Reload Nuclear Design Methodology

VEP-FRD-42 Rev. 1-A

Nuclear Engineering
Engineering and Construction Department
September, 1986

VIRGINIA POWER | NORTH CAROLINA POWER | WEST VIRGINIA POWER

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RELOAD NUCLEAR DESIGN METHODOLOGY

BY

NUCLEAR ENGINEERING STAFF

NUCLEAR ENGINEERING DEPARTMENT

VIRGINIA POWER

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

JUL 29 1986

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

1. F.M. Bordelon, et al., "Westinghouse Reload Safety Evaluation Methodology (WCAP-9272A)," March, 1978.

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

3. J.R. Rodes, "The PDQ07 One Zone Model," VEP-FRD-20A, July, 1981.

4. W.C. Beck, " The Vepco FLAME Model," VEP-FRD-24A, July, 1981.

5. S.M. Bowman, "The Vepco NOMAD Code and Model," VEP-NFE-1-A, May 1985.

6. W.A. Wittkoff, et al., "NULIF - Neutron Spectrum Generator, Few Group Constant Calculator, and Fuel Depletion Code," BAW-10115, June, 1976.

7. N.A. Smith, "Vepco Reactor System Transient Analysis Using the RETRAN Computer Code," VEP-FRD-41, March, 1981.

8. F.W. Sliz, "Vepco Reactor Control Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT Computer Code," VEP-FRD-33A, October, 1983.

9. T.W.T. Burnett, et al., "LOFTRAN Code Description, WCAP-7907," June, 1972.

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.

CLASSIFICATION/DISCLAIMER

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

SECTION 1.0 - INTRODUCTION

The Virginia Power methodology for determining a reload design for its nuclear units is an iterative process. The process involves determining a fuel loading pattern which provides the required energy and then showing through analysis or evaluation that the loading pattern meets all safety criteria imposed on the plant. Should the proposed loading pattern not meet the safety analysis criteria for the current operating requirements, the loading pattern is revised or changes are made in the operating requirements (Technical Specifications) to ensure the plant will not be operated at conditions which violate the applicable safety analysis criteria for the proposed loading pattern.

This report presents the methodology employed by Virginia Power for performing a nuclear reload design analysis. It covers analytical models and methods, reload nuclear design, reload safety analysis, and an overview of analyzed accidents and key parameter derivations.

Detailed in this report are: (1) design bases, assumptions, design limits and constraints which must be considered as part of the design process, (2) the determination and fulfillment of cycle energy requirements, (3) loading pattern determination, (4) the safety evaluation of the loading, 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 used by Virginia Power for reload design and safety analysis are:

1. the Vepco PDQ07 Discrete Model
2. the Vepco PDQ07 One-Zone Model
3. the Vepco FLAME Model
4. the Vepco NOMAD Model
5. the Vepco RETRAN Model
6. the Vepco COBRA-IIIc/MIT Model

Topical reports for each of these models have been approved for reference in licensing applications by the Nuclear Regulatory Commission (References 1-6). Prior to January 15, 1985 Virginia Power was known as Virginia Electric and Power Company (Vepco) and the topicals referenced were submitted using Vepco in their titles.

2.1.1 Virginia Power PDQ07 Models

The Virginia Power PDQ07 Discrete and One-Zone Models perform two-dimensional (x-y) geometry diffusion-depletion calculations for two neutron energy groups. These models utilize the NULIF (Reference 7) code and several auxiliary codes to generate and format the cross section input, perform shuffles, and other operations. The two models are differentiated according to their mesh size (i.e., either a discrete mesh or coarse mesh). The Discrete model utilizes one mesh block per fuel pin, while the One-Zone model has 6x6 mesh blocks per fuel assembly. An eighth, quarter, or half core symmetric two-dimensional geometry or a full core two-dimensional geometry may be specified for either model. The effects of nonuniform moderator density and fuel temperatures are accounted for with thermal-hydraulic feedback. More complete descriptions of these models and their auxiliary codes may be found in References 1 and 2 for the Discrete and One-Zone models, respectively.

The PDQ07 Models are used to calculate two-dimensional radial power distributions, delayed neutron data, radial peaking factors, assemblywise burnup and isotopic concentrations, integral rod worths, differential boron worths and boron endpoints, xenon and samarium worths and core average reactivity coefficients such as temperature and power coefficients. In addition, the PDQ-INCORE decks used in startup physics testing and core follow are generated using the PDQ07 Discrete model. These decks contain PDQ07 predicted power and flux distributions used by the INCORE Code (Reference 8) along with thimble flux measurements to make predicted to measured power distribution comparisons.

2.1.2 Virginia Power FLAME Model

The Virginia Power FLAME Model is used to perform three-dimensional (x-y-z geometry) nodal power density and core reactivity calculations using modified diffusion theory with one neutron energy group. The model utilizes the NULIF code and several auxiliary codes to generate and format cross section input, perform shuffles, and other operations. Each fuel assembly in the core is represented by one radial node and 32 axial nodes. As with the PDQ07 Models, the effects of nonuniform moderator density and fuel temperature are accounted for by thermal-hydraulic feedback. A more complete description of this model and its auxiliary codes may be found in Reference 3. The FLAME Model is used in calculating and evaluating three-dimensional or axial effects such as differential rod worths, axial power and burnup distributions, and control rod operational limits. FLAME Model predictions are normalized to those of the PDQ07 model when applicable.

2.1.3 Virginia Power NOMAD Model

The Virginia Power NOMAD Model performs one-dimensional (z) geometry, diffusion-depletion calculations (with thermal-hydraulic feedback) for two neutron energy groups. The NOMAD model makes use of data from the PDQ07 Discrete, PDQ07 One-Zone, and FLAME models for normalization. As in the FLAME model the active fuel length is represented by 32 axial nodes. The NOMAD model and its auxiliary codes are described in detail in Reference 4. The NOMAD model is used in the calculation of core average axial power distributions, axial offset, axial peaking factors, differential control rod bank worths, and integral control rod worths as a function of bank position. In addition, NOMAD has the capability to perform criticality searches on boron concentration, control rod position, core power level, and inlet enthalpy. Simulation of load follow maneuvers, performance of Final Acceptance Criteria analysis, and Relaxed Power Distribution Control (RPDC, Reference 9) may also be performed with the NOMAD model.

For the remainder of this report the PDQ07, FLAME, and NOMAD models will be referred to generically as the 2-D, 3-D, and 1-D models, respectively.

2.1.4 Virginia Power RETRAN Models

The Virginia Power RETRAN Models (Reference 5) are used to perform reactor coolant system (RCS) transient analyses. As part of the reload methodology, these models are used with the safety analysis criteria to provide additional support for those instances where there has been a violation of the previously identified licensing limit. Such reanalysis begins with either the one loop or the two loop base model with the transient specific input modifications necessary to perform the licensing analysis.

The Virginia Power 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.

2.1.5 Virginia Power COBRA Models

The Virginia Power COBRA models are used to perform a detailed thermal-hydraulic analysis of the reactor core. Details of this model are described in Reference 6. COBRA solves 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 W-3 CHF correlation. COBRA can perform either steady state or DNBR calculations or transient DNBR analyses with forcing function which have been supplied by the RETRAN code. Steady state applications include thermal limit generation, DNBR statepoint analyses and axial shape verification for RPDC. Examples of transient applications are loss of flow and locked rotor DNBR analysis.

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, a depletion of the reload core is performed based on a nominal, (i.e. best estimate), end-of-cycle (EOC) burnup for the previous cycle. The reload core loading pattern is depleted at hot full power (HFP), all rods out (ARO) conditions using a 2-D model in quarter-core geometry. During the depletion, criticality is maintained by varying the boron concentration (i.e., performing a criticality search). These calculations provide x-y relative power distributions, burnup predictions and an estimate of the cycle's full power capability.

For the safety evaluation of a reload loading pattern, additional depletions using the 1D, 2D, and 3D models are performed to bound the EOC burnup window for the previous cycle which is typically +/- 30 effective full power days (EFPD) about the nominal EOC burnup. These window depletions allow the sensitivity 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 must be made as near to the actual operating conditions of the reload as possible. To ensure this, those predictions dependent on burnup are calculated based on a previous EOC burnup that is within +/- 2 EFPD of the actual burnup.

2.2.2 Core Reactivity Parameters and Coefficients

The kinetic characteristics of the core are described by the core reactivity parameters and coefficients. 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 corewise basis using a 2-D model 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 Temperature and Power Coefficients

The Doppler temperature coefficient (DTC) is defined as the change in reactivity per degree change in the fuel temperature. This change in reactivity is due mainly to the change in the resonance absorption cross sections for Uranium 238 and Plutonium 240 as the fuel temperature changes.

The moderator temperature coefficient (MTC) is defined as the change in reactivity per degree change in the moderator temperature. The moderator defect is the integral of the moderator temperature coefficient over the appropriate temperature range, usually from HZP to HFP.

The isothermal temperature coefficient (ITC) is defined as the change in reactivity per degree change in the moderator and fuel temperatures. Thus, the isothermal temperature coefficient is the sum of the moderator and Doppler temperature coefficients. Isothermal temperature coefficients are of particular interest at hot zero power (HZP) when the core is uniformly heated and reactivity changes due to temperature changes can be readily measured and compared to predicted values.

The total power coefficient (TPC) is defined as the change in core reactivity per percent change in power due to the combined effect of the moderator and fuel temperature changes brought about by core power level changes. The Doppler "only" power coefficient (DPC) is defined as the change in reactivity per percent change in power due only to the fuel temperature changes brought about by core power level changes. The power defect is the integral of the power coefficient over the appropriate power range, usually zero to full power.

For Virginia Power, the method of calculating temperature or power coefficients depends on whether the parameter is desired for HZP conditions or "at-power" conditions. In the calculation of at-power coefficients, the thermal-hydraulic feedback is included in the 2-D calculations while the HZP calculations are performed without thermal-hydraulic feedback.

Coefficients at HZP

Temperature coefficients at HZP (ITC, DTC, MTC) are calculated using a set of four 2-D calculations run without thermal-hydraulic feedback. Two of the calculations are performed at core average fuel and moderator temperatures +/-5°F about the HZP temperature. These two cases will provide an isothermal temperature coefficient at HZP power using the following formula:

$$\text{ITC (pcm/°F)} = \frac{(\text{Keff1} - \text{Keff2})*(10^5 \text{ pcm})}{\text{Keff1*Keff2*(Tmod1} - \text{Tmod2)}}$$

The additional two calculations are used to calculate a Doppler temperature coefficient. By holding the moderator temperature constant at the HZP value and varying the fuel temperature by +/-5°F about the HZP value, the DTC can be calculated as:

$$\text{DTC (pcm/°F)} = \frac{(\text{Keff1} - \text{Keff2})*(10^5\ \text{pcm})}{\text{Keff1}*\text{Keff2}*(\text{Tfuel1} - \text{Tfuel2})}$$

From these calculations a moderator temperature coefficient for HZP conditions may be obtained by taking the difference between the isothermal and Doppler temperature coefficients.

Coefficients at Power

When calculating the ITC, DTC, and MTC for at power conditions four 2-D calculations are again performed. However, the calculations are run with thermal-hydraulic feedback which incorporates cross-section fits on fuel temperature and moderator temperature over the range of conditions from HZP to above full power conditions. The isothermal temperature coefficient at power is calculated by performing calculations, at core temperatures slightly above and below the reference values (normally +/-5°F about the reference). The core average temperatures are adjusted by changing the moderator inlet enthalpy of the core in the 2-D model. For these calculations the power levels are held constant. The coefficient for the change in reactivity due to the core average temperature change (ITC) can then be calculated using the same formula used for the HZP coefficient.

To calculate the Doppler temperature coefficient for at-power conditions two calculations are needed. These calculations adjust the fuel temperatures to values +/-5°F about the reference value by adjusting the power above and below the reference power while adjusting the moderator inlet enthalpy to keep the core average moderator temperature constant at the reference value. The at-power Doppler temperature coefficient can now be calculated using the same formula as the HZP Doppler temperature coefficient.

The moderator temperature coefficient for the reference at-power condition is the difference between the isothermal and Doppler temperature coefficients for the at-power conditions.

To calculate the power coefficients (TPC, DPC) requires 2-D calculations using thermal-hydraulic feedback. The total power coefficient is calculated by performing two calculations +/-5% about the reference power. The total power coefficient is calculated as the change in reactivity divided by the change in power:

$$\text{TPC (pcm/\%P)} = \frac{(\text{Keff1} - \text{Keff2})*(10^5 \text{ pcm})}{\text{Keff1}*\text{Keff2}*(\text{P1} - \text{P2})}$$

The Doppler only power cofficient is calculated using the results from the Doppler temperature and total power coefficients. As the fuel temperature is essentially linear with respect to power level in the range of interest the Doppler power coefficient may be expressed as follows:

$$\text{DPC (pcm/\%P)} = \text{DTC (pcm/}^\circ\text{F)} * \frac{(\text{Tfuel1} - \text{Tfuel2})}{(\text{P1} - \text{P2})}$$

where Tfuel1, Tfuel2, P1, and P2 are the fuel temperatures and power levels used to calculate the total power coefficient.

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 with a 2-D model by noting the change in core average reactivity due to a change in the corewise boron concentration, (normally +/-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 calculated using a 2-D model, and are found by weighting the delayed neutron fraction for each fissionable isotope in each group by the core integrated fission rate of that isotope.

2.2.2.4 Xenon and Samarium Worths

Xenon and samarium are fission product poisons with relatively large thermal absorption cross sections. Their effect on core reactivity requires the calculation of the reactivity worth of xenon and samarium during changes in core power level under various core conditions, particularly for plant startups, power ramp-up and ramp-down maneuvers and reactor trips. Xenon and samarium worths are determined using information from the 2-D model.

2.2.3 Core Reactivity Control

Relatively rapid reactivity variations in the core are controlled by the full length control rods. The full length control rods are divided into four control banks (designated D, C, B, and A) and two shutdown banks (designated SB, and SA). The control banks D, C, B, and A 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 with a 2-D model 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 as a function of axial position are calculated using a 3-D or 1-D model. 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 no load, the reactivity change from CZP to HZP is controlled by changing the soluble boron concentration. 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 with a 2-D model.

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 and satisfy 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 in which they are performed, are:

I. Core loading pattern design and optimization.

II. Determination of core physics related key analysis parameters for reload safety analysis.

III. Design report 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). In addition, some preliminary Phase II calculations are performed to verify that conditions on radial peaking factors, moderator temperature coefficient, and shutdown margin 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 a "worst case" assumption philosophy to ensure the results are conservative for the reload. If a key analysis parameter for the reload cycle exceeds the limiting value, the corresponding transient must be evaluated or reanalyzed using the reload value. Should the reload value cause a violation in the safety criteria, a new reload design or possibly new operating limits (Technical Specifications) may have to be instituted.

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, insure 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 insure 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).
4. Specific operating limitations on the plant as contained in the Technical Specifications.
5. Plant or Technical Specification changes implemented or expected to be implemented since the last reload.
6. Reload safety analysis parameters (mechanical, nuclear, and thermal/hydraulic) used in the safety analyses up to and including the previous cycle.

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 prior experience an 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. Using a 2-D model this loading pattern will be modeled and depleted to determine the cycle's energy output and radial 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 met.

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

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

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

When a pattern meets the above conditions, the enrichment and number of fresh assemblies along with any burnable poison requirements are set. At this point, the loading pattern is optimized for cycle length and power distribution by shuffling the fuel and/or burnable poison. Once the optimum pattern has been established it is evaluated and analyzed to determine whether all core physics related limits can be met during the operation of the unit.

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 reload safety analysis. For each reload cycle, the effects of reload core physics related or plant related changes must be 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 FSAR 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 any consequence 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 design stages and documented in the FSAR. The Virginia Power accident analysis is typical in that the complete FSAR analysis was performed by the NSSS vendor. However, Virginia Power has verified the key Condition I, II, III, and IV FSAR analyses (excluding LOCA) and the safety of its plants using its own analysis capability (References 5 and 13). The four categories of accidents based on their anticipated frequency of occurrence and potential for public harm are described in References 10 and 11. The accident analyses consider all 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 analysis process. First, steady state nuclear calculations are made for the 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 as a function of these key analysis parameter values. The accident analyses are transient calculations which 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 first determined the key nuclear parameter values expected to be bounding over the plant lifetime. The bounding values for these key parameters may occur sometime during the first cycle of operation or during a subsequent cycle. Therefore, depletion studies were performed and the key parameters were determined for several cycles of operation in order to obtain a set of key parameters which had a high probability of being bounding over plant life. These bounding key parameters are called the (initial) current limits. FSAR accident analyses were performed using these bounding parameters.

The FSAR demonstrates by determining key nuclear parameters and detailing the results of the accident analyses that the plant is safe. However, an unbounded key analysis parameter could occur in a reload cycle. For this reason, all key analysis parameters must be explicitly determined for each reload.

For a typical reload cycle, some depleted fuel is removed from the core and replaced by fresh fuel. The depleted fuel remaining in the core and the new fuel are arranged within the core so that power peaking criteria are met. Other 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 key parameter. The reload evaluation process is complete if the acceptance criteria delineated in the FSAR are met, and internal documentation of the reload evaluation is provided for the appropriate Virginia Power 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. The NRC will be informed of the results of the evaluation process in accordance with the requirements of 10CFR50.59.

Therefore, the overall process is as follows:

1) Determine expected bounding key analysis parameters (initial "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 initial 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 Virginia Power Surry Units 1 and 2 and North Anna Units 1 and 2 and will be used by Virginia Power in the future.

The accidents analyzed for the FSAR 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 at conservative conditions, and the specific parameters (designated "S" in Table 2) are generated by statically simulating an accident. The generation of these parameters are performed under conservative conditions for such core parameters as xenon distribution, power level, control rod position, and operational history. The third type of key parameters are fuel performance and thermal-hydraulic related parameters (designated "F" in Table 2).

The methods which will be employed by Virginia Power to determine these key parameters will be consistent with the methods documented in References 9 and 12.

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 the limiting values of the parameter are determined. These include conservative assumptions for such core parameters as xenon distributions, power level, control rod position, operating history, and burnup. 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.

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 to maintain an acceptable power distribution during normal operation, acceptable consequences following postulated accidents, and also insure that the minimum shutdown margin (SDM) assumed in the safety analyses is available. The current RIL's for the unit are given in the plant Technical Specifications.

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

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 a 1-D or 3-D model simulation of the control banks moving into the core with normal overlap while assuring the minimum shutdown margin is maintained at all power levels and insertions from HFP to HZP. The calculation is performed at 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 bases. 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 current Technical Specifications. If they violate the current Technical Specifications, a change is submitted to the NRC requesting approval of these limits which would then become the final rod insertion limits following NRC review and approval of the associated Technical Specifications change.

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 with a 2-D model 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. This reactivity comes from several sources. The negative power coefficient at HFP implies there will be a positive reactivity insertion for reduction in power when going from HFP to HZP conditions. This reactivity is calculated as a power defect using a 2-D model. The defect is conservatively calculated by increasing the total moderator temperature change above that seen from HFP to HZP conditions. In addition, axial flux redistribution and void collapse may occur when going from HFP to HZP causing positive reactivity insertion. As these will not be seen when performing the defect calculations with the 2-D model they must be accounted for separately. The redistribution factor may be explicitly calculated with a 3-D model or a conservative generic value may be assumed. For the reactivity associated with void collapse a conservative generic estimate is used in the shutdown margin calculation.

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 for both beginning and end of cycle.

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 must be verified by rod drop measurements taken during the startup tests for each cycle.

The trip reactivity shape is generated with a 1-D model. The model is depleted with all rods out at hot full power, equilibrium xenon to the end of cycle (EOC) to determine the depletion step (time in life) which has the most bottom peaked axial power distribution. This time in life is used in order to minimize the initial worth of the rods when tripped in. A control bank is inserted to push the axial offset to its negative Technical Specifications limit. A single bank normalized to the minimum trip reactivity worth is then inserted in discrete steps and the integral worth of the control rods corresponding to each step is calculated.

A conservative trip reactivity shape curve is one which shows less negative reactivity insertion for the major part of the rod insertion (i.e., except for the endpoints which are always equal), than would be expected for an actual best estimate trip calculation based on operational power shape data. The FSAR safety analysis is based on a conservative curve generated using the methodology described above.

A trip reactivity shape is generated for each reload. If the reload shape shows the same reactivity insertion or more reactivity insertion than the current limit shape for the rod insertion, it is bounded by the current limit shape. If the reload shape shows less negative reactivity insertion than the current limit shape for any part of the insertion, the reload shape is unbounded and the effect must be evaluated. If the reload shape has only minor deviations over some parts of the current limit shape, a simple quantitative evaluation may be made which conservatively estimates the magnitude of the effect and explains why reanalyses (of transients affected by trip reactivity shape) do not have to be made. In this case the current limit reactivity shape is not changed. If the reload shape is found more limiting than the current limit shape, the transients affected by trip reactivity shape are reanalyzed. The reload trip reactivity shape will become the new current limit if the results of the analyses show no violations of appropriate analysis acceptance criteria. As previously stated, the NRC will be informed of the results of the evaluation process in accordance with the requirements of 10CFR50.59.

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 2.0.

For each core there is a range of possible values for the coefficients to assume. The coefficients used as key analysis parameters are derived using the appropriate techniques and at the appropriate conditions to obtain the limiting (the maxima and minima which are physically possible) 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 in the manner outlined in Reference 12.

3.3.3.5 Neutron Data

The 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 with a 2-D model using the appropriate cross-section data. The total delayed neutron fraction (total Beta) is the sum of the delayed neutron fractions for the six groups.

The key analysis parameter is the Beta-effective, which is the product of the total Beta 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 Beta-effective, while for others, the maximum expected value is more conservative. The use of conservatively large versus small Beta-effective values is treated on an event by event basis in the manner outlined in Reference 12. Beta-effective is calculated at the beginning and end of each reload cycle to obtain the bounding values for the cycle.

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 NULIF for each region in the core. The key analysis parameter used for transients is the maximum prompt neutron lifetime which occurs at the end of a reload cycle.

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 may be characterized by the radial peaking factor, FdH, and the total peaking factor, Fq. The Technical Specifications give the peaking factor limits. The nuclear design of the core, by judicious placement of new and depleted fuel and by the use of burnable poisons, constrains the peaking factors to be well within the Technical Specification limits. Furthermore, operational instructions, such as the axial power distribution control procedures and the rod insertion limits, also protect the core from power distributions more adverse than those allowed by the Technical Specifications.

For transients which may be DNB limited, the radial peaking factor is of importance. The allowable radial peaking factor increases with decreasing power level and increasing rod insertion. For transients which may be overpower limited, the total peaking factor is of importance. Above 50% power the allowable value of Fq increases with decreasing power level such that the full power hot spot heat flux is not exceeded, i.e., Fq * Power = design hot spot heat flux. For a reload, peaking factors are checked for various power levels, rod positions, and cycle burnups assuming "worst case" 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. The parameters are (or are directly related to) rod worths, reactivity insertion rates, or peaking factors. The static conditions selected are the most conservative conditions for the accident and 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 (References 1, 2, 3, 4, 9, 15).

3.3.4.1 Uncontrolled Control Rod Bank Withdrawal

The rod withdrawal accident occurs when control rod 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 at HZP or HFP and a 1-D or 3-D model is used to perform the calculation.

For the rod withdrawal from subcritical (HZP),the parameter of interest is the maximum differential worth of two sequential control banks (D and C, C and B etc.) moving together at HZP with 100% overlap. The parameter is usually recorded in pcm/inch (where, pcm = percent mille = 100,000 * delta-keff/keff).

In calculating the maximum differential rod worth for two sequential highest worth control banks the following assumptions and conservatisms are used:

1) The shutdown banks are not present in the core.

2) The axial xenon distribution causing the maximum peak differential worth is used.

3) The calculations are performed at the cycle burnups which are expected to maximize the peak differential worth.

The peak differential worth obtained in pcm/step is multiplied by the steps to inches conversion factor to obtain pcm/inch.

The rod withdrawal at power accident differs from the rod withdrawal from subcritical, in that it occurs at-power and assumes that control banks D and C are moving with the normal overlap. It is similar in that a xenon shape which maximizes the peak differential rod worth is used. The parameter of interest is the maximum differential rod worth.

The conservatisms associated with these calculations are:

1) The use of a xenon shape which maximizes the peak differential worth.

2) The performance of the calculations at the cycle burnups which are expected to maximize the peak differential 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, and 4) dropped bank.

The important parameter 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 (FdH). These peaking factors are determined using a 3-D model or a 1-D/2-D synthesis technique. For conservatism, all of the rod misalignment cases are performed at the cycle burnup which maximizes the radial peaking factors. This is generally at the beginning of the cycle, but may have to be determined from the depletion. Typically, a search is made to determine worst case rods for each type of rod misalignment. In addition, 1-D power sharings used in the synthesis are generated assuming conditions which maximize the synthesized FdH and uncertainty factors appropriate to the models used are applied. The maximum FdH peaking factors calculated for each of these types of rod misalignments are used to confirm that the DNB design basis limit 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. To simulate the RCCA misalignment above the bank, full core 2-D calculations with D bank in are made with the worst (the one that causes the highest FdH peaking factor) D Bank rod fully withdrawn. Next a 1-D calculation with D bank in to its insertion limit and the misaligned rod fully out is performed. The 2-D radial power distributions are then synthesized with the 1-D power sharings to determine the maximum FdH. The RCCA misalignment below its bank is bounded by the dropped RCCA analyses for Surry and North Anna as described later. Note that results of the RCCA misalignment upward analysis bound the FdH 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 Surry and North Anna Units have differing protection systems in the event of dropped rod or dropped bank events. A dropped rod or bank in the Surry plant will initiate a turbine runback upon receipt of a rods on bottom signal or a negative flux rate signal which exceeds the system's setpoint. In addition a rod block is activated which precludes the control rods from being withdrawn in the event they are in the automatic mode. The North Anna Units are protected by a negative flux rate trip which trips the plant when a negative flux rate sufficient to exceed the setpoint is received on two of the four excore detectors.

For Surry the maximum FdH for the dropped rod event is calculated using a 1-D/2-D synthesis or a 2-D/3-D synthesis method. Full core 2-D calculations are performed to determine the radial power distributions assuming any control rod (from either control bank or shutdown bank) may have dropped into the core. The radial power distributions are then synthesized with conservative 1-D axial power sharings to determine the maximum FdH.

The dropped rod event for North Anna involves the same type of calculation as above to determine the maximum FdH. However due to the possibility of a dropped rod having insufficient worth to provide a large enough negative flux rate signal for a trip, additional calculations are performed. The automatic rod controller for North Anna receives a signal from one of the excore neutron detectors. Should a rod which has insufficient worth to trip the plant drop in the vicinity of this detector, the controller may begin to withdraw the control rods to compensate for the negative reactivity of the dropped rod. To determine the control bank response the tilt seen by the detectors due to the dropped rod is analyzed. This is provided by the 2-D full core power distributions generated during the FdH calculation. In addition, there is the possibility of two rods dropping which together have insufficient worth to trip the plant. To determine the FdH values for this scenario requires the calculation of 2-D power distributions assuming two seperate rods may have dropped into the core at the same time. Due to the way in which the North Anna control rods are wired, only certain combinations or pairs of rods must be analyzed. Again the detector response is analyzed to determine the effect of the control bank withdrawal.

The dropped bank analysis is performed using 2-D quarter core runs to model the radial power distributions which arise assuming any bank may drop into the core. These radial power distributions are then synthesized with conservative 1-D power sharings to generate FdH values. This analysis is performed only for the Surry Units as the North Anna Units are protected by a negative flux rate trip which is actuated in the case of dropped banks.

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 "worst" ejected rod fails. The rod is ejected completely from the core resulting in a large positive reactivity insertion and a high Fq in the neighborhood of the ejected rod. The "worst" ejected rod is that rod that gives the highest worth (or positive reactivity addition) and/or the highest Fq 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 (Fq). These key parameters are generated using steady state neutron diffusion theory or nodal methods. The rod ejection key analysis parameters for the bounding power levels and burnups must be derived for each initial and reload core. The detailed procedures for producing the rod ejection key analysis parameters are analytical simulations of the above scenario and include determining peaking factors and ejected rod worths. The 1-D, 2-D and 3-D computer models may be used in the rod ejection analysis.

The rod ejection parameter derivation is performed in a conservative manner. One conservatism is the "adiabatic assumption". Although the rod ejection accident is limited by Doppler feedback, the key analysis parameters are derived with all feedback frozen. The adiabatic assumption is that any fuel damage is done in some small time increment after the rod ejection and before feedback can reduce the peaking factor. Deriving the rod ejection parameters with feedback would result in a smaller Fq and ejected rod worth; therefore, deriving them without feedback is conservative.

Another conservatism is that the 1-D and 3-D models are depleted in such a way as to insure 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 D Bank partially inserted, which insures higher worths and peaking factors, for both HZP and HFP, as compared to the best estimate axial burnup shape.

3.3.4.4 Steamline Break

The steamline break (or steambreak) accident is an inadvertant 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 therefore more limiting at EOC, when the MTC is most negative.

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 (3-D model) is performed at end of cycle, hot zero 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 12). 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 is input to the 3-D model, 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 COBRA 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 evaluation methodology which has been employed by Virginia Power is consistent with the methodology used for past cycles of the Surry and North Anna Units by the fuel vendor for units operating under a constant axial offset strategy (CAOC). A description of this methodology can be found in References 4, 12, and 14.

The two (2) primary LOCA key analysis parameters are the "limiting Fq times relative power versus core height envelope" and the "maximum Fq times relative power versus core height points". The first key parameter is a Technical Specifications limit which is based on the total peaking factor assumed in the currently applicable LOCA analysis. As discussed in Reference 14, LOCA analyses assume that the reactor is operating in such a manner that the peak linear heat generation rate in the core is maximized and the most limiting power shape is present. The limiting Fq times relative power versus core height envelope (Fq * P * K(z)) is conservative with respect to the limiting cosine and top peaked power shapes assumed for large and small break LOCA analyses respectively.

To determine these parameters Virginia Power uses either a standard CAOC FAC analysis as described in Reference 4 or a methodology which involves finding an allowable delta-I versus power space which if the reactor is operated within, the Fq limits will not be violated. Delta-I is defined as the difference in power in the top and bottom halves of the core. This methodology, Relaxed Power Distribution Control (RPDC), is described in detail in Reference 9.

These parameters are determined analytically for RPDC in much the same manner as under the CAOC methodology. However, where the analysis performed for CAOC operation determines that no violations occur when the unit is operated within a narrow delta-I band which is constant over the range of 50% to hot full power, the RPDC analysis determines a delta-I space (which bounds the CAOC delta-I space) within which the unit may operate and not produce Fq violations.

The objective of the RPDC analysis is to determine acceptable delta-I 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.

The axial power distributions encountered during normal operation (including load follow) are primarily a function of four parameters: the xenon distribution, power level, control rod bank position, and burnup distribution. For RPDC, reasonable incremental variations that span the entire expected range of values for these parameters must be considered when generating the axial power distributions.

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 to ensure that all possible cases have been accounted for, a xenon "free oscillation" method is used to generate these distributions. 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.

For 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. Control rod bank insertion is limited by the technical specification rod insertion limits. These 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.

In addition 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 cycle burnups, near beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC), are chosen for the RPDC analysis.

The final power distributions used in the RPDC normal operation analysis result from combining the 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 in the 1-D code. Each calculated axial power distribution is used to synthesize an Fq(z) distribution for these conditions using the 1D/2D/3D synthesis method described in Reference 9. Each of these distributions is examined to see if LOCA limits will be met. In addition, the shapes generated within this space are examined to ascertain whether they will meet the thermal-hydraulic constraints imposed by the loss of flow accident (LOFA), and the delta-I range is adjusted accordingly.

To summarize, the procedure for insuring LOCA safety analysis coverage for the reload cycle consists of (1) determining the current limiting (maximum) Fq times relative power versus core height curve; (2) determining the reload core maximum Fq times relative power values for all normal operational modes; and (3) specifying the appropriate Technical Specifications changes if there are envelope violations.

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 minimum shutdown margins are specified in order to provide the required operator response time. For each reload it must be determined if the minimum shutdown margins actually exist at the core conditions and boron concentrations specified. For that determination, 2-D model calculations at the indicated core conditions and boron concentrations are performed.

3.3.4.7 Overpower Evaluations

An overpower condition occurs in a reactor when the 100% power level is inadvertently exceeded due either to an uncontrolled boron dilution or an uncontrolled rod withdrawal. The overpower evaluation key analysis parameter for both of these accidents is the overpower peak kw/ft. The methodology used to derive the key analysis parameter for CAOC is described in Reference 14 (Section 6-2 in particular for rod withdrawal and Section 6-3 in particular for boron dilution).

For RPDC, these accidents may initiate from any condition within the normal operation space determined in the RPDC analysis, therefore the configurations defined by this space are used as initial conditions from which to start the accident. This analysis is performed with the 1-D code and again axial power shapes are generated and Fq(z) distributions are synthesized. These are examined for violations of peak power and DNB limits.

3.3.5 Non-Nuclear Design Key Parameters

Non-nuclear design key parameters are safety analysis inputs from non-nuclear areas such as fuel performance and core thermal-hydraulics. These inputs are derived at the FSAR stage and reviewed for each reload cycle to ensure that the safety analysis assumptions continue to bound the parameter values for the current plant configuration.

The derivation and use of these parameters is discussed in Reference 12 (Section 4.3 in particular).

3.4 SAFETY EVALUATIONS OF RELOAD SAFETY ANALYSIS

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 is 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 5 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 5 and 13.

For those transients requiring core minimum DNBR analyses, the Virginia Power COBRA code is used. The necessary core operating condition inputs are determined from the RETRAN code. Peaking factor inputs are determined from the appropriate nuclear design code. More discussion of the specific COBRA models and inputs is provided in Reference 6.

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 Main Steam System

CONDITION III EVENTS

a) Complete Loss of Forced Reactor Coolant Flow

b) Single Rod Cluster Control Assembly Withdrawal at Full Power

TABLE 1 (CONT.)

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) 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 Differential Worth (NS)
9) Dropped Rod Worth (S)
10) Ejected Rod Worth (S)

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

16) Limiting Total Peaking Factor * Power Vs. Core Height (F)
17) Maximum (from Depletion) Total Peaking Factor * Power Vs. Core Height (S)
18) Radial Peaking Factor (S)
19) Ejected Rod Hot Channel Factor (S)
20) Initial Fuel Temperature (F)

21) Initial Hot Spot Fuel Temperature (F)
22) Fuel Power Census (NS)
23) Densification Power Spike (F)
24) Axial Fuel Rod Shrinkage (F)
25) Fuel Rod Internal Gas Pressure (F)

26) Fuel Stored Energy (F)
27) Decay Heat (F)
28) Overpower Peak KW/FT (S)

NS: Non-Specific
S: Specific
F: Fuel Performance and Thermal-Hydraulics related

3.5 NUCLEAR DESIGN REPORT

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 by Reactor Engineering. This report is used by the Nuclear Operations Department in the preparation of startup physics tests and operator curves 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. The startup physics and core follow 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 as a function of burnup, power, temperature, and rod configuration;

2) Boron worths as a function of burnup, power, temperature, and rod configuration;

3) Isothermal temperature coefficients as a function of

burnup, temperature, rod configuration, and boron concentration;

4) Doppler only temperature coefficients as a function of burnup;

5) Integral bank worths as function of burnup, power, and rod configuration;

6) Differential bank worths as a function of burnup, power, and rod configuration;

7) Delayed neutron data;

8) Relative power distributions and Fxy data as a function of burnup, power, and rod configuration;

9) Xenon reactivity data following startup, trip, and orderly shutdown as a function of power;

10) Samarium worth following various startup and trip scenarios;

11) Total power defects as a function of burnup, power, and boron concentration;

12) Doppler only power defects as a function of burnup and power;

12) Moderator temperature defects as a function of moderator temperature, burnup, and boron concentration;

13) Assemblywise burnup as a function of cycle burnup;

14) As built isotopic tables for average batch as a function of burnup.

15) Most reactive stuck rod worths as a function of burnup, temperature, and boron concentration;

16) K-effective at refueling conditions as a function of temperature and rod configuration.

Core physics measurements taken during the cycle startup and operation are compared to the physics design predictions documented in the Nuclear Design Report to insure that the plant is being operated within safety limits. Results of the measurements and the comparisons to predictions are published by Nuclear Operations 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 Virginia Power closely parallels that of Westinghouse, but 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 groups responsible for reload core safety analysis at Virginia Power are the Reactor Engineering Group and the Safety Engineering Group. These are presently organized as branches of the Nuclear Engineering (NE) Section of the Engineering and Construction Department.

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. This list, which is based on the assumptions made in the currently applicable safety analysis, is prepared by the Nuclear Safety Engineering Group and forwarded to the Reactor Engineering Group of the Nuclear Engineering Department. The Reactor Engineering Group performs the appropriate calculations for generation of the reload values of the key parameters (generally static nuclear calculations) based on this list. The Nuclear Safety Engineering Group then evaluates and, if necessary, reanalyzes any accidents (using transient methods) as required by the results of the key parameter calculations. A

Reload Safety Evaluation (RSE) report is then issued by Nuclear Safety Engineering documenting the results of the safety analysis for the reload cycle. Figure 1 presents a summary of the documentation and information flow of the safety analysis administration for a reload cycle.

Designing a core that meets all safety criteria is sometimes an iterative process involving interaction and trade-offs between the Reactor Engineering and the Nuclear Safety Engineering Groups. 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 slighty 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 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 such a Technical Specifications change. If any 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.

Close interaction between the Reactor Engineering and the Nuclear Safety Engineering Groups allows the development for each reload cycle of a safety evaluation strategy which best suits that particular cycle.

FIGURE 1

SAFETY ANALYSIS ADMINISTRATION FOR A RELOAD CYCLE

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 ---------------------------
|      Nuclear Safety       |
|    Engineering Group      |
 ---------------------------
             |
             |
             V
     Key Parameters List
             |
             |
             V
 -------------------------
| Reactor Engineering     |  --------->  Nuclear Design Report
|       Group             |                      |
 -------------------------                       |
             |                                   V
             |                          --------------------
             V                         |      Nuclear       |
Calculated Key Parameter Values        | Operations Group   |
             |                          --------------------
             |                            |               |
             V                            |               |
 ---------------------------              V               |
|      Nuclear Safety       |      Startup Physics        |
|    Engineering Group      |        Test Report          |
 ---------------------------                              |
             |                                            V
             |                                Core Performance
             V                                       Report
   Reload Safety Evaluation
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