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RELOAD NUCLEAR DESIGN METHODOLOGY

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RELOAD NUCLEAR DESIGN METHODOLOGY

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SECTION 1.0 - INTRODUCTION

Analytical methods used to insure the safety of Vepco nuclear plants after reload core and system changes will be discussed in this report. The topics covered, primarily from a nuclear design standpoint, will be the standard reload design, reload safety analysis, and an overview of analyzed accidents and key parameter derivations.

The standard reload design section details: (1) the 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; and (3) preparation of the cycle design report and related documents.

The reload safety analysis section discusses systematic ways of insuring the safety of the reactor after reload changes to the plant or core. The section indicates in general the analyses performed for the Surry and North Anna Units. However, due to differences in the units, a limited number of the analyses described do not pertain to all units. Each unit has specific license requirements that indicate exactly what analyses are necessary for the reload.

Key analysis parameters determine the severity of each accident. The accident analysis key parameter derivation section presents a conceptual discussion of all the accidents of concern for the FSAR or subsequent licensing submittals, and outlines the procedures

used to derive each core physics related key parameter. If all key parameters for the reload cycle are bounded by the values used in the reference safety analysis, the reference safety analysis applies for the reload core. When any reload parameter is not bounded by the value used in the reference analysis, safety analyses or evaluations must be performed for the affected transients.

This basic reload analysis philosophy has been used by Westinghouse (Reference 1) for all of the reload cores for the Vepco Surry Units 1 and 2, and North Anna Units 1 and 2. This philosophy will be used for future reload cycle designs by Vepco.

SECTION 2.0 - STANDARD RELOAD DESIGN

2.1 Introduction

This section describes the nuclear design effort performed by Vepco for a reload core. The design objectives for each of the three phases of the nuclear design are reviewed along with a description of the design codes used. The design procedures and methodology used for the preliminary and final design phases (i.e., core loading pattern design and optimization; and the design report predictions) are briefly described in this section. The remaining design phase, which is concerned with the determination of nuclear related key safety parameters, is considered in detail in Section 3 of this report.

The three nuclear design 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.

Section 2.2 below presents a summary of the nuclear reload design objectives, followed by sections on the design initialization process and the analytical models used for reload design. The

design methodology for Phases I and III are examined in detail in Section 2.5. Phase II design, key safety analysis parameter calculations, is described in Section 3.

2.2 Design Objectives

The overall objective in the design of a reload core is to determine the enrichment and number of new fuel assemblies and construct a core loading pattern which will fulfill the energy requirements for the cycle while satisfying the design basis and meeting all applicable safety analysis limits.

The objective of Phase I design is to produce a core loading pattern which meets the constraints outlined in the design initialization, (see Section 2.3). In addition, to be acceptable, the loading pattern must fulfill the following conditions:

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 do not exceed the Technical Specifications limits.

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

3. Sufficient rod worth is available to allow for shutdown with the most reactive rod in the core withdrawn.

The objective of Phase II design 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 input to the

safety analysis are verified by comparing the values of the parameters calculated for the reload cycle with the limiting values for these parameters assumed in the reference safety analyses. If a key analysis parameter for the reload cycle exceeds the limiting value, the corresponding transient must be reevaluated or reanalyzed.

Physics design predictions for the support of station operations are calculated in Phase III. The analysis techniques used in Phase III are consistent with those of Phase II. These predictions include reactivity parameters and coefficients, control rod worths, boron endpoints, core power distributions and core isotopics as a function of cycle burnup. The predictions are published in the form of a Nuclear Design Report for each reload cycle. In addition, PDQ INCORE decks are generated. These decks contain predicted power and flux distributions from PDQ07. The INCORE code (Reference 2) uses the PDQ07 predictions and thimble flux measurements to make predicted to measured power distribution comparisions. Using INCORE, design predictions are compared with measurements during startup physics testing and core follow to:

1. Verify the design calculations.

2. Insure that the core is properly loaded.

3. Verify that the core is operating properly.

2.3 Design Initialization

Before any nuclear design calculations are performed for a reload core, a design initialization must be 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. A design initialization includes the collection and review of design basis information to be used in initiating design work. It also insures 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 in the plant, the actual plant operating history, and any plant system changes projected for the next cycle.

The design basis information to be reviewed includes:

1. Reload cycle energy 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. Reload safety analysis parameters (mechanical, nuclear, and thermal/hydraulic) used in the safety analysis 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 cycle energy and operational requirements.

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.

2.4 Analytical Models

The major analytical models currently used in the reload design are:

1. the Vepco PDQ07 Discrete Model
2. the Vepco PDQ07 One-Zone Model
3. The Vepco FLAME Model

The Vepco PDQ07 Models perform two-dimensional (2-D,x-y) geometry diffusion-depletion calculations for two neutron energy groups. These models utilize the NULIF (Reference 3) code and several auxilliary 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 one-zone mesh). The discrete model generally has one mesh line per fuel pin, while the one-zone model has a mesh size of 6x6 per fuel assembly. A quarter-core symmetric two-dimensional geometry or a full core two-dimensional geometry may be specified for either model. Effects of nonuniform moderator density and fuel temperatures are accounted for by

thermal-hydraulic feedback. More complete descriptions of these models and their auxilliary codes will be found in References 4 and 5 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 Model.

The Vepco FLAME Model is used to perform three-dimensional (3-D, 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 in the FLAME Model.

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 auxilliary codes will be found in Reference 6. 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.

Specifics on the use of the analytical codes in key analysis paramater generation are discussed later in this report. Additional support codes are used in conjunction with the PDQ07 and FLAME Models to perform special calculations such as xenon and samarium worths.

Numerical uncertainty factors appropriate to the model used and to the calculation performed are applied to the key analysis parameter determinations. These uncertainty factors will be detailed in a forthcoming Vepco topical report. The PDQ07 and FLAME Models will be referred to generically in the remainder of this report (i.e., as the 2-D and the 3-D models respectively).

2.5 Analytical Methods

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

2.5.1 Core Depletions

Each reload core loading pattern is depleted at hot full power (HFP), all rods out (ARO) conditions using a 2-D model in

eighth-core or quarter-core geometry. Criticality is maintained by varying the boron concentration. The calculations provide x-y relative power distributions, burnup predictions and an estimate of the end of cycle (EOC) reactivity. During Phase I design, a depletion of the reload core is performed based on a nominal, (i.e. best estimate), EOC for the previous cycle. Additional depletions are performed for an EOC burnup window for the previous cycle (typically +/- 25 effective full power days (EFPD) about the nominal EOC burnup). These additional depletions allow the sensitivity of the predicted reload cycle parameters to be examined as a function of the previous EOC burnup. The majority of design predictions will be based on the nominal previous EOC burnup. However, the PDQ-INCORE decks, predictions of assembly average burnups and the HFP, ARO boron letdown curve are calculated based on a previous EOC burnup that is within +/- 2 EFPD of the actual burnup.

2.5.2 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 or fuel temperature or core power level. The reactivity coefficients and parameters are calculated on a corewise basis and are evaluated at a representative range of core conditions at the beginning, middle and end of the reload cycle. These conditions include zero power,

part power and full power operation, with various rodded core configurations, with equilibrium xenon or no xenon. A description of each type of calculation follows.

2.5.2.1 Temperature Coefficients

The isothermal temperature coefficient is defined as the change in reactivity per degree change in the moderator, clad, 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 Doppler temperature coefficient is defined as the change in reactivity per degree change in the fuel and clad temperatures.

The moderator temperature coefficient 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.

Temperature coefficients are calculated with a 2-D model. The change in reactivity is determined due to a change in the appropriate core temperature parameter(s), (e.g., the moderator temperature or fuel temperature), with all other conditions in the

core being maintained at a constant value.

2.5.2.2 Differential Boron Worth

The differential boron worth, sometimes referred to as the boron coefficient, 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, all other core parameters being held at a constant value.

2.5.2.3 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 for each group by the core integrated fission rate of that isotope.

2.5.2.4 Power Coefficients and Defects

The total power coefficient is defined as the combined effect on the core reactivity of moderator and fuel temperature changes brought about by core power level changes. The Doppler "only" power coefficient relates to the change in power which produces a change in the fuel and clad temperature. The power defect is the integral

of the power coefficient over the appropriate power range, usually zero to full power. Power coefficients are calculated using a 2-D model. The total power coefficient is found by noting the change in core average reactivity with core power level. The Doppler "only" power coefficient is calculated by noting the change in core average reactivity due to a change in core power level, but with the moderator temperature maintained at a constant value.

2.5.2.5 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 NULIF and the 2-D model.

2.5.3 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 control banks and shutdown banks. The control banks can be used to compensate for reactivity changes associated with changes in operating conditions such as temperature and power level. 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.

2.5.3.1 Most Reactive Rod Stuck (MRRS)

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 control rod trip. In calculating the shutdown margin it is conservative to reduce the total rod worth by the amount of the most reactive stuck rod. Calculation of the MRRS worth is usually performed at both HZP and cold zero power (CZP) core conditions with a 2-D model. The MRRS worth is found by noting the change in reactivity between a core configuration with all rods inserted (ARI) and a core configuration with ARI less the MRRS, all other core conditions remaining constant.

2.5.3.2 Integral and Differential Rod Worths

Integral rod worths are calculated with a 2-D model using a method similar to that described above for the MRRS prediction. Differential rod worths are calculated using a 3-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.

2.5.3.3 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 the boron controls 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 are performed with a 2-D model.

2.5.3.4 Rod Insertion Limits

Rod insertion limits (RIL) are required to maintain an acceptable power distribution during normal operation, and acceptable consequences following a postulated rod ejection accident, and also insure that the minimum shutdown margin (SDM) assumed in the safety analyses is available.

The rod insertion limits are lines (drawn on a curve showing rod insertion versus power level) which show the deepest allowed insertion of the control rods at any given power level. The rod insertion allowance (RIA) is the amount of control bank reactivity which is allowed to be in the core at HFP. The rod insertion limits are primarily a guarantee that the RIA is not exceeded at any power level. Additional amounts of control bank reactivity can be allowed to be in the core at lower power levels if SDM is preserved.

The relationship between the RIA and the RIL is such that RIL lines determined purely from RIA considerations are usually shallow enough that the other considerations (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 3-D model simulation of the control banks moving into the core with normal overlap while ascertaining that at least 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 (D Bank partly inserted) that the burnup and xenon distribution force the power to the top of the core. This guarantees that the calculated RIL lines are conservative from a reactivity insertion standpoint because they predict more reactivity insertion at rod insertions that are more shallow than would normally be seen in the actual core.

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. They would then become the final rod insertion limits.

SECTION 3.0 - NUCLEAR DESIGN ASPECTS OF RELOAD SAFETY ANALYSIS

3.1 Introduction

This section discusses the derivation of the core physics related 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 3.2 and 3.3. 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 is given in Section 3.4.

3.2 Safety Analysis Philosophy

The Vepco safety analysis philosophy is: (1) to perform at an early stage in plant design, a bounding conservative safety analysis and (2) to systematically determine whether or not reanalysis is needed when changes occur in the core or plant systems. This approach is taken to minimize the amount of reanalysis which must be performed as a result of plant reload changes.

Nuclear power plants are licensed to operate by the NRC. To receive and retain an operating license, 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, 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 Vepco accident analysis is typical in that the complete FSAR analysis was performed by the NSSS vendor. However, Vepco has verified the key Condition I, II, and III FSAR analyses and the safety of its plants using its own analysis capability (Reference 7).

Accident analyses must show that the reactor is operated under conditions that assure complete public protection from hazard in almost all accident cases. The four categories of accidents based on their anticipated frequency of occurance and potential for public harm are described in References 8 and 9. 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 accident analysis, which yields the accident results as a function of these key analysis parameter values. The key analysis parameters are derived primarily from steady state diffusion theory calculations. 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.

In the FSAR stage, the analyses proceed by first determining 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 are performed and the key parameters are determined for several cycles of operation in order to obtain a set of key parameters which has a high probability of being bounding over plant life. These bounding key parameters are called the (initial) current limits. Accident analyses are 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 new undepleted 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 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 to determine if a new accident analysis will be required. If an accident reanalysis is performed, that key analysis parameter then becomes the bounding value and is called the current limit in subsequent cycles.

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) Make a decision on whether to reanalyze accidents based on the effect of the new key analysis parameters.

3.3 Safety Analysis Administration

The groups responsible for reload core safety analysis at Vepco are

the Nuclear Design Group and the Safety Analysis Group. These are presently organized as branches of the Nuclear Fuel Engineering (NFE) Subsection of the Fuel Resources Department (FRD). The roles of these two groups in providing systematic and inclusive safety analysis coverage is outlined in this section.

The first step in the reload safety analysis of a core is the preparation of a listing of the physics design calculations to be performed in support of the safety analysis. This list is prepared by the Safety Analysis Group of NFE and forwarded to the Nuclear Design Group of NFE. The Nuclear Design Group performs the designated calculations (generally static nuclear calculations) based on this list. The Safety Analysis 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 NFE documenting the results of the safety analysis for the reload cycle.

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 NFE. This report is used by the Nuclear Fuel Operation (NFO) Group of the Fuel Resources Department in the preparation of operator curves for use by station personnel in the operation of the cycle.

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 prediction are published by NFO as a Startup Physics Test Report and a Core Performance Report for each reload cycle.

Figure 1 presents a summary of the documentation and information flow of the safety analysis administration for a reload cycle.

3.4 Overview of Accidents and Key Parameter Derivations

The accidents analyzed for the FSAR and evaluated at each reload cycle are listed in Table 1. The key parameters to be determined at each reload cycle are listed in Table 2. There are three types of key parameters. Two types of parameters are generated by the Nuclear Design Group. The non-specific parameters (designated "NS" in Table 2) are generated by evaluating core characteristics at conservative conditions, and the specific parameters (designated "S" in Table 2) are generated by statically simulating an accident. The third type of key parameter is not generated by the Nuclear Design Group (designated "NND" in Table 2).

The methods which will be employed by Vepco to determine these key parameters will be consistent with the methods used for past cycles of the Surry and North Anna units by the fuel vendor (Westinghouse). These methods have been documented in Reference 1.

3.4.1 Non-Specific Key Parameters

Non-specific key parameters are key parameters derived by evaluating core characteristics at conservative conditions. The conservative conditions assure that the limiting values of the parameter are determined. 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.

3.4.1.1 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 (4% delta-Keff is a typical amount) of trip worth is assumed to be available. This section will discuss the derivation of the reactivity insertion versus rod insertion curve, also called the trip reactivity shape.

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 3-D model. The model

is depleted with all rods out at hot full power, equilibrium xenon to the end of cycle (EOC). The calculation is performed at the depletion step (time in life) which has the most bottom peaked power (usually EOC). The D bank is inserted to push the axial offset to its negative Technical Specifications limit. The trip bank of rods is inserted in several discrete steps and the Keff at each step noted. Thus, trip reactivity versus rod insertion (trip reactivity shape) is determined.

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 as or more reactivity insertion than the current limit shape (which could be the FSAR shape) for the rod insertion, it is bounded by the 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 slight 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. The current limit reactivity shape is not changed. But, if the reload shape is clearly more conservative than the current limit shape, the transients affected by trip reactivity shape are reanalyzed using the reload shape which then becomes the new current limit.

3.4.1.2 Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedbacks, in particular the moderator temperature coefficient and the Doppler power coefficient. The reactivity coefficients and their generation for the standard reload design were 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 maximums and minimums 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. The justification for the use of conservatively large versus small reactivity coefficient values is treated on an event by event basis in the safety analysis.

3.4.1.3 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; for others, the maximum expected value is more conservative. The justification for the use of conservatively large versus small Beta-effective values is treated on an event by event basis in the safety analysis.

Since the maximum Beta-effective occurs at the beginning of the reload cycle, and the minimum Beta-effective occurs at the end, Beta-effective is calculated at the beginning and the 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 a core average parameter calculated with the cross section generating code. The key analysis parameter used for transients is the maximum prompt neutron lifetime. The maximum

occurs at the end of a reload cycle.

Numerical uncertainty factors, appropriate for the codes used, are applied to the Beta-effective and prompt neutron lifetime to conservatively increase or decrease those parameters, as appropriate.

3.4.1.4 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 radial peaking factor increases with decreasing power level and with rod insertion. For transients which may be overpower limited, the total peaking factor is of importance. 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 times Power = design hot spot heat flux.

3.4.2 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 can be shown to be the most conservative conditions for the accident. For example, in the rod ejection accident the post-ejected rod condition (rod fully withdrawn) can be shown to be more conservative (i.e., gives higher peaking factors and rod worths) than the pre-ejected rod condition or any point between the "pre" and "post" ejected conditions. This is shown by performing static model calculations before the rod ejection and at some of the intermediate configurations, and noting that the peaking factors and rod worths from these calculations are smaller than the post ejection configuration simulation. For all of the specific key parameter derivations, similar arguments can be made for the conservatism of the selected static conditions.

3.4.2.1 Uncontrolled Rod Bank Withdrawal

The rod withdrawal accident occurs when the two control rod banks having the maximum combined worth are withdrawn from the core due to some control system malfunction. A reactivity insertion results. The accident can occur at HZP or HFP. A 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 = 100000. times delta-Keff divided by Keff).

This parameter is derived by calculating the maximum differential rod worth for two sequential highest worth control banks. The following assumptions are also made:

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

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

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 the maximum worth sequential banks 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) The application to the peak differential worth of a numerical uncertainty factor which is appropriate to the model being used.

3.4.2.2 Rod Misalignment

Rod misalignment accidents result from the malfunctioning of the control rod positioning mechanisms. Rod misalignment accidents include: 1) static misalignment of a control bank, 2) dropped RCCA (Rod Cluster Control Assembly, i.e., a control rod), and 3) 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 the resultant radial peaking factors (FdH). These peaking factors are determined using 2-D and 3-D models. The maximum FdH peaking factors calculated for each of these types of rod misalignments are given to the Safety Analysis group for evaluation.

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 this accident, a full core 3-D calculation with D Bank in to its respective insertion limits at several power levels is made. Sequential banks are assumed to be inserted with the appropriate overlap. A series of calculations is made with the worst (the one that causes the highest FdH peaking factor) D Bank rod fully

withdrawn over the entire power range.

The dropped RCCA accident is simulated with a full core 3-D model calculation. The dropped bank accident is simulated with a quarter core 3-D model calculation.

The key analysis parameters for rod misalignment accidents are the radial peaking factors. 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. The appropriate 2-D discrete mesh calculations are made to correct the 3-D coarse mesh results. Uncertainty factors appropriate to the models used are applied.

3.4.2.3 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 that take place at the beginning of the cycle at hot zero power and hot full power, and at the end of cycle at hot zero power and hot full power are assumed to bound all other burnups and power levels.

The key parameters are ejected rod worth and total peaking factor

(Fq). From an information flow point of view the key parameters are generated by the Nuclear Design Group using steady state neutron diffusion theory or nodal methods and transmitted to the Safety Analysis Group to be analyzed using kinetics methods. The rod ejection key analysis parameters for the bounding power levels and burnups must be derived for each initial and reload core.

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 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.

The detailed procedures for producing the rod ejection key analysis parameters are analytical simulations of the above scenario. The 3-D and 2-D computer models are used.

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 will result in a smaller Fq and ejected rod worth; therefore, deriving them without feedback is conservative.

Another conservatism is that the 3-D model is 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. Less burnup at the top of the the core insures higher worths and peaking factors, for both HZP and HFP, as compared to the best estimate axial burnup shape.

In addition, numerical uncertainty factors appropriate to the models used are applied to the calculated ejected rod worth and highest Fq.

3.4.2.4 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. And for the hypothetical break, DNB or clad damage may occur, but the safety analysis must show that the 10CFR100 limits are not exceeded.

The starting point for both analyses is a reference safety analysis using a system transient code with kinetics capability (Reference 7). The input parameters for the system transient code include nuclear parameters which are considered conservative for the reload core being analyzed. This system transient analysis code 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 an end of cycle, hot zero power condition with all rods fully inserted, except the highest reactivity worth stuck rod. These conditions are conservative assumptions for steambreak (see References 8 and 9). A non-uniform inlet temperature distribution derived from the system transient code loop temperature data is input to the 3-D model.

The point in the system transient analysis code which is appropriate for the minimum DNBR is analyzed. The temperature and pressure information from the system transient calculation and peaking factor information from the detailed nuclear calculation are input to a thermal/hydraulic analysis code (see Reference 10) to accurately determine the minimum DNBR for the steambreak transient.

3.4.2.5 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 will be employed by Vepco is consistent with the methodology used for past cycles of the Surry and North Anna Units by the fuel vendor (Westinghouse). A description of this methodology is found in References 1 and 11.

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 1, LOCA analyses assume that the reactor is operating in such a manner that the peak linear heat generation rate in the core is maximized. The most limiting power shape is also assumed. The limiting Fq times relative power versus core height envelope is conservative with

respect to the limiting cosine and top peaked power shapes assumed for large and small break LOCA analyses respectively.

The second key parameter is derived by the Nuclear Design Group, and involves determining what the highest core Fq will be at each axial point (along the core height) as the core operates throughout the reload cycle. The reload cycle depletion is simulated using 3-D modeling as described in Reference 11. Base load and load follow depletion schemes are used in which the control rods are moved and the power level varied in ways that are typical of the way that the plant would be or could be operated.

Beginning, middle, and end of cycle conditions are included in the calculations. Different histories of operation are assumed prior to calculating the effects of load follow transients on the axial power distribution. These different histories assumed either base loaded operation or extensive load following. The following Technical Specification requirements are observed during the load follow maneuvers:

(1) Control rods in a single bank move together with no individual rod insertion differing by more than a pre-specified number of steps from the bank demand position.

(2) The full length control bank insertion limits are not violated.

(3) The recommended axial power distribution control procedures,

which are given in terms of flux difference (power at top of core minus power at bottom of core) control are observed.

The recommended axial power distribution control procedures require the control of the core axial offset (flux difference/fractional power) at all power levels within a permissible operating band about a target value that is related to the equilibrium full power value. Controlling the axial offset in this manner minimizes xenon transient effects on the axial power distribution.

To insure that the analysis is comprehensive, the load follow maneuvers are simulated using a number of power demand schedules along with the limiting (bounding) permitted variations in the control strategy (i.e., control to the center and control to the edge of the permitted band of axial offsets).

To insure that every operational mode is covered, a complete set of shallow, medium, and steep power demand schedules are used with the center of the control band strategy and with the edge of the control band strategy, in beginning, middle, and end of life load follow maneuvers. The end of life maneuvers follow both base load and load follow depletion histories. For all of the depletions performed, and at each axial position, the magnitude of every Fq times relative power is compared. The highest Fq times relative power at each axial position is thereby determined for all of the depletions performed. Numerical uncertainty factors appropriate for the models used are applied to the limiting Fq points for

conservatism.

If all of the combinations discussed in the previous paragraph are performed the result is 18 unique depletion cases. However, calculational studies already performed using this methodology have shown that for the great majority of reload cycles a 3 case subset of the 18 unique depletion cases produces the limiting Fq points. In the few instances in which this 3 case subset did not produce the limiting points, the points that were limiting were never more than a certain percentage higher than those from the 3 case subset (see Reference 1, Section 3.3.3.1 in particular).

Therefore, the methodology will be to perform the 3 case subset and multiply the Fq points obtained by a conservative uncertainty factor based on that limiting percentage. If the points fall below the Fq envelope, this phase of the analysis is complete. If some points fall on or above the Fq envelope, the remaining 15 cases will be performed and the limiting Fq points from all 18 cases (the extra conservatism is now removed from the initial 3 case subset, and not applied to the remaining 15) will be tabulated and compared to the envelope. If the 18 cases are bounded by the envelope, this phase of the analysis is complete.

The location of all limiting Fq points below the Fq envelope means that for the reload cycle being analyzed all normal base load and load follow operations can be performed without producing power distributions more limiting than those assumed in the current LOCA

analysis. If Fq points still remain on or above the Fq envelope, plant operational adjustments have to be made, the nature of which depend on the magnitude of the violations. A typical adjustment would be power distribution surveillance above certain power levels to insure that actual power distributions experienced in the plant are less than the LOCA envelope.

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, and (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.

3.4.2.6 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; i.e., boron dilution during refueling and startup, cold and hot shutdown, and at power conditions.

For each reload, 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 and the Cycle Design Report. But, 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.

In addition, the change in boron concentration to make the core critical from a minimum shutdown margin initial condition must be determined for each phase of plant operation. The 2-D model is also used for these determinations.

3.4.2.7 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 is described in Reference 11 (Section 6-2 in particular for rod withdrawal and Section 6-3 in particular for boron dilution).

3.4.3 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 1 (Section 4.3 in particular).

SECTION 4.0 - SUMMARY AND CONCLUSIONS

Designing a core that meets all safety criteria is sometimes an iterative process involving interaction and trade-offs between the Nuclear Design and the Safety Analysis Groups. For the typical reload, the derived key analysis parameters will be 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 parameter which exceeded its limit becomes the new current limit and the reanalysis becomes part of the reference analysis.

Sometimes reanalysis with out of bounds parameters 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. Of course, if any Technical Specifications changes are necessary to keep key parameters bounded, these changes must be approved by the NRC before they can be used in the plant. Loading pattern adjustments may be required to bring some key parameters within the current limits or reduce the size of the deviation.

Interaction between the Nuclear Design and Safety Analysis Groups allows the development for each reload cycle of a safety evaluation strategy which best suits that particular cycle.

SECTION 5.0 - REFERENCES

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FIGURE 1

SAFETY ANALYSIS ADMINISTRATION FOR A RELOAD CYCLE

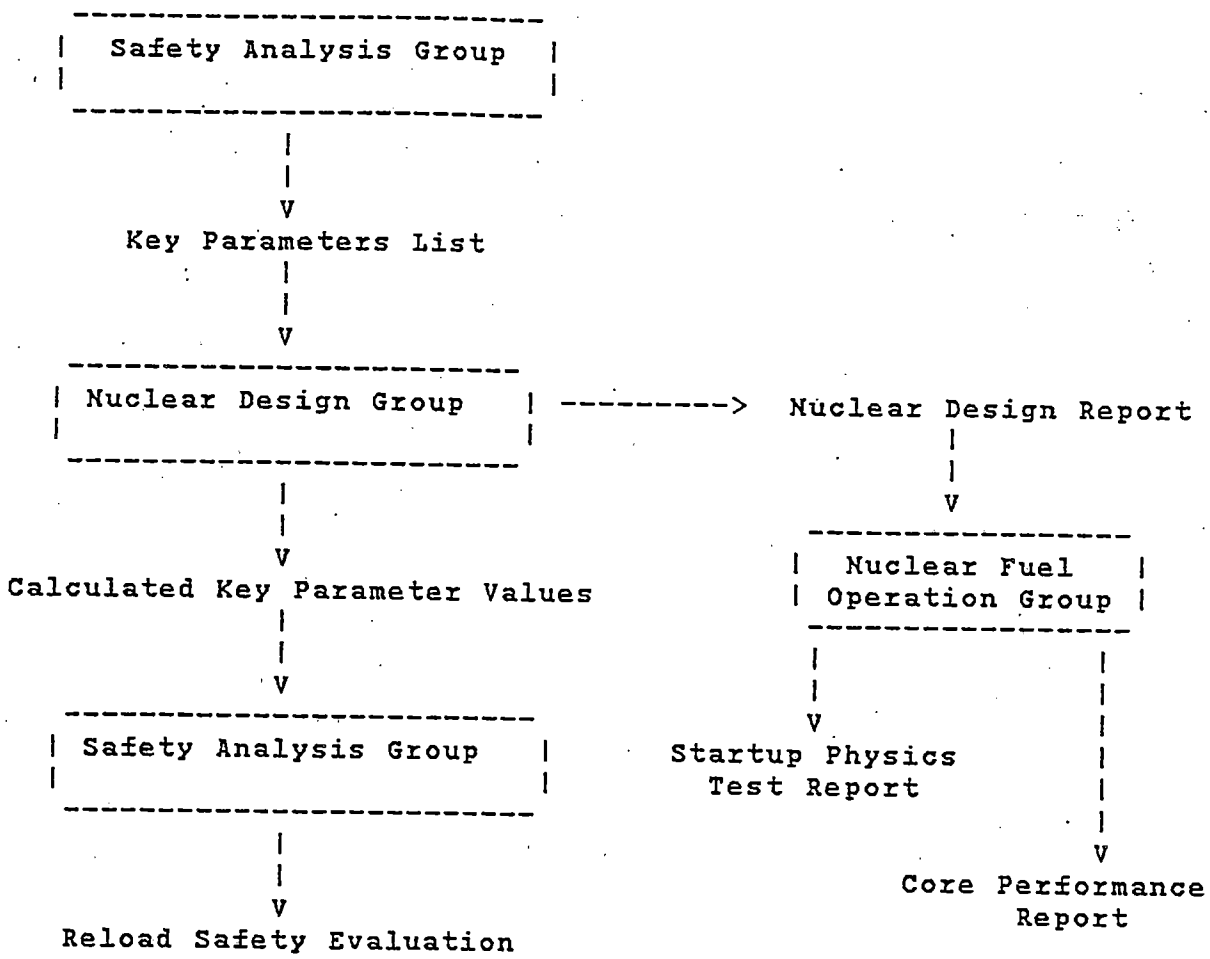

TABLE 1

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

TABLE 1 (CONT.)

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

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 (NND)
2) Moderator Temperature 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) Initial Boron Concentration for Required Shutdown Margin (NS)
13) Reactivity Insertion Rate (S)
14) Trip Reactivity Shape (NS)
15) Power Peaking Factor (S)

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

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

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

NND: Non-Nuclear Design
NS: Non-Specific
S: Specific

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25) Fuel Rod Internal Gas Pressure (NND)

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NND: Non-Nuclear Design
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