indicated that they have not encountered such systematic biases.

Virginia Power uses the NOMAD code to simulate operation under Constant Axial Offset Control (CAOC) and Relaxed Power Distribution Control (RPDC). Use of NOMAD in the simulation of CAOC and RPDC has been reviewed and approved by the NRC staff.[5] The main impact of RPDC operation would be on the trip reset function, f(I), associated with the overpower and overtemperature T trips. Virginia Power has indicated that analyses to date show that ample margin exists in the existing f(I) function to accommodate the wider range of axial power shapes inherent in RPDC (Reference 10, response to Question 16). Since the Virginia Power safety evaluation process utilizes the bounding concept using calculational methods that are acceptable by themselves,we find the general methodology used by Virginia Power acceptable for the safety evaluation of reload cores. However, the clear dependence of VEP-FRD-42 on Westinghouse methodology precludes the application of VEP-FRD-42 in its present form to non-Westinghouse or mixed reloads.

4. CONCLUSIONS

We have reviewed the Reload Nuclear Design Methodology described in VEP-FRD-42, Revision 1 and find it acceptable for referencing by Virginia Power in licensing Westinghouse supplied reloads of Westinghouse supplied reactors.

References

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10. Letter from W.L. Stewart (Virginia Power) to Harold R. Denton (NRC) dated May 2, 1986.

11. Letter from J.R. Miller (NRC) to W.L. Stewart (Vepco), "NRC Audit for Vepco Utilization of Westinghouse Computer Codes - Surry 1 & 2 and North Anna 1 & 2," June 19, 1984.