VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
DOMINION'S RELOAD NUCLEAR DESIGN METHODOLOGY TOPICAL REPORT

In April 25 and July 24, 2001 meetings between Dominion and the NRC Staff on our planned fuel transition program, we discussed the program milestones and the key licensing submittals that were necessary to transition to Framatome-ANP fuel at North Anna Power Station. Accordingly, this letter submits Revision 2 of VEP-FRD-42, "Reload Nuclear Design Methodology Topical Report," for review and approval. In order to meet the program milestones, Dominion requests that the NRC review be completed by August 31, 2002.

Dominion's Reload Nuclear Design Methodology Topical Report, VEP-FRD-42-1A Revision 1, September 1986 has been revised to support the transition to Framatome-ANP Mark-BW fuel at North Anna. The NRC, in the SER for Revision 1 stated, "it is clear that the methodology presented is closely related to the Westinghouse methodology, and is applicable in its present form only to Westinghouse supplied reloads of Westinghouse nuclear plants." Revision 2 of this topical report has been revised to address this restriction and to present a revised discussion of the Dominion reload core design methodology. The changes address several types of items that are listed here:

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

The revised topical discusses the Dominion capability to assess changes in fuel design. The focus of these particular changes is primarily upon nuclear core design and NSSS safety analysis design inputs. Dominion predictions of behavior for Framatome-ANP

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Lead Test Assemblies in North Anna Unit 1 demonstrate that nuclear core design tools can model the Framatome-ANP fuel to the same accuracy as the current Westinghouse fuel. This demonstration was accomplished by comparing the predicted and measured power distributions for cores containing both Westinghouse fuel and the Framatome-ANP LTAs. There was no observable difference in the predictions from the Dominion codes for either fuel type.

The minor changes in Framatome-ANP fuel features that could affect safety analysis design inputs are within the modeling capability of Dominion safety and core design analysis codes. Such minor changes include: small change in nominal fuel density, use of the advanced M5 alloy cladding and the inclusion of mid-span mixing grids (MSMGs) in the Framatome-ANP fuel. These changes are within the scope of similar plant and fuel design changes that Dominion has successfully analyzed and implemented during prior operation of the North Anna and Surry plants. Such previous changes include:

- Surry Improved Fuel Implementation
- North Anna Vantage 5H Fuel Implementation
- North Anna and Surry ZIRLO Clad Material Implementation
- North Anna LTAs (Framatome-ANP Fuel)
- North Anna Steam Generator Replacement

The revised topical report describes the methods with which Dominion will perform the reload evaluation for the Westinghouse or Framatome-ANP fuel. Consistent with program milestones, results of transition analyses (including mixed core effects) associated with the use of Framatome-ANP fuel will be transmitted to the NRC by separate correspondence. This submittal is currently scheduled for March 2002.

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

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment

Commitments made in this letter: None

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ATTACHMENT 1

TOPICAL REPORT VEP-FRD-42, REV. 2

RELOAD NUCLEAR DESIGN METHODOLOGY

RELOAD NUCLEAR DESIGN METHODOLOGY

By

NUCLEAR ANALYSIS & FUEL STAFF

DOMINION RESOURCES SERVICES

RICHMOND, VIRGINIA

SEPTEMBER 2001

RELOAD NUCLEAR DESIGN METHODOLOGY

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PREFACE

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

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

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

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

G. L. Darden
J. O. Erb
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R. S. Margolis
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SECTION 1.0 - INTRODUCTION

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

This report presents the methodology employed by Dominion for performing a nuclear reload design analysis at the North Anna and Surry Power Stations. It covers analytical models and methods, reload nuclear design, reload safety analysis, and an overview of analyzed accidents and key parameter derivations. This revision also incorporates generic reference to approved methodologies that are applicable to core thermal-hydraulic analyses. The COLR section of the plant Technical Specifications provides a listing of such applicable methodologies. The generic citation of these methodologies is intended to minimize duplicate NRC Staff review effort, since review and approval of any such methodologies would precede their listing in the COLR.

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

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

SECTION 2.0 - ANALYTICAL MODELS AND METHODS

2.1 Analytical Models

The major analytical models currently used by Dominion for reload design and safety analysis are:

1. Virginia Power PDQ Two Zone model
2. Virginia Power NOMAD model
3. VEPCO RETRAN model
4. Core Thermal-Hydraulics models

The PDQ Discrete model was originally approved for reference in licensing applications by the NRC (Reference 1). The PDQ Two Zone model was subsequently developed to replace the PDQ Discrete model. A Topical Report for the PDQ Two Zone model (Reference 2) was provided to the NRC prior to implementation under the provisions of 10CFR50.59 (Reference 3). The NOMAD model was originally approved for reference in licensing applications by the NRC (Reference 4) and has subsequently undergone significant enhancements. The updated model was implemented under the provisions of 10CFR50.59 (Reference 5).

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

Use of the RETRAN Code and models, as originally approved for reference in licensing applications is documented in Reference 6. Supplemental details concerning the models used with the RETRAN Code were provided to the USNRC in an informational letter, Reference 7. These models were implemented under the provisions of 10CFR50.59. The applicable thermal-hydraulic codes and models are listed in the COLR section of the plant Technical Specifications. These analysis models have been used successfully to model plant transient response for core reloads, as well as various changes to plant configuration including core uprate and steam generator replacement.

2.1.1 Virginia Power PDQ Two Zone Model

The Virginia Power PDQ Two Zone Model performs three-dimensional geometry diffusion-depletion calculations for two neutron energy groups. The model uses the CELL2 code (Reference 8) and several auxiliary codes to generate and format the cross section input, perform shuffles, and other operations. The model employs a non-uniform mesh structure (25 X-Y mesh and 26 axial mesh) to represent each fuel assembly. Quarter core symmetric or full core geometry may be specified. The effects of non-uniform moderator density and fuel temperatures are accounted for with thermal-hydraulic feedback. More complete descriptions of the model and the auxiliary codes may be found in Reference 2.

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

2.1.2 Virginia Power NOMAD Model

The Virginia Power NOMAD Model performs one-dimensional axial diffusion-depletion calculations (with thermal-hydraulic feedback) for two neutron energy groups. The NOMAD Model makes use of data from the PDQ Two Zone model for two group cross sections and for normalization. The NOMAD model and its auxiliary codes are described in detail in Reference 5. The NOMAD model is used in the calculation of core average axial power distributions, axial offset, axial peaking factors, differential control rod bank worth, integral control rod worth as a function of bank position, fission product poison worth, and reactivity defects. 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 (FAC) analysis, and Relaxed Power Distribution Control (RPDC, Reference 9) may also be performed with the NOMAD model.

2.1.3 VEPCO RETRAN Models

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

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

2.1.4 Core Thermal-Hydraulics Models

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

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

2.2 Analytical Methods

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

2.2.1 Core Depletions

During the preliminary fuel loading and loading pattern search, depletions of the reload core are performed based on the low and high estimates of the end-of-cycle (EOC) burnup (the burnup window) for the previous cycle. The reload core loading pattern is depleted at hot full power

(HFP), all rods out (ARO) conditions, typically in quarter-core geometry. During the depletion, criticality is maintained by varying the boron concentration. These calculations provide relative power distributions, burnup predictions and an estimate of the cycle's full power capability.

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

2.2.2 Core Reactivity Parameters and Coefficients

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

2.2.2.1 Reactivity Coefficients and Defects

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

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

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

2.2.2.2 Differential Boron Worth

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

2.2.2.3 Delayed Neutron Data

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

2.2.2.4 Fission Product Poison Worths

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

2.2.3 Core Reactivity Control

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

2.2.3.2 Soluble Boron Concentrations

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

2.3 Analytical Model and Method Approval Processes

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

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

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

SECTION 3.0 - RELOAD DESIGN

3.1 Introduction

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

I. Core loading pattern design and optimization.

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

III. Design report, operator curve, and core follow predictions.

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

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

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

Physics design predictions for the support of station operations are calculated in Phase III using analysis techniques consistent with those of Phase II, except their calculation is performed on a best-estimate basis. These predictions are compared with measurements during startup physics testing and core follow to verify the design calculations, 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) as updated (UFSAR).
4. Specific operating limitations on the plant as contained in the Technical Specifications and COLR.

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

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

This review will establish or define:

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

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

3. Reload design schedules.

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

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

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

7. Expected plant operating conditions.

3.2.2 Fuel Loading and Pattern Determination

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

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

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

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

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

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

3.3 Nuclear Design Aspects of Reload Safety Analysis

3.3.1 Introduction

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

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

3.3.2 Safety Analysis Philosophy

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

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

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

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

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

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

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

Therefore, the overall process is as follows:

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

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

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

3.3.3 Non-Specific Key Parameters

Non-specific key parameters are derived by evaluating core characteristics for conditions bounding those expected to occur during the reload cycle to ensure that sufficiently limiting values of the parameter are determined. These conditions include conservative assumptions for such core parameters as xenon distributions, power level, control rod position, operating history, and burnup. These parameters are designated with '(NS)' in Table 2. Each non-specific key parameter generally serves as safety analysis input to several accidents including the accidents that also require specific key parameters, such as rod ejection. In addition, numerical uncertainty factors that are appropriate to the models being used are applied to the calculated parameter.

3.3.3.1 Rod Insertion Limits

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

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

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

When tentative RIL lines have been selected by the method just outlined, they are then checked to see that they satisfy all of the other evaluation requirements. If any basis is not satisfied by the tentative insertion limits, the insertion limits are raised until the most limiting basis is satisfied. These limits are then checked against the COLR. If these RIL lines exceed those in the COLR, the COLR is revised accordingly.

3.3.3.2 Shutdown Margin

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

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

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

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

3.3.3.3 Trip Reactivity Shape

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

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

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

3.3.3.4 Reactivity Coefficients

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

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

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

3.3.3.5 Neutron Data

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

The key analysis parameter is the β_{eff}, which is the product of the total β and the importance factor. The importance factor reflects the relative effectiveness of the delayed neutrons for causing fission. For some transients, it is conservative to use the minimum expected value of β_{eff}, while for others, the maximum expected value is more conservative. The use of conservatively large versus small β_{eff} values is treated on an event by event basis. β_{eff} is calculated at the times in cycle life that would produce the bounding values for the cycle (typically BOC and EOC).

The prompt neutron lifetime is the time from neutron generation to absorption. It is calculated by core averaging a region-wise power weighted prompt neutron lifetime calculated by a fuel lattice physics code for each region in the core. The key analysis parameter used for transients is the maximum prompt neutron lifetime, which is calculated at the limiting time in cycle life (typically EOC).

3.3.3.6 Power Density, Peaking Factors

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

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

3.3.4 Specific Key Parameters

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

3.3.4.1 Uncontrolled Control Rod Bank Withdrawal

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

The following assumptions and conservatisms are used:

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

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

3.3.4.2 Rod Misalignment

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

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

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

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

The dropped RCCA(s) event (dropped rod or dropped bank) is conservatively evaluated using the methodology described in WCAP-11394-P-A (Reference 15). Dominion acquired the transient databases and methodology information necessary to perform the dropped rod analyses of Reference 15 from Westinghouse. Dominion has performed evaluations which demonstrated the applicability of the methodology, the correlations, and the transient database for the analysis of the dropped rod event for the North Anna and Surry Power Stations. This methodology for the evaluation of the dropped rod(s) event has been implemented for both the North Anna and Surry Power Stations pursuant to the provisions of 10CFR50.59.

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

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

3.3.4.3 Rod Ejection

The rod ejection accident results from the postulated mechanical failure of a control rod mechanism pressure housing such that the coolant system pressure ejects the control rod and drive shaft to the fully withdrawn position. This results in rapid reactivity insertion and high peaking factors. Rod ejections are analyzed at the beginning and end of the cycle at hot zero power and hot full power.

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

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

The key parameters for the rod ejection accident are the ejected rod worth and total peaking factor, F_Q. The rod ejection key analysis parameters for the bounding power levels and burnups are derived for each reload core. The models used for the calculation of axial powers are depleted in such a way as to 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.

The rod ejection parameter derivation is performed in a conservative manner. Although the rod ejection accident is limited by Doppler feedback, the key analysis parameters are derived with all feedback frozen. Conservatism is ensured by calculating all physics parameters at steady state conditions using the "adiabatic assumption." This assumption asserts that any fuel damage which might occur during the transient takes place in a small time interval immediately following the ejection of the rod and before the thermal-hydraulic feedback effects of the core become important. This freezing of the core feedback effects leads to larger values of the total power peaking factor and ejected rod worth than would otherwise be expected in the transient.

3.3.4.4 Steamline Break

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

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

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

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

First, a detailed nuclear calculation is performed at the limiting time in cycle, HZP power conditions with all rods fully inserted, except the highest reactivity worth stuck rod. These conditions are conservative initial assumptions for steambreak (see References 10 and 11). Next, conditions including power, non-uniform inlet temperature distribution, pressure, and flow (derived from the RETRAN code output data at the point where the minimum DNBR may occur) are input, and peaking factors and axial power distributions are generated. The stuck rod is assumed to occur in the coldest quadrant to maximize reactivity insertion.

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

3.3.4.5 LOCA Peaking Factor Evaluation

A loss of coolant accident (LOCA) is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system. The LOCA reload evaluation methodology that is employed by Dominion is consistent with the fuel vendor methodology used for establishing and validating the operational limits for allowable core power distributions. A description of the reload validation methodology can be found in References 5, and 14.

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

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

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

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

3.3.4.6 Boron Dilution

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

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

3.3.4.7 Overpower Evaluations

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

3.3.5 Non-Nuclear Design Key Parameters

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

3.4 Reload Safety Evaluation Process

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

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

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

TABLE 1

EVALUATED ACCIDENTS

CONDITION II EVENTS

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

CONDITION III EVENTS

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

CONDITION IV EVENTS

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

TABLE 2

KEY ANALYSIS PARAMETERS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

4) Delayed neutron data and prompt neutron lifetime;

5) Relative power distributions at various core conditions;

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

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

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

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

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

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

SECTION 4.0 - SUMMARY AND CONCLUSIONS

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

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

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

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

Sometimes reanalysis will produce unsatisfactory results and other steps may have to be taken. Technical Specifications changes, COLR changes, or core loading pattern changes are typical adjustments that may be required. Raising the rod insertion limits, in order to reduce the ejected rod Fq and worth, is an example of a COLR change. If Technical Specifications changes are necessary to keep key parameters bounded, these changes must be approved by the NRC in accordance with 10CFR50.59 prior to implementation at the plant. In addition, loading pattern adjustments may be required to bring some key parameters within the current limits or reduce the size of the deviation.

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