Reload Nuclear Design Methodology

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VIRGINIA POWER NORTH CAROLINA POWER
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RELOAD NUCLEAR DESIGN METHODOLOGY

BY

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TABLE OF CONTENTS

Page

LIST OF TABLES

LIST OF FIGURES

SECTION 1.0 - INTRODUCTION

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

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

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

SECTION 2.0 ANALYTICAL MODELS AND METHODS

2.1 ANALYTICAL MODELS

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

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

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

2.1.1 Virginia Power PDQ07 Models

The Virginia Power PDQ07 Discrete and One-Zone Models perform two-dimensional (x-y) geometry diffusion-depletion calculations for two neutron energy groups. These models utilize the NULIF (Reference 7) code and several auxiliary codes to generate and format the cross section input, perform shuffles, and other operations. The two models are differentiated according to their mesh size (i.e., either a discrete mesh or coarse mesh). The Discrete model utilizes one mesh block per fuel pin, while the One-Zone model has 6x6 mesh blocks per fuel assembly. An eighth,

quarter, or half core symmetric two-dimensional geometry or a full core two-dimensional geometry may be specified for either model. The effects of nonuniform moderator density and fuel temperatures are accounted for with thermal-hydraulic feedback. More complete descriptions of these models and their auxiliary codes may be found in References 1 and 2 for the Discrete and One-Zone models, respectively.

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

2.1.2 Virginia Power FLAME Model

The Virginia Power FLAME Model is used to perform three-dimensional (x-y-z geometry) nodal power density and core reactivity calculations using modified diffusion theory with one neutron energy group. The model utilizes the NULIF code and several

auxiliary codes to generate and format cross section input, perform shuffles, and other operations. Each fuel assembly in the core is represented by one radial node and 32 axial nodes. As with the PD207 Models, the effects of nonuniform moderator density and fuel temperature are accounted for by thermal-hydraulic feedback. A more complete description of this model and its auxiliary codes may be found in Reference 3. The FLAME Model is used in calculating and evaluating three-dimensional or axial effects such as differential rod worths, axial power and burnup distributions, and control rod operational limits. FLAME Model predictions are normalized to those of the PD207 model when applicable.

2.1.3 Virginia Power NOMAD Model

The Virginia Power NOMAD Model performs one-dimensional (z) geometry, diffusion-depletion calculations (with thermal-hydraulic feedback) for two neutron energy groups. The NOMAD model makes use of data from the PD207 Discrete, PD207 One-Zone, and FLAME models for normalization. As in the FLAME model the active fuel length is represented by 32 axial nodes. The NOMAD model and its auxiliary codes are described in detail in Reference 4. The NOMAD model is used in the calculation of core average axial power distributions, axial offset, axial peaking factors, differential control rod bank worths, and integral control rod worths as a function of bank position. In addition, NOMAD has the capability to perform criticality searches on boron concentration, control rod position, core power level, and inlet enthalpy. Simulation of load follow

maneuvers, performance of Final Acceptance Criteria analysis, and Relaxed Power Distribution Control (RPDC, Reference 9) may also be performed with the NOMAD model.

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

2.1.4 Virginia Power RETRAN Models

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

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

2.1.5 Virginia Power COBRA Models

The Virginia Power COBRA models are used to perform a detailed thermal-hydraulic analysis of the reactor core. Details of this

model are described in Reference 6. COBRA solves the governing conservation and state equations to resolve the flow and energy fields within the reactor core geometry. These results are used in turn to calculate the departure from nucleate boiling ratio (DNBR) with the W-3 CHF correlation. COBRA can perform either steady state or DNBR calculations or transient DNBR analyses with forcing function which have been supplied by the RETRAN code. Steady state applications include thermal limit generation, DNBR statepoint analyses and axial shape verification for RPDC. Examples of transient applications are loss of flow and locked rotor DNBR analysis.

2.2 ANALYTICAL METHODS

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

2.2.1 Core Depletions

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

For the safety evaluation of a reload loading pattern, additional
depletions using the 1D, 2D, and 3D models are performed to bound
the EOC burnup window for the previous cycle which is typically +/-
30 effective full power days (EFPD) about the nominal EOC burnup.
These window depletions allow the sensitivity of the predicted
reload cycle parameters to be examined as a function of the
previous EOC burnup.

The calculation of reload design parameters required for startup physics testing and core follow must be made as near to the actual operating conditions of the reload as possible. To ensure this, those predictions dependent on burnup are calculated based on a previous EOC burnup that is within +/- 2 EFPD of the actual burnup.

2.2.2 Core Reactivity Parameters and Coefficients

The kinetic characteristics of the core are described by the core reactivity parameters and coefficients. These parameters and coefficients quantify the changes in core reactivity due to varying plant conditions such as changes in the moderator tempefature, fuel temperature, or core power level. The reactivity coefficients and parameters are calculated on a corewise basis using a 2-D model for a representative range of core conditions at the beginning, middle and end of the reload cycle. These include zero power, part power, and full power operation; at various rodded core configurations; and for equilibrium xenon or no xenon conditions. These parameters are used as input to the safety analysis for modeling the reactor's response during accidents and transients. In addition, they may be used to calculate reactivity defects (integral of the coefficient over a specific range of temperature or power) to determine the reactor's response to a change in temperature or power. A description of each type of calculation follows.

2.2.2.1 Temperature and Power Coefficients

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

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

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

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

only to the fuel temperature changes brought about by core power
level changes. The power defect is the integral of the power
coefficient over the appropriate power range, usually zero to full
power.

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

Coefficients at HZP

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

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

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

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

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

Coefficients at Power

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

To calculate the Doppler temperature coefficient for at-power conditions two calculations are needed. These calculations adjust the fuel temperatures to values +/-5°F about the reference value by adjusting the power above and below the reference power while

adjusting the moderator inlet enthalpy to keep the core average moderator temperature constant at the reference value. The at-power Doppler temperature coefficient can now be calculated using the same formula as the HZP Doppler temperature coefficient.

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

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

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

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

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

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

2.2.2.2 Differential Boron Worth

The differential boron worth is defined as the change in reactivity due to a unit change in boron concentration. Differential boron worths are calculated with a 2-D model by noting the change in core average reactivity due to a change in the corewise boron concentration, (normally +/-20 ppm about the target value), with all other core parameters being held constant.

2.2.2.3 Delayed Neutron Data

Delayed neutron data are used in evaluating the dynamic response of the core. The delayed neutrons are emitted from precursor fission products a short time after the fission event. The delayed neutron fraction and decay constant for six delayed neutron groups at various core conditions are calculated using a 2-D model, and are found by weighting the delayed neutron fraction for each fissionable isotope in each group by the core integrated fission rate of that isotope.

2.2.2.4 Xenon and Samarium Worths

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

are determined using information from the 2-D model.

2.2.3 Core Reactivity Control

Relatively rapid reactivity variations in the core are controlled by the full length control rods. The full length control rods are divided into four control banks (designated D, C, B, and A) and two shutdown banks (designated SB, and SA). The control banks D, C, B, and A are used to compensate for core reactivity changes associated with changes in operating conditions such as temperature and power level and are moved in a fixed sequential pattern to control the reactor over the power range of operation. The shutdown banks are used to provide shutdown reactivity.

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

2.2.3.1 Integral and Differential Rod Worths

Integral rod worths are calculated with a 2-D model by determining the change in reactivity due to the control rod being out of the core versus being inserted into the core with all other conditions being held constant. Differential and integral rod worths as a function of axial position are calculated using a 3-D or 1-D model. The change in core average reactivity is evaluated as a function of the axial position of the rod or rods in the core to obtain the

differential rod worth.

2.2.3.2 Soluble Boron Concentrations

Boron in the form of boric acid is used as the soluble absorber in the reactor coolant. At no load, the reactivity change from CZP to HZP is controlled by changing the soluble boron concentration. At HFP, soluble boron is used to compensate for the reactivity changes caused by variations in the concentration of xenon, samarium and other fission product poisons, the depletion of uranium and the buildup of plutonium, and the depletion of burnable poisons. Predictions of the soluble boron concentration necessary to maintain criticality or subcriticality are performed with a 2-D model.

SECTION 3.0 - RELOAD DESIGN

3.1 INTRODUCTION

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

    I.   Core loading pattern design and optimization.

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

    III.   Design report predictions.

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

The objective of Phase I design is to produce a core loading pattern which meets the constraints outlined in the design initialization, (see Section 3.2.1). In addition, some preliminary Phase II calculations are performed to verify that conditions on radial peaking factors, moderator temperature coefficient, and shutdown margin are met.

The objective of Phase II of the design process is to verify that all core physics related limits are met for the core loading

pattern.   Once the final loading pattern for the reload cycle has been optimized under Phase I, the core physics related key analysis parameters for the reload cycle are verified to determine if they are bounded by the limiting values for these parameters assumed in the reference safety analyses.   These Phase II parameters are calculated using a "worst case" assumption philosophy to ensure the results are conservative for the reload.   If a key analysis parameter for the reload cycle exceeds the limiting value, the corresponding transient must be evaluated or reanalyzed using the reload value. Should the reload value cause a violation in the safety criteria, a new reload design or possibly new operating limits (Technical Specifications) may have to be instituted.

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

3.2 CORE LOADING PATTERN DESIGN AND OPTIMIZATION

3.2.1 Design Initialization

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

The design basis information to be reviewed includes:

1. Unit operational requirements.

2. Applicable core design parameter data.

3. Safety criteria and related constraints on fuel and core components as specified in the Final Safety Analysis Report (FSAR).

4. Specific operating limitations on the plant as contained in the Technical Specifications.

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

6. Reload safety analysis parameters (mechanical, nuclear, and thermal/hydraulic) used in the safety analyses up to and including the previous cycle.

This review will establish or define:

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

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

3. Reload design schedules.

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

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

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

7. Expected plant operating conditions.

3.2.2 Fuel Loading and Pattern Determination

The determination of the fuel loading consists of finding a combination of enrichment and number of fresh fuel assemblies which meets the reload cycle energy and operational requirements established during the design initialization. Based on prior experience an enrichment and number of feed assemblies are chosen. These assemblies along with the assemblies to be reinserted will be arranged in a preliminary loading pattern. Using a 2-D model this loading pattern will be modeled and depleted to determine the cycle's energy output and radial power distributions. This is repeated with different numbers of feed assemblies and/or enrichments until the cycle energy requirements are met. During this time, shuffling of the assemblies to different locations to

improve the power distribution may also be performed. Once a fuel

loading is determined the rearrangement of the fuel assemblies

continues until the following conditions are met.

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

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

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

When a pattern meets the above conditions, the enrichment and

number of fresh assemblies along with any burnable poison

requirements are set.   At this point, the loading pattern is

optimized for cycle length and power distribution by shuffling the

fuel and/or burnable poison.   Once the optimum pattern has been

established it is evaluated and analyzed to determine whether all

core physics related limits can be met during the operation of the

unit.

3.3 NUCLEAR DESIGN ASPECTS OF RELOAD SAFETY ANALYSIS

3.3.1 Introduction

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

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

3.3.2 Safety Analysis Philosophy

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

Plant safety is demonstrated by accident analysis, which is the study of nuclear reactor behavior under accident conditions. Accident analyses are usually performed in the initial design

stages   and   documented   in   the   FSAR.   The   Virginia   Power   accident
analysis   is   typical   in   that   the   complete   FSAR   analysis   was
performed   by   the   NSSS   vendor.   However,   Virginia   Power   has   verified
the   key   Condition   I,   II,   III,   and   IV   FSAR   analyses   (excluding   LOCA)
and   the   safety   of   its   plants   using   its   own   analysis   capability
(References   5   and   13).   The   four   categories   of   accidents   based   on
their   anticipated   frequency   of   occurrence   and   potential   for   public
harm   are   described   in   References   10   and   11.   The   accident   analyses
consider   all   aspects   of   the   plant   and   core   including   the   operating
procedures   and   limits   on   controllable   plant   parameters   (Technical
Specifications)   and   the   engineered   safety,   shutdown,   and
containment   systems.

There   are   two   stages   in   the   analysis   process.   First,   steady   state
nuclear   calculations   are   made   for   the   conditions   assumed   in   the
accident   analysis.   The   nuclear   parameters   derived   from   these
calculations   are   called   the   core   physics   related   key   analysis
parameters   and   serve   as   input   to   the   second   stage.   The   second
stage   is   the   actual   dynamic   accident   analysis,   which   yields   the
accident   results   as   a   function   of   these   key   analysis   parameter
values.   The   accident   analyses   are   transient   calculations   which
usually   model   the   core   nuclear   kinetics   and   those   parts   of   the
plant   systems   which   have   a   significant   impact   on   the   events   under
consideration.

During   the   original   FSAR   analysis,   the   NSSS   vendor   first   determined
the   key   nuclear   parameter   values   expected   to   be   bounding   over   the

plant lifetime.   The bounding values for these key parameters may
occur sometime during the first cycle of operation or during a
subsequent cycle. Therefore, depletion studies were performed and
the key parameters were determined for several cycles of operation
in order to obtain a set of key parameters which had a high
probability of being bounding over plant life. These bounding key
parameters are called the (initial) current limits. FSAR accident
analyses were performed using these bounding parameters.

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

For a typical reload cycle, some depleted fuel is removed from the
core and replaced by fresh fuel. The depleted fuel remaining in
the core and the new fuel are arranged within the core so that
power peaking criteria are met. Other plant changes may take place
between cycles or during a cycle. Examples are changes in
operating temperatures and pressures, and setpoint changes. These
changes may affect the key analysis parameters. If a key parameter
value for a reload exceeds the current limit, an evaluation is
performed using the reload key parameter. The reload evaluation
process is complete if the acceptance criteria delineated in the
FSAR are met, and internal documentation of the reload evaluation
is provided for the appropriate Virginia Power safety review. If,

however an accident reanalysis is necessary, more detailed analysis
methods and/or Technical Specifications changes may be required to
meet the acceptance criteria.   The NRC will be informed of the
results of the evaluation process in accordance with the
requirements of 10CFR50.59.

Therefore, the overall process is as follows:

1) Determine expected bounding key analysis
   parameters (initial "current limits").

2) Perform accident analysis using the bounding key
   analysis parameters and conservative assumptions.

3) Determine, for each reload, the value of each
   key analysis parameter.

4) Compare reload key analysis parameters to initial
   current limits.

5) Evaluate whether an accident reanalysis is needed
   based on the effect the reload key analysis
   parameters may have.

6) Perform reanalysis, change operating limits, or
   revise loading pattern as necessary.

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

The accidents analyzed for the FSAR and evaluated for each reload
cycle are listed in Table 1. The key parameters to be determined
for each reload cycle are listed in Table 2. The non-specific
parameters (designated "NS" in Table 2) are generated by evaluating
general core characteristics at conservative conditions, and the
specific parameters (designated "S" in Table 2) are generated by

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

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

3.3.3 Non-Specific Key Parameters

Non-specific key parameters are derived by evaluating core characteristics for conditions bounding those expected to occur during the reload cycle to ensure that the limiting values of the parameter are determined. These include conservative assumptions for such core parameters as xenon distributions, power level, control rod position, operating history, and burnup. Each non-specific key parameter generally serves as safety analysis input to several accidents including the accidents that also require specific key parameters, such as rod ejection.

3.3.3.1 Rod Insertion Limits

Control rod insertion limits (RIL) define the maximum allowable control bank insertion as a function of power level. Rod insertion limits (RIL) are required to maintain an acceptable power

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

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

The relationship between the RIA and the RIL is such that insertion
limits   determined   purely   from   RIA   considerations   are   usually
shallow   enough   that   other bases for rod insertion limits such as
acceptable   power   distributions   and   acceptable   postulated   rod
ejection   consequences are satisfied.   The determination of the RIL
is   made   by   a   1-D   or   3-D model simulation of the control banks
moving into the core with normal overlap while assuring the minimum
shutdown   margin   is   maintained at all power levels and insertions
from   HFP   to   HZP.   The   calculation is performed at EOC, and for
conservatism,   the   model is depleted in such a way that the burnup
and xenon distribution force the power to the top of the core. This
maximizes   the   worth   of the inserted portion of the control banks
which is not available for shutdown margin.

When   tentative   RIL   lines   have   been selected by the method just
outlined, they are then checked to see that they satisfy all of the

other bases.   If any basis is not satisfied by the tentative
insertion limits, the insertion limits are raised until the most
limiting basis is satisfied.   These limits are then checked against
the current Technical Specifications. If they violate the current
Technical   Specifications, a change is submitted to the NRC
requesting approval of these limits which would then become the
final rod insertion limits following NRC review and approval of the
associated Technical Specifications change.

3.3.3.2 Shutdown Margin

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

The amount of rod worth available is calculated with a 2-D model in
two parts. First, calculations are performed to determine the
highest worth single control rod or most reactive rod (MRR) for the
loading pattern. Next, the total control rod worth assuming the
MRR is stuck out of the core (N-1 rod worth) is determined and
reduced an additional amount for conservatism. The N-1 rod worth is
then reduced by the amount of rod insertion allowance to account
for rods being inserted to the insertion limits.

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

The shutdown margin is the amount by which the available negative reactivity (rod worth) exceeds the positive reactivity to be overcome. This calculation is performed for both beginning and end of cycle.

3.3.3.3 Trip Reactivity Shape

The trip reactivity shape is a measure of the amount of negative reactivity entering the core (in the form of control rods) after a

trip as a function of trip bank insertion. For conservatism in the accident analysis a minimum amount of trip worth based on near full power conditions is assumed to be available. This minimum trip worth is confirmed to be conservative by calculating the available trip worth for near full power conditions on a reload basis.

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

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

A conservative trip reactivity shape curve is one which shows less negative reactivity insertion for the major part of the rod insertion (i.e., except for the endpoints which are always equal),

than would be expected for an actual best estimate trip calculation based on operational power shape data. The FSAR safety analysis is based on a conservative curve generated using the methodology described above.

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

3.3.3.4 Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedbacks, in particular the moderator temperature (density) coefficient and the Doppler power and temperature coefficients. The reactivity coefficient generation for the reload design was discussed in Section 2.0.

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

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, a small reactivity coefficient value would be conservative. Some accidents and their analyses are not affected by reactivity feedback effects. Where reactivity effects are important to the analysis of an event, the use of conservatively large versus small reactivity coefficient values is treated on an event by event basis in the manner outlined in Reference 12.

3.3.3.5 Neutron Data

The delayed neutrons are emitted from fission products. They are normally separated into six groups, each characterized by an individual decay constant and yield fraction. The delayed neutron

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

The key analysis parameter is the Beta-effective, which is the product of the total Beta and the importance factor. The importance factor reflects the relative effectiveness of the delayed neutrons for causing fission. For some transients, it is conservative to use the minimum expected value of Beta-effective, while for others, the maximum expected value is more conservative. The use of conservatively large versus small Beta-effective values is treated on an event by event basis in the manner outlined in Reference 12. Beta-effective is calculated at the beginning and end of each reload cycle to obtain the bounding values for the cycle.

The prompt neutron lifetime is the time from neutron generation to absorption. It is calculated by core averaging a region-wise power weighted prompt neutron lifetime calculated by NULIF for each region in the core. The key analysis parameter used for transients is the maximum prompt neutron lifetime which occurs at the end of a reload cycle.

3.3.3.6 Power Density, Peaking Factors

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

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

3.3.4 Specific Key Parameters

Specific key parameters are generated by statically simulating an accident. The parameters are (or are directly related to) rod worths, reactivity insertion rates, or peaking factors. The static conditions selected are the most conservative conditions for the accident and account for variations in such parameters as initial power level, rod position, xenon distribution, previous cycle burnup, and current cycle burnup. In addition numerical uncertainty factors which are appropriate to the models being used are applied to the calculated parameter (References 1, 2, 3, 4, 9, 15).

3.3.4.1 Uncontrolled Control Rod Bank Withdrawal

The rod withdrawal accident occurs when control rod banks are withdrawn from the core due to some control system malfunction with a resulting reactivity insertion. The accident is assumed to be able to occur at HZP or HFP and a 1-D or 3-D model is used to perform the calculation.

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

In calculating the maximum differential rod worth for two

sequential highest worth control banks the following assumptions
and conservatisms are used:

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

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

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

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

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

The conservatisms associated with these calculations are:

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

2) The performance of the calculations at the cycle
   burnups which are expected to maximize the peak
   differential worth.

3.3.4.2 Rod Misalignment

Rod misalignment accidents result from the malfunctioning of the
control rod positioning mechanisms, and include: 1) static

misalignment of an RCCA (Rod Cluster Control Assembly, i.e., control rod), 2) single RCCA withdrawal, 3) dropped RCCA, and 4) dropped bank.

The important parameter for rod misalignment accidents is the minimum DNBR. The DNBR in the case of a rod misalignment accident is primarily a function of radial peaking factors (FdH). These peaking factors are determined using a 3-D model or a 1-D/2-D synthesis technique. For conservatism, all of the rod misalignment cases are performed at the cycle burnup which maximizes the radial peaking factors. This is generally at the beginning of the cycle, but may have to be determined from the depletion. Typically, a search is made to determine worst case rods for each type of rod misalignment. In addition, 1-D power sharings used in the synthesis are generated assuming conditions which maximize the synthesized FdH and uncertainty factors appropriate to the models used are applied. The maximum FdH peaking factors calculated for each of these types of rod misalignments are used to confirm that the DNB design basis limit has been met.

In the static misalignment accident, an RCCA is misaligned by being a number of steps above or below the rest of its bank. To simulate the RCCA misalignment above the bank, full core 2-D calculations with D bank in are made with the worst (the one that causes the highest FdH peaking factor) D Bank rod fully withdrawn. Next a 1-D calculation with D bank in to its insertion limit and the misaligned rod fully out is performed. The 2-D radial power

PAGE 43

distributions are then synthesized with the 1-D power sharings to determine the maximum FdH. The RCCA misalignment below its bank is bounded by the dropped RCCA analyses for Surry and North Anna as described later. Note that results of the RCCA misalignment upward analysis bound the FdH for the single RCCA withdrawal accident. However the single RCCA withdrawal accident is a condition III event and therefore a small percentage of fuel rods may be expected to fail. The event is analyzed to ensure that only a small percentage (<5%) of the fuel rods could exceed the fuel thermal limits and enter into DNB. The percentage of rods in DNB is determined through the use of a fuel rod census where the peak power for each rod in the core is tabulated.

The Surry and North Anna Units have differing protection systems in the event of dropped rod or dropped bank events. A dropped rod or bank in the Surry plant will initiate a turbine runback upon receipt of a rods on bottom signal or a negative flux rate signal which exceeds the system's setpoint. In addition a rod block is activated which precludes the control rods from being withdrawn in the event they are in the automatic mode. The North Anna Units are protected by a negative flux rate trip which trips the plant when a negative flux rate sufficient to exceed the setpoint is received on two of the four excore detectors.

For Surry the maximum FdH for the dropped rod event is calculated using a 1-D/2-D synthesis or a 2-D/3-D synthesis method. Full core 2-D calculations are performed to determine the radial power

distributions assuming any control rod (from either control bank or shutdown bank) may have dropped into the core. The radial power distributions are then synthesized with conservative 1-D axial power sharings to determine the maximum FdH.

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

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

3.3.4.3 Rod Ejection

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

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

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

The key parameters for the rod ejection accident are the ejected rod worth and total peaking factor (Fq). These key parameters are generated using steady state neutron diffusion theory or nodal methods. The rod ejection key analysis parameters for the bounding power levels and burnups must be derived for each initial and reload core. The detailed procedures for producing the rod ejection key analysis parameters are analytical simulations of the above scenario and include determining peaking factors and ejected rod worths. The 1-D, 2-D and 3-D computer models may be used in the rod ejection analysis.

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

conservative.

Another conservatism is that the 1-D and 3-D models are depleted in such a way as to insure that, at EOC, the top part of the core has less burnup than would be expected from a best estimate calculation based on operational history. The depletion is performed with D Bank partially inserted, which insures higher worths and peaking factors, for both HZP and HFP, as compared to the best estimate axial burnup shape.

3.3.4.4 Steamline Break

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

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

The steamline depressurization caused by this accident results in a temperature decrease in the reactor coolant which in the presence

of a negative moderator temperature coefficient results in a positive reactivity insertion. The reactivity insertion and a possible return to critical are therefore more limiting at EOC, when the MTC is most negative.

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

First, a detailed nuclear calculation (3-D model) is performed at end of cycle, hot zero power conditions with all rods fully inserted, except the highest reactivity worth stuck rod. These conditions are conservative initial assumptions for steambreak (see References 10, 11, and 12). Next, conditions including power, non-uniform inlet temperature distribution, pressure, and flow derived from the RETRAN code output data at the point where the minimum DNBR may occur is input to the 3-D model, and peaking factors and axial power distributions are generated. The stuck rod is assumed to occur in the coldest quadrant to maximize reactivity insertion.

Several limiting statepoints are chosen from RETRAN for minimum

DNBR analysis. The temperature and pressure information from these statepoints along with peaking factor information from the detailed nuclear calculation are input to COBRA to conservatively determine the minimum DNBR for the steambreak transient.

3.3.4.5 LOCA Peaking Factor Evaluation

A loss of coolant accident (LOCA) is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system. The LOCA evaluation methodology which has been employed by Virginia Power is consistent with the methodology used for past cycles of the Surry and North Anna Units by the fuel vendor for units operating under a constant axial offset strategy (CAOC). A description of this methodology can be found in References 4, 12, and 14.

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

power shapes assumed for large and small break LOCA analyses respectively.

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

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

The objective of the RPDC analysis is to determine acceptable delta-I limits that will guarantee that margin to all the applicable design bases criteria has been maintained and, at the same time, will provide enhanced delta-I operating margin over CAOC. Because the RPDC delta-I band is an analysis output quantity rather than a fixed input limit, as in CAOC, axial shapes which adequately bound the potential delta-I range must be generated.

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

The axial xenon distribution is a function of the core's operating history and, as a result, is constantly changing. In order to analyze a sufficient number of xenon distributions to ensure that all possible cases have been accounted for, a xenon "free oscillation" method is used to generate these distributions. By creating a divergent xenon-power oscillation, axial xenon distributions can be obtained that will be more severe than any experienced during normal operation, including load follow maneuvers.

For normal operation analysis, power levels spanning the 50% to 100% range are investigated to establish the RPDC delta-I limits. This range is consistent with the current CAOC technical specifications which do not impose axial flux difference limits or require CAOC operation below 50% of full power. Control rod bank insertion is limited by the technical specification rod insertion limits. These limits are a function of reactor power, and the rods may be anywhere between the fully withdrawn position and the

variable insertion limit.   In order to adequately analyze the various rod positions allowed, control rod insertions versus power level are selected which cover the range of rod insertions allowed for each particular power.

In addition the RPDC analysis is performed at several times in cycle life in order to provide limiting delta-I bands for the entire cycle, typically, three cycle burnups, near beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC), are chosen for the RPDC analysis.

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

To summarize, the procedure for insuring LOCA safety analysis coverage for the reload cycle consists of (1) determining the current limiting (maximum) F_q times relative power versus core

height curve; (2) determining the reload core maximum Fq times relative power values for all normal operational modes; and (3) specifying the appropriate Technical Specifications changes if there are envelope violations.

3.3.4.6 Boron Dilution

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

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

3.3.4.7 Overpower Evaluations

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

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

3.3.5 Non-Nuclear Design Key Parameters

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

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

3.4 SAFETY EVALUATIONS OF RELOAD SAFETY ANALYSIS

As has been discussed in previous sections, past analytical experience has allowed the correlation of the various accidents with those key safety parameters which have a significant impact on them. When a key safety analysis parameter exceeds its previously defined safety analysis limit, the particular transient(s) in question must be evaluated. This evaluation may be based on known sensitivities to changes in the various parameters in cases where the change is expected to be minimal and the effects are well understood. In cases where the impact is less certain or the effects of the parameter on the results is of a more complicated nature, then the transient will be reanalyzed. The majority of these reanalyses are performed with the Virginia Power RETRAN models described in References 5 and 13.

Each transient reanalysis method and assumption will be based on a conservative representation of the system and its response. This includes appropriate initial conditions, conservative reactivity feedback assumptions, conservative reactor trip functions and setpoints, and assumptions concerning systems performance. More discussion of these items can be found in References 5 and 13.

For those transients requiring core minimum DNBR analyses, the Virginia Power COBRA code is used. The necessary core operating condition inputs are determined from the RETRAN code. Peaking factor inputs are determined from the appropriate nuclear design

code.   More   discussion of the specific COBRA models and inputs is
provided in Reference 6.

TABLE 1

EVALUATED ACCIDENTS

CONDITION II EVENTS

a) Uncontrolled Rod Cluster Control Assembly
Bank Withdrawal from a Subcritical Condition

b) Uncontrolled Rod Cluster Control Assembly
Bank Withdrawal at Power

c) Rod Cluster Control Assembly Misalignment

d) Uncontrolled Boron Dilution

e) Partial Loss of Forced Reactor Coolant Flow

f) Startup of an Inactive Reactor Coolant Loop

g) Loss of External Electrical Load and/or
Turbine Trip

h) Loss of Normal Feedwater

i) Loss of all Off-Site Power to the Station
Auxiliaries (Station Blackout)

j) Excessive Heat Removal Due to Feedwater
System Malfunctions

k) Excessive Load Increase Incident

l) Accidental Depressurization of the Reactor
Coolant System

m) Accidental Depressurization of Main Steam
System

CONDITION III EVENTS

a) Complete Loss of Forced Reactor Coolant Flow

b) Single Rod Cluster Control Assembly Withdrawal
at Full Power

TABLE 1 (CONT.)

CONDITION IV EVENTS

a) Rupture of a Steam Pipe

b) Rupture of a Feedline

c) Single Reactor Coolant Pump Locked Rotor

d) Rupture of a Control Rod Drive Mechanism
    Housing (Rod Cluster Control Assembly Ejection)

e) Loss of Coolant Accident

TABLE 2
KEY ANALYSIS PARAMETERS

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

6) Prompt Neutron Lifetime (NS)
7) Boron Worth (NS)
8) Control Bank Differential Worth (NS)
9) Dropped Rod Worth (S)
10) Ejected Rod Worth (S)

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

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

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

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

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

3.5 NUCLEAR DESIGN REPORT

Before the operation of the cycle, a Nuclear Design Report which documents the nuclear design calculations performed in support of the cycle operation is issued by Reactor Engineering. This report is used by the Nuclear Operations Department in the preparation of startup physics tests and operator curves for use by station personnel in the operation of the cycle.

The parameters calculated for the reload safety evaluation are calculated for the most conservative conditions and in addition. have uncertainty factors applied to them. The startup physics and core follow data are best estimate calculations for conditions which the plant may see and be anticipated to operate under. For the most part these parameters are calculated for actual previous end-of-cycle conditions. However, where a parameter shows little or predictable variation for different previous end-of-cycle burnups the calculations may be made for the nominal end of the burnup window if values are needed prior to shutdown of the previous cycle.

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

1) Boron endpoints as a function of burnup, power, temperature, and rod configuration;

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

3) Isothermal temperature coefficients as a function of

burnup, temperature, rod configuration, and boron concentration;

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

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

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

7) Delayed neutron data;

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

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

10) Samarium worth following various startup and trip scenarios;

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

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

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

13) Assemblywise burnup as a function of cycle burnup;

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

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

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

Core physics measurements taken during the cycle startup and operation are compared to the physics design predictions documented in the Nuclear Design Report to insure that the plant is being operated within safety limits. Results of the measurements and the

comparisons to predictions are published by Nuclear Operations as a Startup Physics Test Report and a Core Performance Report for each reload cycle.

SECTION 4.0 - SUMMARY AND CONCLUSIONS

The in-house fuel management and reload design capability developed by Virginia Power closely parallels that of Westinghouse, but utilizes models and techniques developed in-house and licensed by the NRC. These models have been shown to accurately predict the necessary core parameters and simulate the core behavior necessary to perform the reload design process outlined in this report.

The groups responsible for reload core safety analysis at Virginia Power are the Reactor Engineering Group and the Safety Engineering Group. These are presently organized as branches of the Nuclear Engineering (NE) Section of the Engineering and Construction Department.

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

Reload   Safety   Evaluation   (RSE)   report is then issued by Nuclear

Safety   Engineering   documenting the results of the safety analysis

for   the   reload   cycle.   Figure   1   presents   a   summary   of   the

documentation   and   information   flow   of   the   safety   analysis

administration for a reload cycle.

Designing   a   core   that   meets all safety criteria is sometimes an

iterative   process involving interaction and trade-offs between the

Reactor Engineering and the Nuclear Safety Engineering Groups.   For

the typical reload, the derived key analysis parameters are bounded

by the current limit key analysis parameters.

If the current limits are exceeded, that event may be handled in a

number   of ways.   If the parameter only slighty exceeds its limits,

or   the   affected   transients   are   relatively   insensitive to that

parameter,   a   simple   quantitative   evaluation   may   be made which

conservatively   estimates   the magnitude of the effect and explains

why   an   actual   reanalysis   does not have to be made.   The current

limit is not changed.

If   the   deviation   is   large   and/or   expected   to   have   a   more

significant   or not easily quantifiable effect on the accident, the

accident is reanalyzed following standard procedures (such as those

used   in   the   FSAR analyses or other NRC approved methods).   After

the   reanalysis   is performed, and if the results of the reanalysis

meet   all   applicable   licensing   criteria the reload evaluation is

complete   upon completion of the appropriate internal documentation

and review.

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

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

FIGURE 1

SAFETY ANALYSIS ADMINISTRATION FOR A RELOAD CYCLE

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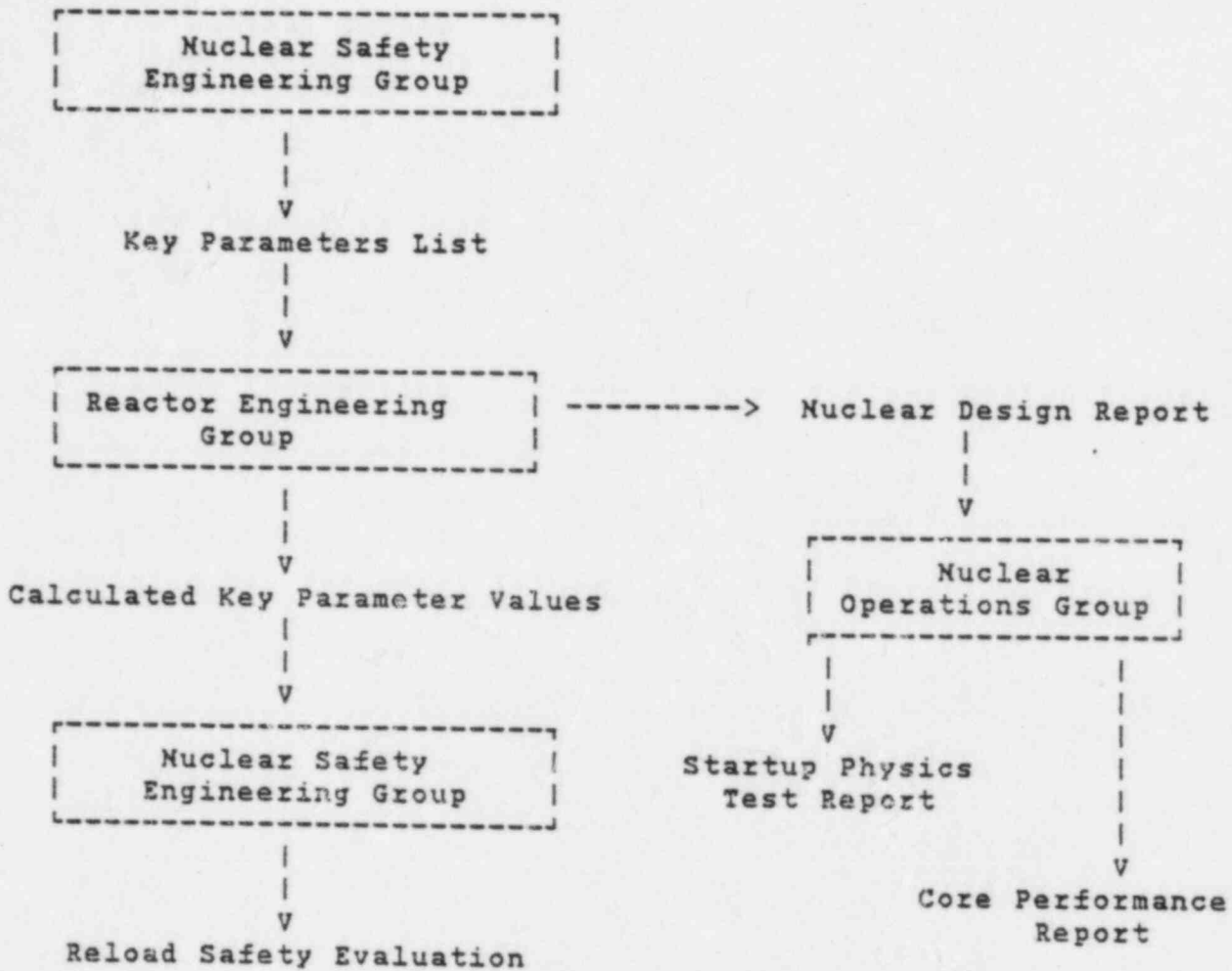
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