Reload Analysis Methodology for the Palo Verde Nuclear Generating Station

February 1993

Arizona Public Service Company
Nuclear Fuel Engineering and Analysis Department

Revision 00-NP-A

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

June 14, 1993

Docket Nos. 50-528, 50-529
and 50-530

Mr. William F. Conway
Executive Vice President, Nuclear
Arizona Public Service Company
P.O. Box 53999
Phoenix, Arizona 85072-3999

Dear Mr. Conway:

SUBJECT: APPROVAL OF RELOAD ANALYSIS METHODOLOGY REPORT - PALO VERDE NUCLEAR GENERATING STATION (TAC NOS. M85153, M85154, AND M85155)

We have completed our review of the Arizona Public Service Company (APS) submittal of April 6, 1993, which transmitted the Palo Verde Reload Methodology Report. This report describes the program undertaken by APS to develop the capability to perform reactor core reload analyses.

Our review consisted of an initial round of questions and an audit which was conducted in your offices from May 10 through May 12, 1993. Based on this review and the fact that APS has successfully participated in the CE Reload Technology Transfer Program, including independent reload core design and verification calculations, we conclude that the APS staff has the capability to use the CE codes as discussed in the report for reload analyses of CE-fueled Palo Verde cores. The topical report "Reload Analysis Methodology for the Palo Verde Nuclear Generating Station," Revision 00-P, describes the reload design process and the scope of the analyses which may be performed by APS, and is acceptable for referencing in Palo Verde license applications.

Our Safety Evaluation is enclosed for your information.

Sincerely,

Charles M. Trammell

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO ON-SITE AUDIT OF

PALO VERDE RELOAD ANALYSIS METHODOLOGY

ARIZONA PUBLIC SERVICE COMPANY

DOCKET NOS. 50-528, 50-529 AND 50-530

1.0 INTRODUCTION

In a letter of April 6, 1993, Arizona Public Service Company (APS) transmitted the Reload Analysis Methodology Report for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 to the U. S. Nuclear Regulatory Commission (NRC) for review (Ref. 1). This report summarizes the program which was undertaken by APS to develop the capability to independently perform the analysis of a PVNGS reload cycle.

In order to develop the capability to independently perform the analyses required for the design, licensing, operation and surveillance of a reload fuel cycle, APS contracted Asea Brown Boveri/Combustion Engineering (CE) to provide a training program referred to as the Reload Technology Transfer Program. This program consisted of classroom lectures, on-the-job training, and independent analysis. The scope of the program included all reload engineering technology except loss of coolant accident (LOCA) analysis, fuel mechanical design, and fuel fabrication engineering. These areas remain the responsibility of the fuel vendor, currently CE. The models and methods used have been previously approved by the NRC for use by CE.

The independent analysis was the final phase of the training program, whereby APS performed parallel reload analyses independent of CE. In order to evaluate the ability of APS to properly utilize the computer codes and methods for reload core design, the NRC conducted an on-site audit at the APS offices in Phoenix, Arizona, from May 10 through May 12, 1993. The NRC staff performing the audit consisted of Charles Trammell, Edward Kendrick, and Laurence Kopp from Headquarters, and Dennis Kirsch from Region 5.

The following CE-developed computer codes are used in the physics analyses:

(1) ROCS (Ref. 2), a coarse-mesh, two-energy group higher order difference diffusion theory neutronics code which can model all aspects of reactor operations from startup to refueling;

(2) MC (Ref. 2), a fine-mesh, two-energy group diffusion theory neutronics code which calculates fine-mesh (pin-wise) flux, power and burnup distributions through the application of the nodal imbedded method to

individual fuel assemblies using inter-assembly currents calculated by the coarse-mesh ROCS code;

(3) HERMITE (Ref. 3), a few-group, space and time-dependent neutron diffusion code which includes feedback effects of fuel temperature, coolant temperature, coolant density and control rod motion;

(4) QUIX (Ref. 4), a two-group, one-dimensional diffusion code used for axial shape analysis; and

(5) VISIONS (FLAIR) (Ref. 5), a three-dimensional, fast-running PWR simulator used to evaluate the response of the excore detectors to core power shape variations.

The core thermal-hydraulic analyses use the following CE codes:

(1) TORC (Ref. 6), a three-dimensional, open-lattice core thermal-hydraulic code used to determine the local coolant conditions and, in turn, the minimum departure from nucleate boiling ratio (DNBR) for the core; and

(2) CETOP-D (Ref. 7), a fast-running variant of the TORC code used as a design code in thermal margin analysis.

The fuel performance analyses use the following CE codes:

(1) FATES3A (Ref. 8), a fuel evaluation code which predicts the steady-state fuel rod temperature distribution, gap conductance, fuel and clad dimensions, plenum pressure, and stored energy for CE-designed fuel, and includes the NRC-required grain size restriction in the fission gas release calculation; and

(2) FATES3B (Ref. 9), a revised version of FATES3A with an improved predictive capability at high burnup.

In addition to the HERMITE and CETOP-D codes mentioned above, the non-LOCA transient and accident analyses are performed with the following codes:

(1) CESEC-III (Ref. 10), a system code which incorporates point kinetics, reactivity feedback, and core thermal-hydraulics to calculate system parameters including core heat, flow, pressure, and temperature, during a transient; and

(2) STRIKIN-II (Ref. 11), a code which provides a single, or dual, closed channel model of a core flow channel to calculate the clad and fuel temperatures during a transient.

2.0 EVALUATION

RELOAD DESIGN

The specific disciplines required of APS to implement a reload design are the following:

(1) Physics design
(2) Core thermal-hydraulics design
(3) Fuel performance design
(5) Transient and accident analyses (except LOCA)
(6) Generation of Core Operating Limit Supervisory System (COLSS) and Core Protection Calculator (CPC) setpoints, data base constants, and core operating margin assessment

The Reload Analysis Methodology Report presented comparisons of characteristic physics parameters calculated by APS for PVNGS 3 Cycle 3 with those calculated by CE. These included critical boron concentration, beginning-of-cycle (BOC) and end-of-cycle (EOC) boron worths, moderator temperature coefficients (MTCs), Doppler coefficients, CEA reactivity worths, and fuel assembly relative power densities. The comparisons show that the results of the CE and APS analyses agreed very closely. The minimal differences can be attributed to the difference in previous cycle burnup assumed by APS and CE. The comparison of core thermal-hydraulics parameter results between APS and CE for PVNGS 3 Cycle 3 were essentially identical.

The low power physics tests and the power ascension tests currently performed at BOC cover sufficient physics parameters to reasonably assure that the core is operating as designed, and adequate shutdown margin is available. In addition, the Core Follow Program currently performed by APS will be maintained to monitor physics and thermal-hydraulic parameters throughout core lifetime.

During the staff audit, selected calculations were examined and discussions were held with the APS staff to clarify specific points. These calculations included fuel performance design, margin setting events (anticipated operational occurrences) such as loss of flow, CEA withdrawal, CEA drop, and inadvertent boron dilution, and fuel failure events (accidents) such as steamline break, locked rotor, and CEA ejection. For some of the analyses examined, CE relied on calculations performed for a previous cycle. The corresponding APS calculations, on the other hand, were more detailed and demonstrated a comprehensive understanding by APS of reload technology.

The APS design calculations for their independent analysis of Unit 3 Cycle 3 were also forwarded to CE for a technical review to verify the ability of APS to correctly implement CE models and methods. In a letter from N. J. Breckenridge (CE) to P. F. Crawley (APS), dated November 30, 1992 (Ref. 12), CE concluded that the final APS Reload Topical Report and the underlying recorded calculations demonstrate proper application by APS Nuclear Fuel Management staff of the transferred CE methods and models to the PVNGS.

CONTROLS ON CPCS AND COLSS CONSTANTS

The NRC also reviewed the procedures that APS uses for review, implementation, and control of the constants that are installed in the Core Protection Calculator System (CPCS) and the Core Operating Limits Supervisory System (COLSS). These procedures were determined to be acceptable.

COMPUTER FACILITIES

APS reload design calculations will be run on Hewlett Packard 9000 Series 433 workstations. The three current APS workstations are configured identical to the workstations used by CE and will be directly connected to APS's existing Sun Microsystems LAN. All CE codes will be installed on each workstation in a controlled directory.

Although CE maintains the source code for APS, both the CE/APS Fuel Contract and the APS Software QA Procedure contain provisions for APS to maintain the source code independently of CE should APS decide to do so. Any source code changes made by APS will be governed by 05AC-ONF11, "NFM Software Quality Assurance Program for Non-Process Computer Software," and by 05DP-ONF09, "NFM Analysis Controls." Should APS make any source code changes, independently of CE, which change the calculated values of any safety related parameters, APS will perform a thorough engineering evaluation, validation, and verification prior to use of the modified source code for licensing related activities. If necessary, a topical report covering the code modifications would be prepared for NRC review and approval.

CHANGE IN FUEL VENDOR

As long as CE remains the PVNGS fuel vendor, qualification of models and methods will be performed using the CE qualification process. A change in fuel vendor will require an evaluation of any changes required to the physics and safety analysis methodology to accommodate that vendor's particular fuel designs. Changes of this type would undergo a thorough engineering evaluation, validation, and verification prior to use of the new fuel design.

DESIGN CONTROL

The Nuclear Fuel Management (NFM) process is controlled by several procedures ranging from program description procedures to detailed implementing procedures. These procedures were sampled and reviewed to assess the degree to which the NFM program implements the requirements of the quality assurance program, in general, and the commitments to ANSI N45.2.11 addressing design control. The staff concluded that the NFM program and implementing procedures adequately implement these requirements and commitments.

The auditors examined two quality assurance audits which addressed various aspects of the NFM program and the resolution of several findings and recommendations. The staff concluded that the oversight of the NFM resulted in finding and correcting the minor problems.

TRAINING

The NFM program for training and qualifying the staff assigned to perform core reload design work was examined and found to be adequate to assure a well qualified staff. The training program procedure was reviewed and found to detail an appropriate mix of training lectures, job performance measures, and knowledge assessment techniques to assure an adequately qualified staff. Written training materials used by the instructors were sampled and reviewed and found to contain the requisite level of detail to accomplish the training lecture purpose. Training records of several engineers were reviewed and found to demonstrate that the NFM organization contains a staff which has been adequately trained and certified to perform the required functions. Four engineers were interviewed to assess their perceptions regarding their preparation to perform their tasks; all professed an adequate knowledge level and supportive management.

ADDITIONAL EXAMPLES OF APS RELOAD DESIGN ABILITY

APS engineers have participated with CE in all phases of the Unit 2 Cycle 5 reload design. In addition, APS engineers have performed specific portions of reload calculations used by CE for Unit 1 Cycle 3, Unit 2 Cycle 3, and Unit 3 Cycle 3 reload analyses.

APS engineers perform the fuel management for current reload cycles and have performed this phase of design for a number of cycles for each of the PV... units. In addition, APS engineers have been involved in a number of fuel management scoping studies including investigations of high or extended burnup and alternative burnable absorber designs. APS has developed a set of "Fuel Management Guidelines" which were reviewed and accepted by CE. These guidelines were based on APS reload analysis knowledge and provide up-front guidance to avoid safety analysis problems by placing limits on key core physics parameters.

APS has supplied in-house support for many plant events and issues including development of reload specific Core Operating Characteristics Reports, reload startup tests, development of core physics for Core Follow Program, Core Data Book and startup test predictions. APS also has the ability to independently complete JCOs, 50.59s, Nuclear Safety Assessments following plant events, and Safety Assessments of Technical Specification changes.

These activities provide a satisfactory assessment of the NFM organization's readiness to independently perform core reload analyses.

3.0 CONCLUSION

Based on the results of this audit, and recognizing that APS has participated in the CE Reload Technology Transfer Program, we conclude that the APS staff has the capability to use the CE codes under discussion for non-LOCA reload analyses of the CE-fueled PVNGS cores. The Topical Report "Reload Analysis Methodology for the Palo Verde Nuclear Generating Station," Revision 00-P (Ref. 1) describes the reload design process and the scope of the analyses

which may be performed by APS, and is acceptable for referencing in PVNGS licensing applications.

4.0 REFERENCES

(1) "Reload Analysis Methodology for the Palo Verde Nuclear Generating Station," Revision 00-P, February 1993.

(2) "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983.

(3) "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients," CENPD-188, March 1976.

(4) "QUIX User's Manual," CE-CES-79, REV. 0-9, May 1987.

(5) Bollacasa, D. and J. C. Stork, "VISIONS - Versatile, Interactive Simulator of Nuclear Systems," American Nuclear Society Meeting, Nov. 29 - Dec. 3, 1981.

(6) "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P-A, April 1986.

(7) "CETOP Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3," CEN-160-(S), Rev. 1-P, September 1981.

(8) "Improvements to Fuel Evaluation Model," CEN-161(B)-P-A, August 1989.

(9) "Improvements to Fuel Evaluation Model," CEN-161(B)-P, Supplement 1-P-A, January 1992.

(10) "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, April 1974.

(11) "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135, April 1974.

(12) Letter from N. J. Breckenridge (CE) to P. F. Crawley (APS), V-92-237, "ABB CE Review of the APS Reload Topical Report," November 30, 1992.

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Date: June 14, 1993

ABSTRACT

This report presents a summary of the Arizona Public Service (APS) program to acquire the Combustion Engineering (CE) technology to perform reload licensing designs for the Palo Verde Nuclear Generating Station (PVNGS). The report describes the reload design process, including the computer programs utilized and the scope of the analyses performed by APS and the reload fuel vendor. Comparisons of the principal results of parallel and independent reload licensing analyses performed by APS and CE are presented. This comparison demonstrates the ability of APS to perform the analyses required for the licensing, operation and surveillance of a PVNGS reload cycle.

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1.0 INTRODUCTION AND SUMMARY

This report summarizes the program undertaken by Arizona Public Service (APS) to develop the capability to perform reload licensing analyses for the Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3. The purpose of this report is to demonstrate the capability of APS to independently perform the analyses required for the design, licensing, operation and surveillance of a reload fuel cycle.

The foundation for the approach and methodology described in this report was for APS to obtain models and methods previously approved by the Nuclear Regulatory Commission (NRC). Specifically, in 1986 APS contracted Asea Brown Boveri/Combustion Engineering (then known as Combustion Engineering and hereafter referred to as "CE") to provide a program which would enable APS to obtain CE technology to perform reload design and licensing activities. CE designed the Reload Technology Transfer Program to provide this training.

Section 2.0 contains an overview of the Reload Technology Transfer Program. This program was implemented in three phases: (a) classroom lecture, (b) on the job training, and (c) independent analysis. Section 3.0 provides an overview of the reload design process for a typical PVNGS reload design and is intended to enhance the understanding of comparisons between the results from the APS and CE analyses presented in section 4.0. This section also provides references to the previously licensed codes and methodology manuals submitted to the NRC by CE. Section 4.0 provides a comparison of the principal results of the independent analysis performed for PVNGS Unit 3 Cycle 3.

The independent analysis was the final phase of the Reload Technology Transfer Program, whereby APS performed parallel reload analyses independent of CE. Based on the comparisons of the principal design results as presented in section 4.0, APS has demonstrated the ability to properly utilize the codes and methods used for the PVNGS reload core design.

2.0 RELOAD TECHNOLOGY TRANSFER PROGRAM OVERVIEW

In 1986 APS contracted CE to provide a training program to transfer the CE licensed reload engineering methodology to the APS engineering staff. The scope of the program included all reload engineering technology except:

Loss Of Coolant Accident (LOCA)
Fuel mechanical design
Fuel fabrication engineering

Currently, APS does not intend to perform any LOCA analyses, since the LOCA analyses presented in the Updated Final Safety Analysis Report (UFSAR) are typically conservative with respect to reload core design. In addition, the fuel mechanical design and fuel fabrication engineering are logical areas to remain the responsibility of the fuel vendor.

The training program was developed to occur in three phases:

Phase 1 - classroom training
Phase 2 - on the job training
Phase 3 - independent analysis

2.1 PHASE 1 - CLASSROOM TRAINING

The objective of the classroom training phase was to provide basic training in the fundamentals of CE reload design methods and software. This training was provided by CE engineers familiar with the design and licensing analyses of PVNGS. The topics covered in this phase included descriptions of the various phenomena being modeled, the basis, limitations and uncertainties of the models, descriptions of the processes used to generate input for the models, and the use of the models to generate inputs to subsequent analyses. Table 2.1-1 provides a list of the 21 classroom modules presented.

Table 2.1-1 Classroom Training

Course ID	Course Title (duration)
LD100	Overview of Reload Design Process (1 day)
LD001	Quality Assurance Procedures (1 day)
LD002	Technical Specification Changes (3 days)
LD003	Reload Analysis Report (1 day)
NA001	Overview of Physics Models (3 days)
NA002	Cross Section Generation (3 days)
NA003	Enrichment Setting, Fuel Management, and As-built Analysis (5 days)
NA004	Core Follow Analysis (3 days)
NA005	Startup Test Predictions / Technical Data Book (3 days)
NA006	CECOR Coefficient Generation (4 days)
NA007	Physics Input to Safety Analysis (10 days)
SA001	Design Thermal-Hydraulics (5 days)
SA002	Fuel Performance Analysis (3 days)
SA003	Non-LOCA Transient Analysis (10 days)
SA103	CPC/COLSS Related Transient Analysis (10 days)
CC001	Overview of COLSS/CPC/CEAC Analysis (3 days)
CC002	COLSS/CPC Power Distributions (5 days)
CC003	On-Line Thermal-Hydraulics (5 days)
CC004	Other CPCS Algorithms (3 days)
CC005	Other COLSS Algorithms (3 days)
CC006	Uncertainty Analysis (4 days)

This phase of the program started in October 1987 and was completed in March 1990. APS invested approximately 6000 engineer hours in this training phase.

2.2 PHASE 2 - ON THE JOB TRAINING

In the On the Job Training (OJT) phase of the Reload Technology Transfer Program, APS engineers prepared design analyses for a reload design previously prepared by CE. In this phase CE provided on site supervision at critical intervals during the process to provide training assistance. The OJT phase included tasks such as: generation of input data, execution of related computer programs, interpretation of computer code output, and documentation and review of related calculations. The objective of this phase was to reinforce the classroom training provided in phase 1 by simulating actual design situations.

Phase 2 started in January 1988 and was completed in December 1990. Approximately 17000 engineer hours were required to complete this phase of the program.

2.3 PHASE 3 - INDEPENDENT ANALYSIS

The final phase of the Reload Technology Transfer Program was the Independent Analysis Phase. In this phase, APS engineers performed reload analyses independent of CE based on the Unit 3 Cycle 3 core design. The objective of this phase was to demonstrate that the reload technology had been effectively transferred to the APS engineering staff. This phase started in April 1990 and was completed in October of 1991. Section 4.0 provides a detailed discussion of this phase of the training program, including a comparison of the principal results of the APS and CE calculations.

3.0 OVERVIEW OF THE RELOAD ANALYSIS PROCESS

The purpose of this section is to provide a general overview of the reload design process used for the Independent Analysis phase. This section is included to enhance the understanding of comparisons between the results from the APS and CE analyses presented in section 4.0. This section also provides references to the previously licensed computer code and methodology manuals submitted to the NRC by CE.

A simplified diagram of the reload analysis process is shown in Figure 3.0-1. The process inputs to the reload analysis include plant and cycle performance objectives, Technical Specifications, the bases for previously licensed analyses, and the groundrules. The groundrules are used to document the analysis bases in the reload engineering effort. The process outputs of the reload analyses include the Reload Analysis Report (RAR), Technical Specification changes, safety and monitoring setpoints, and plant startup and operations data.

This section discusses typical interfaces of functional areas preparing the reload products, and is organized to present the reload analyses process in terms of the major disciplines required to support the reload design. Figure 3.0-2 shows a simplified network diagram which illustrates the relationships between the disciplines required to implement the reload design. These disciplines are defined as:

a. Physics Design - responsible for all neutronics related analyses.

b. LOCA Design - responsible for all LOCA analyses and Emergency Core Cooling System (ECCS) performance related analyses.

c. Core Thermal-Hydraulics Design - responsible for the design and development of models used for Departure from Nucleate Boiling Ratio (DNBR) calculations.

d. Safety Design - responsible for all transient analyses excluding LOCA.

e. Fuel Performance Design - responsible for all fuel rod thermal design.

f. Core Operating Limit Supervisory System (COLSS) / Core Protection Calculators (CPCS) Design - responsible for the generation of COLSS and CPCS setpoints, data base constants and core operating margin assessment.

g. Fuel Assembly Design - responsible for all analyses related to the mechanical design of the fuel assemblies and Control Element Assemblies (CEAs), including direct interface with the fuel fabrication facility.

Currently, APS relies on the fuel vendor to provide engineering support in the LOCA Design and Fuel Assembly Design areas since the amount of engineering support required in these areas for a typical reload design is small. The remainder of section 3.0 provides an overview of the reload design process in each of the disciplines within the APS scope.

Figure 3.0-1 Simplified Diagram of Reload Analyses Process

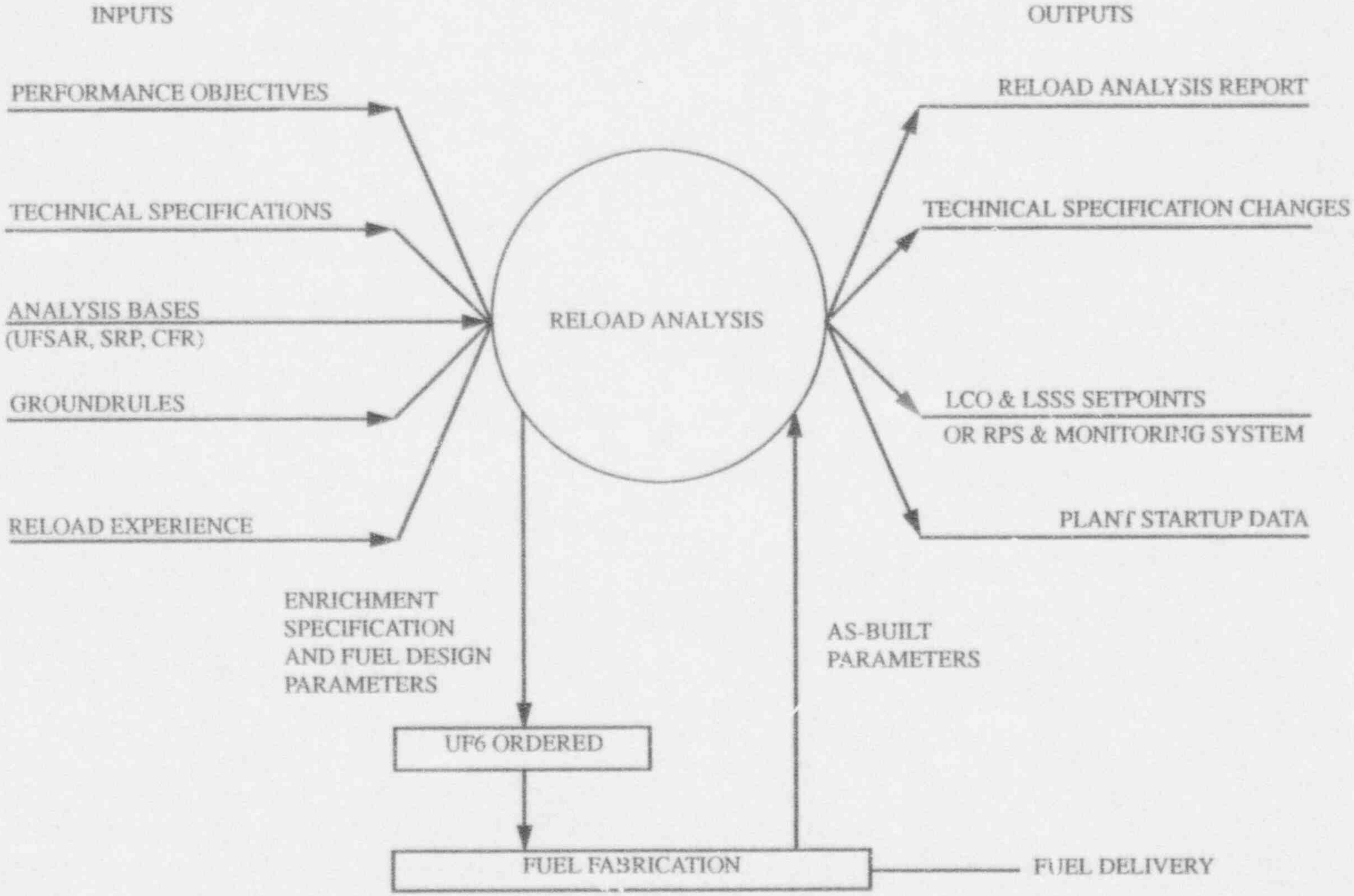


Figure 3.0-2 Simplified Reload Analysis Network Diagram

3.1 PHYSICS DESIGN

This section describes the physics analyses that will be performed by APS on a cycle by cycle basis. Included in this discussion are brief descriptions of the main inputs and outputs of the physics analyses, and the analyses themselves.

3.1.1 Physics Analyses

3.1.1.1 Models and Depletions Analysis

The Reactor Operation and Control Simulator (ROCS) and Mesh-Centered (MC) (reference 1) physics models are constructed for the reload fuel cycle. ROCS is a coarse mesh, two-energy group higher order difference diffusion theory neutronics code which can model all aspects of reactor operations from startup to refueling. MC is a fine mesh, two energy group diffusion theory neutronics code which calculates fine-mesh (pin-wise) flux, power and burnup distributions through the application of the nodal imbedded method to individual assemblies using inter-assembly currents calculated by the coarse mesh ROCS code.

Three separate depletions are performed with the physics models. These depletions are performed at Hot Full Power (HFP) based on three different points within the previous cycle shutdown window, short (best estimate cycle Effective Full Power Days (EFPD) minus delta[1]), nominal (best estimate cycle EFPD), and long (best estimate cycle EFPD plus delta[1]). These depletions supply core isotopic distributions and nominal HFP power distributions, which bound any possible nominal distributions of the reload fuel cycle as long as the previous cycle ends between the short and long end points.

This analysis utilizes inputs from the groundrules, fuel management scoping studies and the previous cycle(s) as-built analysis. The groundrules document is a list of guidelines and assumptions to be used in the reload engineering effort. The groundrules give the previous cycle shutdown date and required future cycle length. The fuel management scoping studies provide a fuel shuffle pattern and new fuel batch design (enrichments and burnable absorber distributions) which have been verified for energy content and acceptable physics parameters. The as-built analysis from the previous cycle furnishes a model of the isotopic characteristics of the fuel that will be retained or reinserted from prior cycles. The above physics models, isotopic distributions, and nominal HFP power distributions are used as input to the analyses described below.

1. delta equates to approximately 1/2 of the burnup window

3.1.1.2 Physics Data for Safety Design

The typical physics data required for the Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses is listed below:

a. Generic Data

1. Fuel temperature coefficient
2. Kinetics parameters
3. Minimum net scram worths with worst rod stuck out and Power Dependent Insertion Limit (PDIL) worth subtracted
4. Maximum integrated radial peaking factor in the cycle as a function of power
5. Radial pin census (statistical population of pins based on radial power level) at nominal full power all rods out and selected rodded configurations and burnups)
6. Axial power distributions
7. Core average linear heat rate
8. Scram reactivity insertion as a function of scram bank position

b. Steam Line Break Event Data

1. Moderator cooldown all rods in with worst rod stuck out curve
2. Inverse boron worths
3. End Of Cycle (EOC) net CEA worth

c. CEA Withdrawal Data

1. Maximum reactivity insertion rate
2. Maximum Fq
3. Delayed neutron source term for subcritical multiplication
4. Axial power distribution
5. Maximum radial distortion factor
6. Minimum temperature decalibration factor for excore signal
7. Maximum power decalibration factor for excore signal

d. Reactor Power Cutback Data

1. Minimum bank worths accounting for allowed rod insertion
2. Maximum radial peaking distortions

e. Full Length CEA Drop Data

1. Maximum static radial distortion
2. Maximum xenon redistribution penalty factor for 15 minutes and one (1) hour

f. Part Length CEA Drop Data

1. Maximum reactivity insertion
2. Maximum product of static radial distortion and xenon redistribution penalty factor for 15 minutes and one (1) hour
3. Axial power distributions, before and after drop

g. Boron Dilution and Inadvertent Startup of a Reactor Coolant Pump Data

1. Maximum Critical Boron Concentration (CBC) and corresponding minimum inverse boron worth
2. Minimum refueling boron concentration
3. Isothermal temperature coefficient
4. Lower bound of stuck CEA worth

h. Asymmetric Steam Generator Transient Data

1. Radial distortion factor versus core inlet temperature tilt

i. CEA Ejection Analysis Data

1. Maximum ejected rod worth with and without doppler feedback
2. Pre-ejected and post-ejected peaking factors
3. Pin census

This list includes both core average and spatially dependent data. This data is either obtained directly from the ROCS, MC, HERMITE (section 4.3.3.1.1.4 of references 2 and 3), and QUIX (section 4.3.3.1.1.4 of references 2 and 4) codes or calculated from quantities found within the outputs of these codes. Additionally, a space-time open channel physics model is required. A One-Dimensional (1-D) HERMITE model is constructed for this purpose.

This analysis utilizes inputs from the groundrules and the models and depletions analysis. The groundrules provides various operational parameter targets, such as PDIL, allowable moderator temperature coefficient range, and moderator temperature control programs. The models and depletions analysis furnishes physics models, isotopic distributions, and nominal HFP power distributions.

3.1.1.3 Physics Data for Fuel Performance Design

The physics data required for the fuel performance design includes both core average and spatially dependent data. The core average data is calculated from ROCS output. The spatially dependent data is calculated in the models and depletions analysis with the ROCS and MC codes. This spatially dependent data (power and burnup) is converted into the format required for the fuel performance design. A typical list of physics data provided for the fuel performance design is shown below:

This analysis has inputs from the groundrules and the models and depletions analysis. The groundrules is the source of the target LOCA limit on initial Linear Heat Rate (LHR). The models and depletions analysis gives the isotopic distributions and nominal HFP power distributions.

3.1.1.4 Physics Data for LOCA Design

The physics data required for the LOCA analyses include both core average and spatially dependent data. The spatially dependent data is calculated from the fine mesh (pin-by-pin) ROCS/MC power data, which is obtained from the models and depletions analysis. This power distribution data is extracted from the ROCS/MC results and converted into the radiant heat transfer factor data input to the LOCA analyses. The remaining data is calculated from ROCS, MC, HERMITE and QUIX code outputs. As noted below, some of this data is calculated in other analyses. A typical list of physics data provided for the LOCA analyses is shown below:

This analysis has inputs from the models and depletions analysis. The models and depletions analysis provides the [

]

3.1.1.5 Physics Data for Core Thermal-Hydraulics Design

The physics information required for the core thermal-hydraulics analyses is fine mesh (pin-by-pin) power distribution data, the maximum burnup for determining rod bow penalties on DNBR, and the number of non-fuel rods (burnable absorber rods) in the core. The power distribution data is calculated in the models and depletions analysis with the ROCS and MC codes. The power distribution data is extracted from the ROCS/MC results and converted to the pin-by-pin power distribution data input to the core thermal-hydraulics analyses. The maximum burnup for determining rod bow penalty on DNBR is determined by finding the burnup at which the decrease in relative rod power with burnup offsets the rod bow penalty on DNBR, which is also burnup dependent. A typical list of physics data provided for the core thermal-hydraulics analyses is shown below:

[ ]

This analysis has inputs from the models and depletions analysis, the fuel performance design, and the CE Fuel and Poison Rod Bowing topical report (reference 5). [

]

3.1.1.6 Physics Data for Fuel Assembly Integrity Design

The physics information required for the fuel assembly integrity design includes both core average and spatially dependent data. The core average data is determined from ROCS output. The spatially dependent data is calculated in the models and depletions analysis with the ROCS and MC codes. This spatially dependent data (burnup, fluence and burnable absorber Boron-10 absorption rate) is extracted from the ROCS/MC results and adjusted by appropriate uncertainties to form the physics data input to the fuel assembly integrity analysis. A typical list of physics data provided for the fuel assembly integrity analysis is shown below:

[ ]

This analysis has inputs from the groundrules and the models and depletions analysis. The groundrules gives the target LOCA limit on LHR. The spatially dependent data, nominal HFP power distributions, and the pertinent cycle lengths are obtained from the models and depletion analysis.

3.1.1.7 Physics Data for COLSS/CPCS Design

The physics data required for the COLSS/CPCS analyses includes both core average and spatially dependent data. The core average data is determined from ROCS output. Some of the spatially dependent data (nominal HFP power distributions) is calculated in the models and depletions analysis with the ROCS and MC codes and then extracted and surveyed to form the maximum F_r data input to COLSS/CPCS analyses. The other spatially dependent data is calculated with the ROCS and MC codes. Additionally, a "quick running" physics model is used in COLSS/CPCS analyses. A three dimensional model for a "quick running" code such as FLAIR (also called VISIONS, Section 4.3.3.1.1.4 of references 2 and 6) is constructed for this purpose. A typical list of physics data provided for the COLSS/CPCS analyses is shown below:

[ ]

[ ]

This analysis has inputs from the models and depletions analysis, physics data for fuel performance design and physics data for safety design. The nominal HFP ARO fine mesh power distributions, rodded fine mesh power distributions, and isotopic distributions are from the models and depletion analysis. The physics data for fuel performance design provides the core average LHR. The physics data for safety design gives the CEA drop distortion factors.

3.1.1.8 As-built Analysis

As-built ROCS and MC models of the reload are constructed from: (a) the design models from the models and depletions analysis, (b) the actual as-built fresh fuel isotopics calculated from the fuel manufacturing assays, and (c) the burned fuel isotopics calculated at the actual shutdown burnup with the as-built model from the previous cycle.

The physics data calculated with the above described as-built models includes both core average and spatially dependent data. The core average data is calculated with the ROCS code. The spatially dependent data is calculated with the ROCS and MC codes. The as-built models are depleted and various restart cases are performed to obtain the predicted as-built power distributions. Some of the spatially dependent data is extracted from the ROCS/MC results and converted to the [ ] The other spatially dependent data is calculated with the ROCS and MC codes. Another product of the as-built analysis is the as-built loading pattern for the next reload fuel cycle. The physics data produced in the as-built analysis is shown below:

a. As-built loading pattern

   1. A core loading map is constructed based on the as-built fuel assembly assays. This map gives the location, orientation, and serial number of each assembly to be loaded into the core. The new fuel assemblies are judiciously placed such that the as-built loading variations effect on radial peak and azimuthal tilt results in the lowest possible radial peak and tilt.

b. Low power physics test predictions

   1. CBC
   2. Isothermal Temperature Coefficient
   3. Fuel Temperature Coefficient
   4. Change in isothermal temperature coefficient with respect to change in boron
   5. Inverse boron worth
   6. CEA bank worths
   7. Delayed neutron parameters

c. Power ascension physics test predictions

   1. Radial power distributions
   2 Axial power distributions
   3. Peaking factors (Fxy, Fr, Fz, Fq)
   4. Isothermal temperature coefficient, power coefficient, fuel temperature coefficient
   5. Lead CEA bank integral curve
   6. Change in isothermal temperature coefficient with respect to change in boron
   7. CBC

d. CECOR Library

   1. A CECOR library consisting of coupling coefficients, W-primes, 1-pin/boxes, axial boundary conditions, azimuthal tilt G-factors, and a CECOR input deck is constructed.

e. COLSS data described in item c of section 3.1.1.7

f. Curves to support shutdown margin determination (Core Data Book)

   1. Inverse boron worth versus core average temperature, versus boron ppm, versus burnup
   2. Isothermal temperature coefficient versus core average temperature, versus boron ppm, versus burnup

3. CEA worth data versus core average temperature versus boron ppm, versus burnup
   a. Total CEA worth
   b. Worth of worst stuck CEA
   c. Worth of worst two stuck CEAs

g. Verification of safety analyses

   1. The physics parameters transmitted for input to the safety analyses, which are sensitive to as-built variations, are reevaluated with respect to data generated based on the design model.

This analysis has inputs from the models and depletions analysis, fuel manufacturing assays, as-built analysis from the previous cycle, and shutdown burnup from the previous cycle. The models and depletions analysis gives a design model of the reload which needs the actual as-built burned and fresh fuel isotopics to accurately model the as-built core. The fuel manufacturing assays provide uranium loading weight data from which the as-built fresh fuel isotopics are calculated. This as-built analysis provides a model of the isotopics of the burned fuel that will be held over for additional burning in the future cycle. This shutdown burnup provides the burnup to which the previous as-built model must be depleted to obtain the reload fuel cycle as-built burned fuel isotopics.

3.2 CORE THERMAL-HYDRAULICS DESIGN

This section describes the core thermal-hydraulics analyses that was performed by APS using previously approved CE methodology. Included in this discussion is a brief description of the main input/outputs of the core thermal-hydraulics analyses and the analyses themselves. The reload core thermal-hydraulics analyses can be broken into two major categories: (a) DNBR analyses using TORC/CETOP-D to produce a core thermal-hydraulics design model, (b) calculation of the design DNBR limit by statistically combining system parameter uncertainties.

One of the limits imposed on the power level of a reactor is the Minimum allowable value for the Departure from Nucleate Boiling Ratio (MDNBR). MDNBR expresses the adequacy of cooling in the most limiting flow channel in the reactor core and is therefore a measure of the core thermal margin. Section 3.2.1 describes the CE open-core thermal-hydraulics code (TORC) used to determine core thermal hydraulic performance and outlines the CE methodology used for the development of a core thermal-hydraulics model.

The principal core thermal-hydraulics design basis was to avoid thermally induced fuel damage during normal steady state operation and during Anticipated Operational Occurrences (AOOs). Specified Acceptable Fuel Design Limits (SAFDLs) exist on peak fuel temperature and DNBR to meet this design basis. The DNBR SAFDL was determined statistically such that there was at least a 95% probability at the 95% confidence (95/95 probability/confidence) level that the limiting fuel rod in the core would not experience DNB. Section 3.2.2 outlines the Statistical Combination of Uncertainties (SCU) methodology used to determine the design DNBR limit.

3.2.1 DNBR Analyses Models

The purpose of the DNBR analysis was to provide a design thermal-hydraulic model (i.e., CETOP-D model) for reload safety and COLSS/ CPCS design. The steady state DNBR analysis was performed using the CE open-core thermal-hydraulics code (TORC) as described in reference 7. TORC has received generic NRC approval for use in licensing analyses (references 8 and 9). TORC solves the conservation equations for a 3-dimensional representation of the open-lattice core to determine the local coolant conditions at all points within the core. Lateral transfer of mass, momentum and energy between neighboring flow channels were accounted for in the calculation of the local coolant conditions. These coolant conditions were then used with the CE-1 Critical Heat Flux (CHF) correlation (references 10 and 11) to determine the MDNBR for the reactor core.

The core thermal-hydraulics modeling in TORC was divided into three stages. In the first stage, coolant conditions throughout the core on the coarse-mesh basis were cal-

culated. A core quadrant was modeled, in which the smallest unit represented by a flow channel was a single fuel assembly. The geometry and heat generation of the fuel assemblies were input. The axial distributions of flow and enthalpy in each fuel assembly were then calculated along with the transport quantities (mass, momentum and energy) that cross the lateral boundaries of each flow channel.

In the second stage, the hot assembly and adjoining fuel assemblies were modeled with a coarse mesh. The hot assembly was divided into four assembly quadrants. One of these quadrants contains the limiting subchannel. The lateral transport of mass, momentum, and energy from the stage 1 analysis were imposed as boundary conditions on the peripheral boundary enclosing the hot assembly and its neighboring assemblies. The inlet flow, inlet temperature, and fuel heat generation were also input into the code. The lateral transport of mass, momentum, and energy between the flow channels within the stage 2 mesh were calculated.

The third stage involved fine-mesh modeling of the assembly quadrant containing the limiting subchannel. In this stage, the limiting subchannel, most of the other subchannels and all of the fuel rods were modeled individually. All of the flow channels used in this stage were hydraulically open to their neighbors. The lateral transport of mass, momentum, and energy from the stage 2 calculations were imposed on the lateral boundaries of the assembly quadrant containing the limiting subchannel. The local coolant conditions were calculated for each flow channel. These coolant conditions were then input into the DNB correlation, and the MDNBR was determined for the most limiting subchannel in the reactor core.

The CETOP-D code, a variant of the TORC code, is used as a design code for the core thermal-hydraulics analyses. CETOP-D was developed to reduce the computer time needed for core thermal-hydraulics analyses, while retaining the capabilities of the TORC design model. CETOP-D used transport coefficients for improved prediction of diversion cross flow and turbulent mixing between adjoining channels. Furthermore, [ ] is used to solve the conservation equations, replacing the iterative method used in the TORC code. A complete description of CETOP-D is contained in reference 12 and the generic applicability of the CETOP-D to PVNGS is detailed in reference 13.

The CETOP-D code provided an additional simplification to the conservation equations due to the specific geometry of the model. It had a total of four core thermal-hydraulics channels to model the open-core fluid phenomena. Figure 3.2-1 shows a typical layout of these channels. Channel 2 was a quadrant of the hottest assembly in the core and Channel 1 was an assembly which represented the average coolant conditions for the remaining portion of the core. The boundary between Channel 1 and 2 was open for crossflow, but there was no turbulent mixing across the boundary. The

outer boundaries of the total geometry were assumed to be impermeable and adiabatic. The lumped Channel 2 includes Channels 3 and 4. Channel 3 comprised the subchannels adjacent to the MDNBR hot Channel 4. Channels 2' and 2" were used for self-generation of enthalpy transport coefficients. The location of the MDNBR channel was determined from the TORC analysis of the core.

To produce a core thermal-hydraulics design model for a specific reload core, [

] This DNBR limit was derived using the statistical method discussed in section 3.2.2.

3.2.1.1 Major Inputs to the Core Thermal-Hydraulics Design

Pin-by-pin power distributions and the corresponding core wide power distribution for the potentially limiting assemblies are required as input to the TORC modeling process. Potentially limiting assemblies are determined based on the inlet flow distribution, assembly power, and the "flatness" of the pin-by-pin power distribution within the assembly. [

]

The detailed fuel assembly dimensions (clad outer diameter, pitch, spacer grid locations, etc.) and the dimensions of pertinent reactor internals (e.g., core shroud) are based on a set of engineering drawings of the fuel and reactor internals. Other inputs include the core inlet flow distribution and core exit pressure distributions obtained from flow model tests.

The operating conditions that require definition in a TORC or CETOP-D case include system pressure, core inlet temperature, core average heat flux and core average mass velocity. These operating conditions are based on the groundrules, and are consistent with the safety and COLSS/CPCS design. Core physics data is required for each reload while most other inputs are cycle independent, barring any design changes to the fuel, other fuel related components, or reactor internals.

3.2.1.2 Major Outputs of the Core Thermal-Hydraulics Design

A 4-pump CETOP-D model (i.e. a model based on full RCP flow) was provided for the reload safety design and for use in the COLSS and CPCS design. Typically, a cycle independent CETOP-D model is provided with a penalty applied to the core average heat flux. The CETOP-D model is valid for steady state normal operation allowed by the PDILs and for quasi-steady state analysis of transients, such as:

a. Increased main steam flow

b. Loss of condenser vacuum

c. Loss of normal AC power

d. Uncontrolled CEA withdrawal from a subcritical or low power condition

e. Uncontrolled CEA withdrawal at power

f. CEA misoperation

g. CEA ejection

h. Total loss of forced reactor coolant flow

i. Steam system piping failure

j. Feedwater system pipe breaks

k. Reactor coolant pump shaft breaks

l. Steam generator tube rupture

Typically, the CETOP-D model is [

]

3.2.2 Statistical Combination of Uncertainties

The Statistical Combination of Uncertainties (SCU) methodology described in Ref. 15 is used to calculate the minimum DNBR limit value (1.24 since Cycle 2), and the DNBR Probability Density Function (PDF) used with the 95/95 probability/confidence DNBR tolerance limit in the COLSS and CPC uncertainty analyses.

The data required for a detailed thermal-hydraulic analysis are divided into two main groups:

system parameters, which describe the physical system and are not monitored during reactor operation, and

state parameters, which describe the operational state of the reactor and are monitored during operation.

There is a degree of uncertainty in the value used for each of these parameters. The SCU methodology is used to statistically combine uncertainties of the system parameters and incorporate their effects on DNBR to derive the minimum DNBR limit.

The individual uncertainties that are combined in the system parameter SCU analysis are the following:

These uncertainties are statistically combined to yield the DNBR PDF. This DNBR PDF is then deterministically combined with the [ ] to determine the minimum DNBR limit. This DNBR limit, while using the nominal values of system parameters in design analysis, will ensure with at least 95 percent probability and 95 percent confidence level that departure form nucleate boiling will not occur anywhere in the core. This limit is also used in the on-line COLSS DNBR power operating limit calculation and as the CPCS DNBR trip setpoint. The DNBR PDF is also used in the COLSS and CPC overall uncertainty analyses.

In the Modified SCU methodology (reference 14), the system parameter uncertainties are combined in the same way to determine the DNBR PDF. However,

Thus, both the system and state parameter uncertainties are combined statistically in the COLSS and CPCS overall uncertainty factors.

The use of as response surface to represent a complicated, multivariate function is an established statistical method. A response surface relating MDNBR to system parameters is created. Conservatism is achieved by selecting the "most adverse set" of state parameters that maximizes the sensitivity of MDNBR to system parameter variations. TORC analyses are performed to determine the sensitivity of the system parameters at several sets of operating conditions (state parameters). Data to estimate the coefficients of the response surface is generated in an orthogonal composite

design using the TORC code with the CE-1 CHF correlation. [

]

The MDNBR PDF is estimated using the response surface in a Monte Carlo simulation. The estimated MDNBR PDF is approximately normal and a 95/95 probability/ confidence limit is assigned using normal theory. The SIGMA code applies Monte Carlo and stratified sampling technique to combine arbitrary PDFs numerically. (The Monte Carlo simulation and the SIGMA code have been reviewed and found acceptable (reference 16)). This code is used with the response surface to combine system parameter PDFs with the CHF uncertainty and the TORC code uncertainty into a resultant MDNBR PDF.

3.2.2.1 Major Inputs for the SCU Analysis

The TORC model used in the DNBR analysis as discussed in section 3.2.2 is also used in the development of the MDNBR limit. Physics design information on the maximum burnup for determining the rod bow penalty on DNBR is an input. Other inputs include the detailed fuel assembly dimensions and tolerance values used to determine many of the system parameter PDFs.

[

]

Other penalties imposed by the NRC in the course of their review of the Unit 1 Cycle 1 SCU analysis (reference 15) are included in the overall uncertainty penalty factors derived in the MSCU Analysis. [

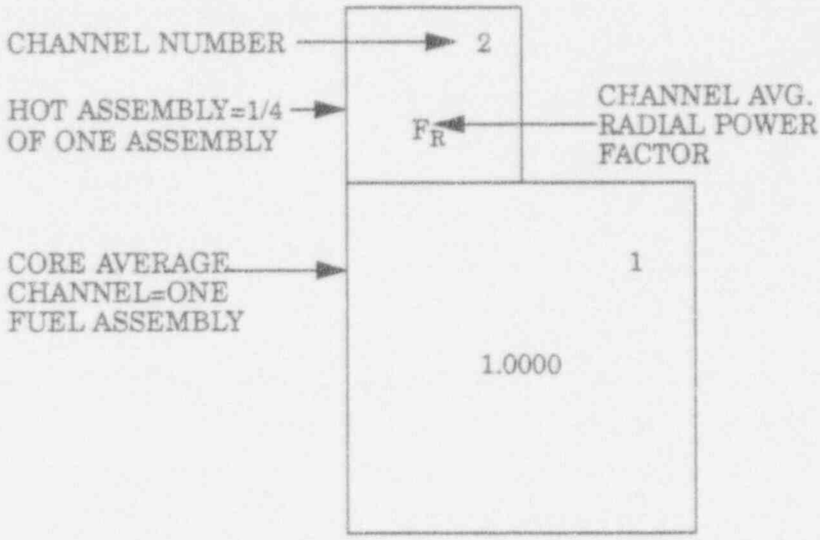
]

3.2.2.2 Major Outputs of the Statistical Combination of Uncertainties Analysis

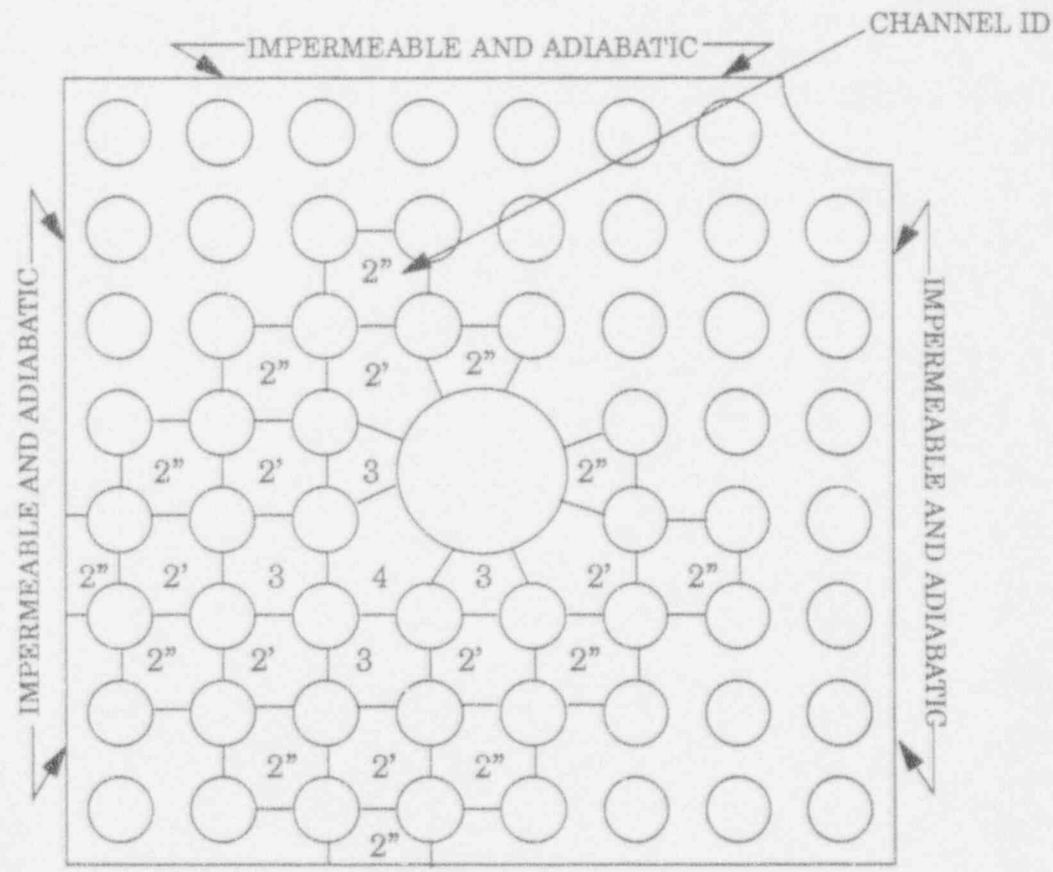
The DNBR PDF and its tolerance limit were provided for use in CPC and COLSS DNBR overall uncertainty analyses (sections 3.5.2 and 3.6.2) using the MSCU method. This PDF included NRC imposed statistical penalties (e.g., a 5% increase on the CHF value) but excluded penalties that are applied deterministically to determine the DNBR limit (e.g., rod bow and grid design penalty).

The DNBR limit with deterministic adjustments was used in calculating the CPC DNBR trip setpoint and the COLSS DNB-Power Operating Limit (POL) alarm setpoint. Additionally, the DNBR limit was used in the safety design, and was cited in the Technical Specifications.

Figure 3.2-1 Channel Geometry of CETOP-D Model



(A) CORE REPRESENTATION



(B) CHANNEL 2 IN DETAIL

Note: See section 3.2.1 for definition of various channel IDs.

3.3 FUEL PERFORMANCE DESIGN

The primary licensing and design objective of the fuel performance design is to evaluate the steady state fuel thermal and mechanical behavior of individual nuclear fuel rods as a function of time or burnup. Generally, this requires generation of representative values for fuel rod temperatures, rod internal gas pressure, and fuel rod deformation. This section describes the fuel performance licensing and design methodology for the PVNGS reload analyses.

The CE fuel evaluation model computer code, FATES, was developed to predict the steady state fuel rod temperature distribution, gap conductance, fuel and clad dimensions, plenum pressure, and stored energy for CE designed fuel. FATES modeled fuel rods consisting of pelletized solid or annular UO_2 fuel encapsulated in a zircaloy cladding tube. The fuel pellet is assumed to be a right circular cylinder. The fuel pellet model accommodates dimensional changes due to fuel relocation, densification, thermal expansion and fission-induced swelling. The fuel is modeled as a collection of discrete axial segments, where an independent radial thermal equilibrium calculation is performed at each segment. The converged results for each segment are coupled to those of other segments through the assumption of complete mixing of the gases within the fuel rod.

An improved fuel performance code, FATES3, contains a new fission gas release model. The NRC completed the safety evaluation and approved the FATES3 model for safety analyses (reference 17), but also imposed a restriction on the grain size used for the fission gas release calculation. This version of FATES, with the grain size restriction, is known as FATES3A. Currently, FATES3A is used for the reload licensing analyses of all PVNGS units. Models contained in FATES3A describe the principal fuel rod behavioral phenomena, including thermal expansion, relocation, densification, creep, swelling, fission gas generation and release, and elastic deformation. Detailed descriptions of the models are contained in references 18 and 19. More recently, FATES3B has been approved generically (reference 20) and will be used in all future analyses. FATES3B has an improved predictive capability at high burnup.

3.3.1 Major Inputs for Fuel Performance

The fuel rod geometric parameters, the actual or projected power history, and the core thermal-hydraulics conditions are required inputs to the FATES3A analysis. The fuel rod input parameters are conservatively biased and included clad length, clad inner and outer diameter, active fuel length, fuel pellet diameter, pellet dish and chamfer dimensions, and fill gas pressure. Additionally, fuel grain size, fuel surface roughness, enrichment, initial and final densities, and the open porosity fraction are required for input.

The power history data is described by the fuel rod operating history. Typically, the core or rod average power data with axial and radial multipliers is used to obtain the LHR at each axial node at every time step. Two power distributions are applied to the composite rod to characterize both long-term and short-term behavior. The composite rod represents several different rods at a given burnup and may represent many different rods during the course of burnup from beginning of life to end of life. The long-term power distribution represents the average value of power over a long burnup interval and is representative of normal operation (i.e., full power, ARO, and equilibrium xenon). It has a nearly flat power distribution axially that is characteristic of the burnup distribution that the fuel rod eventually achieves.

The short-term power distribution is generated using an axial array of multipliers on the long-term power distribution. The short-term power distribution is used in determining fuel temperatures and fission gas release for transient power conditions (i.e., to model control rod motion and xenon oscillations) at each time step. All computed values associated with the transient power calculations are ignored in terms of their effect on the long-term rod performance calculations.

The FATES3A model assumes a closed subchannel with coolant inlet at the bottom of the fuel rod. The reactor core subchannel geometry, coolant inlet enthalpy, coolant pressure, and coolant mass flow are input. An enthalpy balance up the coolant subchannel is performed to obtain the coolant temperature at each axial node. The code either used an input heat transfer coefficient or the Jens-Lottes correlation to calculate a coolant-to-clad heat transfer coefficient. This heat transfer coefficient is used to determine the temperature rise across the film to the clad surface.

3.3.2 Major Outputs for Fuel Performance

The fuel performance analysis performed and data transmitted depends on the specific application. The safety and LOCA analyses typically requires initial fuel rod conditions, maximum rod internal pressure, core minimum and maximum gap conductances, rod minimum gap conductances, the minimum power to fuel centerline melt, the axial densification factor, and the engineering factor on LHR. The data is in various forms, including values calculated directly by FATES3A, FATES3A generated output files used as initial conditions for transient codes, and values calculated external to FATES3A. Additionally, FATES3A is used to verify Technical Specifications on permissible LHRs, peaking factors, and other limits to preclude fuel damage.

3.4 SAFETY ANALYSIS DESIGN

This section discusses the methodology that APS utilizes to analyze transients and events for reloads. A brief overview of the analysis methodology, the codes and highlights of the analyses will be presented. Two types of events are presented in the following section: UFSAR events whose consequences may be adversely affected by changes in reloads and those that are not. The UFSAR Chapter 15 events which are not normally considered during a reload design are discussed in section 3.4.1 and are listed below:

a. Decrease in Feedwater Temperature

b. Increase in Feedwater Flow

c. Loss of Load, Turbine Trip, or Loss of Condenser Vacuum

d. Loss of Non-Emergency AC Power to Station Auxiliaries

e. Loss of Feedwater Flow

f. Feedwater System Pipe Breaks

g. Startup of an Inactive Reactor Coolant Pump

h. Chemical Volume Control System Malfunction - Pressurizer Level Control System Malfunction

i. Inadvertent Operation of the ECCS during Power Operation

j. Pressurizer Pressure Decrease Events

k. Small Primary Line Pipe Break Outside Containment

l. Steam Generator Tube Rupture

The UFSAR Chapter 15 events which are normally considered during a reload design are discussed in section 3.4.2 and are listed below:

a. Fuel Failure Events

   1. Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve

   2. Steam System Piping Failures

   3. Single Reactor Coolant Pump Shaft Seizure/Sheared Rotor (SS/SR)

   4. CEA Ejection

b. Margin Setting Events

1. Total Loss of Forced Reactor Coolant Flow
2. Asymmetric Steam Generator Events
3. Uncontrolled CEA Withdrawal
4. Single CEA Drop Events
5. Part Length CEA Drop
6. CEA Subgroup Drop Events
7. CEA Withdrawal within Deadband

c. Technical Specification Setting Event

1. Inadvertent Boron Dilution

The above events were covered as part of the Reload Technology Transfer Program.

The inputs to the safety analyses are derived from the physics design, core thermal-hydraulics design, fuel performance design, groundrules, and other cycle specific parameters that affect the reload analyses.The major outputs of the safety design are (a) data for the COLSS/ CPCS design, (b) input to the core thermal-hydraulics design, and (c) update of appropriate Technical Specifications.

3.4.1 Events Not Normally Considered in the Reload Analysis

Several of the safety design events presented in the UFSAR are not normally reanalyzed as part of a reload design. These events are not typically reanalyzed since the results presented in the UFSAR are unaffected by reload changes. Although detailed analyses are generally not performed, each event is evaluated to ensure that the current cycle reload parameters are bounded by the assumption of the UFSAR analyses. The following sections contain a brief discussion of the major events of this type, including the criteria used to determine if reanalysis is required.

3.4.1.1 Decrease in Feedwater Temperature

The decrease in feedwater temperature events are typically less limiting than the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV) event because the cooldown that results from this type of event will cause less of a core power increase and therefore, less transient DNBR decrease. The limiting single failure, with respect to fuel performance for this event, is the Loss Of AC power (LOAC) on turbine trip. The decrease in feedwater temperature event, in combination with LOAC, is similar to the IOSGADV event plus LOAC, which is explicitly analyzed as a part of the reload design.

3.4.1.2 Increase in Feedwater Flow

The increase in feedwater flow event is less limiting than the IOSGADV event since the cooldown that results from an increased opening of the feedwater valve or an increase in feedwater pump speed will cause relatively less power increase and therefore, less transient DNBR decrease. The limiting single failure with respect to fuel performance is LOAC following turbine trip. The increase in feedwater flow event, in combination with LOAC, is similar to the IOSGADV event plus LOAC which is explicitly analyzed as a part of the reload design.

3.4.1.3 Loss of Load, Turbine Trip or Loss of Condenser Vacuum

The loss of load, turbine trip or Loss Of Condenser Vacuum (LOCV) analyses presented in the UFSAR were based on generic core physics parameters chosen to bound all future cycles. The LOCV event is the limiting Chapter 15 transient for moderate frequency overpressurization events, for which primary and secondary transient pressures must not exceed 110% of the system design pressures. In such events, the Pressurizer Safety Valve (PSV) and Main Steam Safety Valve (MSSV) lift setpoints limit the primary and the secondary system maximum pressures. Since these are pressurization events, transient DNBR improves. These events are typically not reanalyzed as part of a reload design unless changes occur in the performance characteristics of the valves, the initial power level, or the allowable RCS inlet temperature range.

3.4.1.4 Loss Of Non-Emergency AC Power To Station Auxiliaries

The analysis of this event as presented in the UFSAR concludes that the fuel performance for the LOAC event is no more limiting than that for the loss of coolant flow event. The loss of flow event is typically analyzed each reload.

3.4.1.5 Loss of Feedwater Flow

The loss of feedwater flow is bounded by the LOCV event with respect to RCS pressurization limits. As discussed above, the LOCV event typically is not reanalyzed if the physics parameters bound the reload fuel cycle values. The limiting single failure for this event with respect to fuel performance is bounded by the loss of flow event which is typically analyzed each reload.

3.4.1.6 Feedwater System Pipe Breaks

The feedwater line break transient is the limiting Chapter 15 transient for very low probability overpressurization events, for which primary and secondary transient pressures must not exceed 120% of the system design pressures, and for verification of the adequacy of long-term RCS heat removal. The analysis presented in the UFSAR was based on a limiting 0.2 square foot break size, which will

remain bounding unless the generic kinetics parameters or applicable system parameters change. Similar to the loss of load event, reanalysis is needed if the PSV or MSSV characteristics change. Reanalysis is also required if the deliverable flows from the main or auxiliary feedwater pumps are reduced.

3.4.1.7 Startup of an Inactive Reactor Coolant Pump

The UFSAR analysis of the startup of an inactive reactor coolant pump event considered both heatup and cooldown cases with sufficiently conservative isothermal temperature coefficients to bound future reload designs. The conclusion of this analysis was that these conditions do not result in the loss of the minimum required shutdown margin. Since the shutdown margin is not lost during the event, there is no increase in heat flux and therefore no decrease in minimum DNBR. Also, when the RCS is above the conditions requiring Low Temperature Overpressure Protection (LTOP), the peak RCS pressure will not exceed 110% of design pressure in response to the startup of an inactive reactor coolant pump event. While the RCS is in the LTOP mode, the Shutdown Cooling System (SCS) relief valves will prevent violation of RCS integrity limits (reference 2).

3.4.1.8 Chemical and Volume Control System Malfunction-Pressurizer Level Control System Malfunction

The pressurizer level control system malfunction in combination with a loss of flow event has been identified as the limiting event in the UFSAR for this category. The pressure transient is due to an increase in primary coolant inventory and not due to thermal expansion, as in the section 3.4.1.3 events. The peak pressure reached for the UFSAR case was far less than 110% of the design pressure (i.e. 2750 psia). Additionally, since the RCS pressure increases, the transient DNBR improves. These events are typically not reanalyzed, unless changes occur in the performance characteristics of the PSVs, the initial power level, or the RCS inlet temperature range.

3.4.1.9 Inadvertent Operation of the ECCS During Power Operation

The inadvertent operation of the ECCS event, it is assumed that the two high pressure safety injection pumps actuate and the corresponding discharge valves open. The UFSAR analysis shows that the design limits would not be violated as a result of inadvertent actuation of the ECCS system. Since the shutdown cooling relief valve settings do not change, no reanalysis is needed.

3.4.1.10 Pressurizer Pressure Decrease Events

The inadvertent opening of a PSV event is the most limiting event of this category and is evaluated by the fuel vendor as part of the ECCS analyses.

3.4.1.11 Small Primary Line Pipe Break Outside Containment

In the UFSAR analysis of a small primary line pipe break category, a double-ended break of the letdown line outside the containment and upstream of the letdown line control valve was selected. This event was selected since this break location results in the largest release of reactor coolant outside the containment. The transient MDNBR remains above the DNBR limit. The radiological release rate was based on an assumed, initial coolant activity of 4.6 μCi/gm, corresponding to the maximum equilibrium value that would occur with 1% prior failed fuel. Since this limit is not dependent on reload design parameters, this event is typically not reanalyzed.

3.4.1.12 Steam Generator Tube Rupture

The UFSAR presents the steam generator tube rupture event with and without an assumed LOAC. For the case which includes the LOAC, an additional assumption of a fully stuck open Atmospheric Dump Valve (ADV) was postulated. The latter analysis was performed to determine the radiological doses to the environment and was based on an assumed initial coolant activity of 1.0 μCi/gm, as specified in Technical Specifications section 3.4.7. Since this value is not dependent on reload design parameters, this event is typically not reanalyzed.

3.4.2 Events Normally Analyzed in the Reload Analysis

This section presents the methods used to analyze the events which may change as the result of a reload. Descriptions of the analyses are presented in the following five sub-sections:

1. Section 3.4.2.1 - fuel failure events.
2. Section 3.4.2.2 - margin setting events
3. Section 3.4.2.3 - Technical Specification setting event
4. Section 3.4.2.4 - degraded performance of CPCS & COLSS category
5. Section 3.4.2.5 - verification of transient related CPCS constants.

3.4.2.1 Fuel Failure Events

The four events typically evaluated for fuel failure and dose consequences as part of a reload safety design are: (a) inadvertent opening of a steam generator safety valve or atmospheric dump valve, (b) steam system piping failures, (c) single reactor coolant pump SS/SR and (d) CEA ejection. Each event is discussed separately below.

3.4.2.1.1 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve (IOSGADV)

a. Description of the event

The most limiting combination of an increased heat removal event with a single failure in terms of DNBR is the IOSGADV (or inadvertent opening of a steam bypass valve or MSSV, with a concurrent LOAC). An ADV or a turbine bypass valve may be inadvertently opened by the operator, or a steam bypass control system valve may open due to failure of the associated control system. A steam generator safety valve could remain open only as a result of valve failure. The opening of any of these valves will result in similar consequences because they release steam at the same flow rate. The opening of any of these valves increases the rate of heat removal by steam generators, causing a cooldown of the RCS.

The most limiting single failure is chosen to yield the greatest decrease in DNBR after initiation of a reactor trip signal. In addition, the event is evaluated with the lowest possible pre-trip DNBR. The LOAC concurrent with a turbine trip just after a low DNBR trip represents the most limiting single failure. The loss of RCS flow that results from the LOAC causes a greater decrease in DNBR after a reactor trip than any other possible single failure.

b. Analysis criteria

Since the IOSGADV + LOAC is an infrequent event, violation of the SAFDLs is permissible. This analysis must demonstrate, however, that a coolable core geometry is maintained (i.e., only a small fraction of the fuel fails), and that the radiological consequences do not exceed a small fraction of the 10CFR100 limits.

c. Objectives of the analysis

The objective that only a small fraction of the fuel fails may be met by showing that the predicted fuel failure for this analysis is no larger than that obtained in the previously reported PVNGS analysis or through the performance of a new radiological consequences calculation to show that the event doses do not exceed a small fraction of the 10CFR100 dose limits.

d. Basic assumptions and justifications

The HERMITE and CETOP-D (references 3 and 12) codes have been used to simulate the reactor core response during this transient. The methodology is similar to the loss of flow event methodology described in Appendix 15D of reference 2. In the analysis of this event, it is conservatively assumed that the

initial excess load portion of the event reduces the core DNBR to just above the SAFDL. The LOAC then begins at the SAFDL. This is conservative because a CPCS trip will occur before the DNBR SAFDL is reached, since the CPCS conservatively calculates the actual core MDNBR. Therefore, the actual IOSGADV + LOAC event would be less limiting than a postulated loss of flow event which is initiated at the DNBR SAFDL. Other assumptions that are used in this analysis are listed below.

1. The most conservative Moderator Temperature Coefficient (MTC) is used (i.e., the MTC which produces the most adverse DNBR results for the IOSGADV+LOAC event).
2. A conservative F_r is used, such that the transient initial condition DNBR is at the DNBR limit.
3. A conservative flow fraction versus time function is used to represent the LOAC.
4. Minimum fuel-clad gap conductance is used. This is conservative because a smaller gap conductance delays the heat flux decay due to reactor trip.

e. Analysis method

This transient is initiated at SAFDL conditions as determined using the CETOP-D code. The HERMITE code is then used to produce the time dependent core average heat flux, hot assembly heat flux, and the core inlet coolant mass flux. The CETOP-D code is used to determine the actual transient DNBR and iterates on either [

] The convolution technique is utilized for the fuel failure calculation (reference 21). The radiological consequences are calculated in accordance with reference 22.

If the amount of fuel failure exceeds the previously reported value, the alternatives are, (1) to submit the analysis and results to the NRC for review of the higher fuel failure, provided the dose consequences are bounded by the 10CFR100 limits, or [

]

f. Conservatism of results

The analysis includes the following conservatisms:

1. No credit is taken for the delay between the turbine trip and the LOAC. The MDNBR due to the four pump coast down resulting from the LOAC would be less severe if this delay was included in the simulation.

2. The LOAC portion of the transient is assumed to begin at SAFDL conditions. In actuality, a Variable Overpower Trip (VOPT) would be expected to occur before the SAFDL conditions are reached.

3.4.2.1.2 Steam System Piping Failures

a. Description of the event

A pipe break in the main steam piping increases the steam flow from the affected and unaffected steam generators. This greatly increased steam flow enhances the rate of RCS heat removal by the secondary system. The increased rate of heat removal causes a decrease in core coolant inlet temperature. In the presence of a conservatively assumed negative MTC, the decrease in core inlet temperature causes an increase in core reactivity and results in an increase in core power. The excursion in core power is terminated by the action of one of the following RPS trips: CPCS generated trip, low steam generator pressure, high containment pressure, high linear power level, or VOPT.

Following the reactor trip, the continued decrease in inlet temperature could cause the shutdown margin to be degraded to the point that the core returns to power. Steam line break cases are chosen to maximize the potential for (a) a post-trip return to power, or (b) degradation in fuel cladding performance, which would maximize dose at the site Exclusion Area Boundary (EAB). Of the six cases presented in the UFSAR (reference 2), four cases were chosen to maximize potential for a post-trip return to power and two cases (defined as pre-trip cases) were chosen to maximize potential for degradation in fuel performance and dose at the site EAB.

In the reload design, the two limiting cases are analyzed to determine if a post-trip core return to power will occur. Additionally, the most limiting steam line break outside containment during full power operation is evaluated to determine the worst case fuel performance.

b. Analysis criteria

The steam line break event is a limiting fault event. The results of the analysis must show that:

1. Fission power remains sufficiently low following reactor trip to preclude degradation in fuel performance as a result of post-trip return to power.

2. Degradations in the fuel performance prior to trip is of sufficiently limited extent.

3. The core will remain in place and intact with no loss of core cooling capability.

4. Doses are within the 10CFR100 guidelines.

c. Objectives of the analysis

The objective of the analysis is to demonstrate that the fuel failure for this event does not exceed previously reported values. This ensures that the site boundary doses do not exceed those previously reported. If the amount of fuel failure is greater than previously reported, new dose calculations are required to ensure that the 10CFR100 limits are not violated. If the 10CFR100 limits are violated, the dose consequences are reduced by either providing more margin in COLSS or by obtaining more detailed physics and/or pin census data. In addition, an analysis is performed to ensure that the core does not return to power as a result of the cooldown.

d. Basic assumptions and justification

The analysis assumptions are similar to those given in reference 2. Only those parameters that are reload dependent are discussed here:

1. Assumptions used in the cases chosen to maximize potential for a post-trip return to power:

[ ]

2. Assumptions used in the cases chosen to maximize potential for degradation in fuel performance:

[ ]

e. Analysis method

The Nuclear Steam Supply System (NSSS) response to the steam line break is simulated using the CESEC-III computer program (reference 23). The CETOP-D code (reference 12) is used to determine the MDNBR for the fuel performance case. More details on simulating pre-trip and post-trip cases can be found in Appendix 15C of reference 2.

f. Conservatism of results

Conservatisms are included in the analysis, such as [ ]

3.4.2.1.3 Single Reactor Coolant Pump Sheared Shaft / Seized Rotor

a. Description of the event

A SR event can be caused by seizure of the upper or lower reactor coolant pump thrust journal bearings. A single reactor coolant pump SS could be caused by a mechanical failure of the pump shaft. Following an event of this type, the core flow decreases asymptotically to the 3-pump flow rate. The reduction in coolant flow rate causes an increase in the core average coolant temperature and could result in some fuel pins experiencing DNB.

In the case of a SR event, a reactor trip on pump speed occurs soon after the pump shaft seizes. For a SS event, a reactor trip is generated when the rapid flow reduction causes the pressure drop across the steam generator (in the affected loop) to decrease below the trip setpoint.

b. Analysis criteria

The SS/SR events are limiting fault events, and the corresponding off-site doses must be within 10CFR100 limits.

c. Objective of the analysis

The objective of the analysis is to demonstrate that the fuel failure for this event does not exceed the value previously reported. This ensures that the site boundary doses will not exceed those previously reported. If the 10CFR100 limits are violated, the dose consequences are reduced by [

]

d. Basic assumptions and justification

The analysis assumptions are similar to those given in reference 2. The initial conditions are chosen to maximize the amount of fuel failure. For example:

[

]

[ ]

e. Analysis method

[

] Statistical convolution technique was used to describe the number of failed pins (references 2 and 24).

f. Conservatism of results

A number of conservatisms are included in the analysis, such as: [

]

3.4.2.1.4 CEA Ejection

a. Description of the event

A CEA ejection is postulated by the circumferential rupture of a Control Element Drive Mechanism (CEDM) housing or nozzle. The CEA ejection which results in the most rapid positive reactivity addition is evaluated. A CEA ejection would cause a large increase in the overall power and a highly skewed and severely peaked power distribution. Protection against the effects of a CEA ejection is provided by the inherent Doppler feedback and the VOPT.

b. Analysis criteria

The CEA ejection is classified as a Limiting Fault event. The criteria used are:

1. All fuel pins which are calculated to experience DNB, based on the synthesis method described in reference 25 and utilizing the statistical convolution technique (references 24 and 26), are assumed to fail.

2. All fuel pins with greater than 250 cal/gm centerline enthalpy are assumed to fail.

3. No fuel pins shall exceed 280 cal/gm radial average enthalpy.

4. The peak pressure of the primary and secondary systems are below the emergency condition limits (120% of the design values) as defined in Section III of the ASME Boiler and Pressure Vessel Code (3000 psia and 1500 psia, respectively).

5. The two-hour off-site dose at the EAB is well within 10CFR100 guidelines.

c. Objective of the analysis

The objective of the analysis is to demonstrate that the fuel failure for this event is less than previously reported. This ensures that the site boundary doses are less than those previously reported. If the 10CFR100 limits are violated, the dose consequences are reduced by either providing more margin in COLSS or by obtaining more detailed physics and/or pin census data.

d. Basic assumptions and justification

The analysis assumptions are similar to those given in reference 2. Only those parameters that are reload dependent are discussed below.

[

]

e. Analysis method

[

]

For cases where the ejected rod worth is insufficient for an immediate high flux trip, the CESEC code is used to predict system response due to its inclusion of a secondary system model. The CETOP-D model is used to determine the DNBR time response and the F_r which would cause the DNBR to drop below the SAFDL. Fuel pins which do not meet the specified acceptance crite-

ria (items 1, 2 or 3 of part b, above) are assumed to fail. The convolution technique may be utilized for the fuel failure calculation (references 24 and 30).

f. Conservatism in the analysis

A number of conservatisms are included in the analysis, such as: [

]

3.4.2.2 Margin Setting Events

Many of the events presented in this section have a close relationship with the reactor protection and monitoring systems. The Core Protection Calculator System (CPCS) and Core Operating Limit Supervisory System (COLSS) are key subsystems within the plant protection and monitoring systems. The following text presents a brief discussion of the relationship of the safety design to these systems and provides some key definitions.

The Plant Protection System (PPS) is composed of two subsystems: the Engineered Safety Features Actuation System (ESFAS) and the Reactor Protection System (RPS). [

]

The CPCS has a bidirectional relationship with the safety analyses. In one direction, the analysis of some events determine the values of CPCS setpoints or constants. In the other direction, many transients credit CPCS response to provide protection or mitigate event consequences.

The COLSS is closely related to the Plant Monitoring System (PMS) since the COLSS program is contained in the PMS and receives input from and provides output to other parts of the PMS. It also has a direct relationship to the Technical Specifications, since the system is designed to provide information to assist the reactor operators in monitoring the Limiting Conditions for Operation (LCOs) for LHR margin, azimuthal tilt, DNBR margin and Axial Shape Index (ASI). These LCOs require that if the COLSS is in service, the core power must be less than the COLSS DNBR and LHR Power Operating Limits (POLs), and the COLSS calculated azimuthal tilt and ASI must be within the specified limits. By assisting the operator in maintaining these LCOs, COLSS maintains the initial conditions

assumed in the safety analyses and for the calculation of certain PPS and RPS setpoints, including CPCS constants. Most safety and setpoint analyses assume initial conditions at a COLSS calculated POL in order to assure the limiting condition of minimum initial thermal margin. In addition, the analysis of events[1]
[

]

Thermal margin enters into the relationship between the core monitoring and protection systems (COLSS and CPCS) and reactor operation, since the amount of thermal margin affects the flexibility and power capability of the reactor. In order to understand the importance of thermal margin, it is necessary to define the following:

Available Overpower Margin (AOPM) is defined as the ratio of the power to the DNBR SAFDL at any given combination of the other thermal hydraulic parameters (mass flowrate, RCS pressure, core inlet temperature, radial peaking factor and axial power distribution) to the actual core power at the time point of interest.

Required Overpower Margin (ROPM) is the ratio of the AOPM at the beginning of a transient to the AOPM at the point of minimum DNBR during the transient.

The required initial margin is inherently reserved in the COLSS calculated POL through the use of two methods: [

]

1. i.e., events categorized as anticipated operational occurrences (AOOs).

A number of events are evaluated during each reload to determine the margin requirements, [ ] that are needed to ensure that there is no fuel failure for these events. These events are discussed below.

3.4.2.2.1 Total Loss of Forced Reactor Coolant Flow

a. Description of the event

A loss of flow event may result from a LOAC to one or more of the four Reactor Coolant Pumps (RCPs). As the flow through the core decreases, the system temperature and pressure increase. The loss of flow event is analyzed to determine the minimum initial margin that must be maintained by the Technical Specification LCOs, such that the DNBR SAFDL is not violated during the event. The initial margin is monitored by the COLSS through [ ] as described in section 3.4.2.2.

b. Analysis Criteria

The loss of flow event is an AOO for which the following criteria must be met: (1) the transient MDNBR must be greater than or equal to the DNBR SAFDL, and (2) the peak LHR must be less than or equal to the LHR SAFDL.

c. Objectives of the analysis

The objective of this analysis is to determine the [ ] that will ensure that the analysis criteria is met. The resulting [ ] Since there is no power excursion during the transient, the LHR SAFDL is not challenged during the event.

d. Basic assumptions and justification

In general all initial conditions such as: F_r, ASI, temperature, pressure, etc. are chosen for maximum adverse sensitivity during loss of flow event. The two key reload dependent assumptions that are reassessed are:

[ ]

[

]

e. Analysis method

[

]

f. Conservatism in the analysis

A number of conservatisms are included in the analysis, such as not crediting the pressure increase during the transient and assuming a conservative initial RCS flow rate.

3.4.2.2.2 Asymmetric Steam Generator Events

a. Description of the event

There are four postulated events that could cause an Asymmetric Steam Generator Transient (ASGT) event: (1) loss of load to one steam generator, (2) loss of feedwater flow to one steam generator, (3) excess feedwater to one steam generator, and (4) excess load to one steam generator. Of these four events, the loss of load to one steam generator produces the largest core inlet temperature distortion which results in the largest power distortion and is described below.

The reactor is assumed to be initially operating at full power when both of the Main Steam Isolation Valves (MSIVs) on one of the steam generators instantaneously close, isolating the steam flow from one steam generator. With the loss of load to one steam generator caused by a spurious closure of the MSIV, the pressure and temperature of the RCS increases. Also, the water level of the isolated steam generator rapidly drops as the increasing secondary system pressure and temperature collapse the steam bubbles in the liquid inventory. The pressure could continue to increase until the secondary safety valves open. The pressure of the other steam generator remains steady or drops depending on the mode of operation of the turbine generator control system.

An asymmetry in the core inlet temperature distribution occurs when the temperature in the primary coolant loop associated with the isolated steam generator increases due to a reduction in the primary-to secondary heat transfer rate caused by the termination of steam flow from the isolated steam generator. The core inlet temperature in the primary coolant loop associated

with the unaffected steam generator decreases due to the cooling action of the unaffected generator which "picks up" the load lost by the isolated steam generator.

In the presence of a negative MTC and fuel temperature coefficient, the radial power increase will occur on the cold side of the core where no mixing of the core inlet flows is assumed. The outermost fuel bundles experience the greater increase in radial peak (resulting in a higher hot pin power). A greater increase in the hot pin power occurs since the local MTC at the hot spot is more negative than the core average MTC. This radial peaking factor increase leads to an increase in the peak LHR and a decrease in DNBR which is partially mitigated by the decreasing coolant temperature at the peak location. By contrast, the power on the hot side of the core decreases due to negative reactivity addition arising from moderator temperature effects. Doppler reactivity feedback effects act to restabilize the core and flatten the radial power distribution.

A reactor trip occurs when the asymmetry in cold leg temperatures exceed a setpoint value implemented in the CPCS. Sufficient initial ROPM must be preserved so that the SAFDLs will not be violated for the ASGT type of events.

b. Analysis criteria

The ASGT events are classified as AOOs for which the following criteria must be met: (1) the transient MDNBR must be greater than or equal to the DNBR SAFDL and (2) the peak LHR must be less than or equal to the LHR SAFDL.

c. Objectives of the analysis

The calculated ROPM must be such that the DNBR LCO preserved by the COLSS, in conjunction with the CPCS generated reactor trip, assures that the DNBR and LHR SAFDL are not violated.

d. Basic assumptions and justification

Many of the assumptions in this analysis, such as temperature stratification in the core, instantaneous MSIV closure, pressurizer pressure, and steam bypass control system status are the same as in the UFSAR analysis (reference 2). Other assumptions that are reload related are listed below.

[ ]

e. Analysis method

[ ]

f. Conservatism of results

A number of conservatisms are included in the analysis, such as [ ]

3.4.2.2.3 Uncontrolled CEA Withdrawal

a. Description of the event

A malfunction of the Control Element Drive Mechanism Control System (CEDMCS) or rod regulating system could cause an uncontrolled withdrawal of the CEAs. The core power would increase due to the positive reactivity addition. The power excursion is terminated by the VOPT when the reactor is critical, and by the CPC trip when the reactor is in a subcritical condition prior to the event.

b. Analysis criteria

The uncontrolled CEA withdrawal is a moderate frequency event. Therefore, the fuel must remain within the SAFDLs.

c. Objectives of the analysis

The objective of the analysis is to demonstrate that the fuel remains within the design limits for DNBR and LHR.

d. Basic assumptions and justification

The analysis assumptions are similar to those given in reference 2. Only those parameters that are reload dependent are discussed below.

[ ]

e. Analysis method

[ ]

f. Conservatism of results

[ ]

3.4.2.2.4 Single CEA Drop Events

a. Description of the event

The dropping of a single CEA initially causes a reduction in core power and a primary to secondary side power to load mismatch. This mismatch results in a cooldown of the RCS due to the excess heat removal by the secondary system. In the presence of a negative MTC, the cooldown will add positive reactivity and the core power will tend to return to its pre-drop level. This return to power is accompanied by a power distribution distorted by the full insertion of the CEA. This combination of effects leads to a loss of thermal margin for the CEA drop transient.

The PVNGS CEAs have either 4 or 12 fingers each. [

]

b. Analysis criteria

Single CEA deviation events are AOOs and violation of the SAFDLs during the transient are not allowed.

c. Objective of the analysis

[

]

d. Basic assumptions and justifications

None.

e. Analysis method

[

]

f. Conservatism of results

The most conservative POL derivative in the LCO space is used.

3.4.2.2.5 Part Length CEA Drop

a. Description of the event

The Part Length CEA (PLR) drop event is typically analyzed from a 50 percent initial power condition. Above this power level, the Technical Specifications limit PLR insertion such that the drop of the PLR can only add negative reactivity and the power distribution distortion is less than that of the drop of a single full length CEA. Below 50 percent power, it is possible that dropping a PLR out of an axial region of high power could have the effect of adding positive reactivity to the core. This positive reactivity insertion, along with a constant secondary system demand, could result in the core power generation being higher than that removed by the secondary system. The power to load mismatch will increase RCS temperatures. The RCS heatup, in the presence of a positive MTC, could result in a further increase in core power. The event is postulated to continue until a new quasi-steady state condition is reached due to the action of the secondary safety valves limiting the RCS temperatures.

b. Analysis criteria

Since the PLR drop event is an AOO the fuel must remain within the SAFDLs.

c. Objectives of the analysis

The objective of the analysis is to determine the amount of COLSS ROPM necessary to ensure that there is no violation of the SAFDLs in the event of a PLR drop.

d. Basic assumptions and justifications

Only those assumptions that are reload dependent are discussed here.

e. Analysis method

[

]

f. Conservatisms in the analysis

The maximum F_r is used instead of the post-drop F_r

3.4.2.2.6 CEA Subgroup Drop Events

a. Description of the event

The dropping of a CEA subgroup is much like the dropping of a single CEA, in that the drop inserts negative reactivity into the core, reducing power. Reactivity feedback effects will respond gradually to bring the core power back to its pre-drop level if the secondary system is held at its pre drop level. However, the core power distribution will be greatly distorted by the presence of the CEAs. When the core returns to the initial power, the distorted power distribution may cause the hot channel DNBR to approach the SAFDLs.

With the CEACs operable, a CEA subgroup drop will cause the generation of a penalty factor sufficient to cause a reactor trip before the core approaches violation of the SAFDLs. No analysis is needed since the reactor trips when a CEA subgroup drops to avoid violation of the SAFDLs.

b. Analysis criteria

None.

c. Objectives of the analysis

None.

d. Basic assumptions and justifications

None.

e. Analysis method

None.

f. Conservatism of results

None.

3.4.2.2.7 CEA Withdrawal within Deadband

a. Description of the event

Misalignment within CPC/CEAC deadbands do not generate penalty factors. The deadbands are typically [

] Sufficient thermal margin must be preserved in COLSS to enable the plant to ride through the consequences of a CEA withdrawal within the deadband without resulting in violation of SAFDLs.

The withdrawal of the CEA, combined with a constant secondary system power demand creates a power to load mismatch. The mismatch causes an increase in RCS temperatures and in the presence of a positive moderator reactivity temperature coefficient, the core power levels increase. The addition of positive reactivity would continue until either the Doppler reactivity coefficient added sufficient negative reactivity to halt the power increase, or the action of the secondary safety valves prevented a further increase in the RCS temperatures.

The cause of the thermal margin degradation is a power increase due to the addition of positive reactivity with the withdrawal, an upward shift in the axial power distribution as the power follows the upward motion of the CEAs, and a change in the radial power distribution which will be undetected by the CPCs.

b. Analysis criteria

CEA withdrawal events are AOOs and violation of the SAFDLs during the transient are not allowed. The peak RCS pressure for these events must not exceed the 110% of design value (i. e. 2750 psia).

c. Objective of the analysis

The objective of the analysis is to determine the amount of COLSS ROPM necessary to provide protection against violation of the SAFDLs due to the withdrawal within deadband. The COLSS ROPM is determined to prevent violation of the SAFDLs during the transient.

d. Basic assumptions and justifications

Only those assumptions that are reload dependent are discussed here.

[ ]

e. Analysis method

[ ]

f. Conservatisms in the analysis

All parameters that impact the analyses are assumed at their worst value.

3.4.2.3 Technical Specification Setting Event - Inadvertent Boron Dilution

a. Description of the event

Controlled boron dilution normally occurs during startup. An inadvertent boron dilution event may lead to unplanned changes in reactivity if the dilution is not monitored and controlled. Sufficient time must be available for the reactor operators to take corrective action to prevent criticality if an inadvertent boron dilu-

tion event occurs. A Boron Dilution Alarm System (BDAS) provides the necessary operator warning. When the BDAS is unavailable, Technical Specification surveillance tables are provided.

b. Analysis criteria

An inadvertent boron dilution event is a moderate frequency event. The fuel integrity must be assured by allowing sufficient time for operator action to prevent loss of shutdown margin. The analysis is required to demonstrate acceptable results assuming at least 15 minutes (30 minutes in Mode 6) is required between the time that the operator is made aware of the boron dilution and the loss of shutdown margin. The UFSAR analysis represents the limiting worse case without operator action.

c. Objective of the analysis

The inadvertent boron dilution analysis for a reload has three objectives:

1. Demonstrate that the minimum time to criticality is bounded by the value reported in the UFSAR (reference 2).
2. Demonstrate that the BDAS out-of-service surveillance tables in the Technical Specifications are conservative. A reload specific Technical Specification change may be required.
3. Demonstrate the acceptability of the existing BDAS setpoint. The setpoint is a relative fractional increase in subcritical multiplication after which the alarm annunciates.

d. Basic assumptions and justification

The analysis assumptions are similar to those given in reference 2. Only those parameters that are reload dependent are discussed below.

[ ]

e. Analysis method

The ordinary differential equation which describes a well stirred mixing tank is used for the calculations.

$$M \cdot \frac{dC}{dt} = -(W \cdot C)$$

where, M is RCS water mass
C is RCS boron concentration and
W is charging mass flowrate of unborated water

The equation is evaluated to determine the time necessary for the boron to change from its initial concentration to the critical concentration. All combinations of RCS volume and possible operating modes are examined. The shutdown margin requirements are evaluated and Technical Specification changes are made when the requirements are altered.

f. Conservatism in the analysis

The calculations are based on the assumption that the [

]

3.4.2.4 Degraded Performance of CPCS & COLSS Category

The events described in section 3.4.2.2 are based on the assumption that the CPCS and the COLSS are not in a degraded condition. Since potential hardware failures could affect the operability of these systems, the events must also be evaluated for the following three conditions: (a) COLSS out-of-service and at least one CEAC operable, (b) COLSS in-service and both CEACs inoperable, and (c) COLSS out-of-service and both CEACs inoperable. A brief description of the impact of each of these modes on the reload event analysis is presented in the following sections.

3.4.2.4.1 COLSS Out-of-Service and at Least One CEAC Operable

When COLSS is out-of-service, the CPCS calculated DNBR is used to monitor the DNBR LCO (or COLSS ROPM) per Technical Specification 3.2.4. This specification requires that the CPCS calculated DNBR must be larger than a pre-determined CPCS DNBR vs. ASI limit line. Analyses are performed to determine the ROPM versus power curve for the condition of COLSS out-of-service and CEACs operable. This curve is used to determine a conservative CPCS DNBR limit line (to be used in Technical Specification 3.2.4) which will ensure that the DNBR LCOs are maintained if the core is operated within the Technical Specifications limits. Since the Technical Specifications require a more restrictive PDIL when COLSS is out-of-service, the ROPMs for several events are smaller than those when COLSS is in service because a smaller amount of CEA insertion is allowed. Analysis methods are similar to those given previously.

3.4.2.4.2 COLSS In-Service and Both CEACs Inoperable

When at least one of the CEACs is operable, the CPCS monitors the individual and subgroup CEA movement directly. When both CEACs are inoperable, information on CEA position and the associated radial peaking factors for the DNBR calculation are not provided for the CPCs. As a result additional thermal margin is needed to accommodate CEA misoperation events when the CEACs are inoperable. Typically, the [

]

During periods in which CEACs are inoperable, the Technical Specification on RPS operability requires that the lead bank CEA insertion be restricted to the top 15 percent of the core. As described above the CEACs are unable to provide the CPCs with penalties to compensate for CEA deviations from this configuration. Inward deviation is more than sufficient to compensate for the lack of radial peaking information. Withdrawals from these conditions leads to both increase and a change in the core power distribution which is undetected by the protection system. It is necessary to ensure that sufficient margin is maintained to enable the plant to ride through the transients without violation of SAFDLs. Analysis criteria, objective of the analysis, etc. are similar to the CEA Withdrawal within Deadband presented in section 3.4.2.2.7.

When CEACs are operable, [

] The additional thermal margin which is calculated from this event is implemented in the plant by a Technical Specifications item instructing the plant operators to reduce the POL as calculated by COLSS by a certain amount when both CEACs are inoperable and COLSS is in-service.

As a result, with CEACs inoperable, restrictions are placed upon the allowed insertion of the CEAs (typically limited to top 15 percent). This restriction is found in the RPS operability Technical Specifications. Additional extra margin is maintained at high power levels by forcing the operators to reduce the POL calculated by COLSS per the requirements of Technical Specifications Figure 3.2-1.

3.4.2.4.3 COLSS Out-of-Service and Both CEACs Inoperable

When COLSS is out-of-service and both CEACs are inoperable, the CPCS calculated DNBR is used to monitor the COLSS DNBR LCO. However, when both CEACs are inoperable, CEA movement cannot be detected by the CPCs (similar to section 3.4.2.4.2). As a result, CEA misoperation events must be analyzed for this condition. Again, in the case of COLSS in-service and both CEACs inoperable, a single CEA withdrawal (allowed inserted per Technical Specifications) needs to be protected. These events are used to determine the ROPM versus power curve for the condition of COLSS out-of-service and both CEACs inoperable. This curve is used to determine a conservative CPCS DNBR limit line (to be used in Technical Specification 3.2.4) which ensures that the DNBR LCOs are maintained if the core is operated within the CPCS limits.

Methodology similar to that described previously for CEA misoperation events is used to determine the ROPM when both CEACs are inoperable. The [ ] Therefore, the results obtained for COLSS in-service and CEACs inoperable apply for COLSS out-of-service and CEACs inoperable conditions also. The most restrictive event [ ] results in the maximum ROPM requirements and is used to determine the CPCS DNBR limit line to be used with Technical Specification 3.2.4.

3.4.2.5 Verification of Transient Related CPCS Constants

The CPCS is a set of four digital computers and associated software which initiates two of the RPS trips (see discussion in section 3.5). Each CPC continuously calculates the DNBR and LPD. A trip signal from any two CPC computers will initiate a reactor trip when needed to prevent a violation of the SAFDLs. The CPCS perform the required calculations through the use of algorithms and associated constants and setpoints. The CPCS dynamically process information related to the NSSS parameters that affect the margin to the fuel design limits. The output of the CPCS is the generation, if necessary, of a reactor trip on either low DNBR, high LHR generation, or on several auxiliary trip functions such as cold leg differential temperature. The purpose of the CPCS transient response analysis is to ensure that the CPCS calculations remain conservative relative to the actual system state during postulated system transients. The analyses in this section describe the transient related CPCS constants analyses typically performed each reload.

3.4.2.5.1 Dynamic Compensation

The CPCS is provided with filter algorithms and associated constants which provide dynamic compensation of instrument response. The temperature sensors in particular, have sizable lags associated with dynamic process changes. Because of this, the cold leg temperatures and power signals are dynamically compensated. The coefficients used in the filter calculations are established based on the assumed instrument characteristics and loop transport times. Conservative cold leg temperature instrument time constants and excore detector "shadowing" factors are verified. The rapid temperature changes introduced by overcooling and heatup events are used to verify conservative temperature filter performance. The CEA withdrawal event is typically used for verifying the power filter response.

3.4.2.5.2 Transient Offset Power Penalties

The CPCS allows modification of the received thermal and neutron power signals by fixed multipliers or offset terms. The offset terms ensure that the overall response of the CPCS is conservative by increasing the power signals above the values actually received by the plant sensors. The conditions which have required the applications of power penalties are:

[ ]

During the reload safety design, these types of conditions are evaluated to determine if CPCS power offset adjustments are required.

3.4.2.5.3 Radial Penalty Factor Delay

The CPCS imposes a sizable penalty on the F_r if the drop of a 12-finger CEA is detected. The penalty is such that a reactor trip will normally occur. The CEA positions are monitored by the two CEACs. The penalty factor is imposed immediately if both CEACs simultaneously sense a dropped CEA. If only one CEAC senses a dropped CEA, a delay time is imposed before the penalty is applied. The delay time allows the CEAC to clear any spurious signal, thereby preventing unnecessary reactor trips. The value is based on the loop transport time such that the penalty factor is not applied until the immediate change in power, and therefore temperature, is fed back to the core inlet. This delay time is [

]

3.4.2.5.4 Reactor Coolant Pump Speed Trip Setpoint

An addressable constant is included in the CPCS which represents the fraction of rated RCP speed above which a pump is assumed to be operating. When any RCP speed drops below this setpoint, a reactor trip signal is generated. This setpoint may be used in the IOSGADV+LOAC, loss of flow analyses and seized rotor analyses, and is typically verified for each reload.

3.4.2.5.5 Variable Overpower Trip Setpoint

A VOPT function is implemented in the CPCS as an auxiliary trip. The trip is similar in operation to the analog VOPT in the RPS. Since the VOPT is credited in several transient analyses, the constants associated with the trip function must be reviewed each reload, but generally do not change.

3.4.2.5.6 Asymmetric Steam Generator Trip Setpoint

The analysis of the ASGT event credits a reactor trip generated by the CPCS. The ASGT trip setpoint is based on a cold leg temperature differential as perceived by the CPCS, and is evaluated each reload.

3.4.2.5.7 Reactor Power Cutback Delay Time

The reactor power cutback system is a control system which normally handles load rejections by dropping preselected CEA groups into the reactor and running back the load to match the reactor power. The reactor power cutback delay time is designed to avoid an unnecessary CPC generated reactor trip during a reactor power cutback. The CPCs are designed to ignore CEA deviations for a specific period of time while the plant reestablishes equilibrium after a cutback. This delay time is required to prevent an unnecessary trip since the CEA deviations would normally impose penalty factors if multiple CEAs are perceived to be dropping into the core. The delay time is based on [ ] The resulting power mismatch, when accompanied by negative moderator temperature feedback, would increase reactor power while the power distribution is distorted by the rod configuration.

3.5 CORE PROTECTION CALCULATOR SYSTEM DESIGN

This section includes a brief description of the CPCS and the related analyses performed to support a typical reload design. The CPCS is a set of digital computers and associated software which initiates the DNBR and LPD reactor trip signals. The CPCS is comprised of four Core Protection Calculators (CPCs) and two CEACs. As shown on Figure 3.5-1, each CPC continuously calculates the core DNBR and LPD and will initiate a reactor trip when needed to prevent a violation of the SAFDLs. Each CEAC continuously measures the position of all individual CEAs to detect deviations and provide appropriate penalties to each of the CPCs as shown on Figure 3.5-2. These penalties are applied as appropriate in the DNBR and LPD calculation in the CPCs. The CPCS design basis also includes provisions for auxiliary trip functions, which provide protection for certain design basis events. The auxiliary trips are used to: (a) provide protection more conveniently than the DNBR or LPD trips, (b) aid the CPCS in meeting the primary design basis, and (c) assure CPCS protection in the presence of hardware failures. These auxiliary trip functions are shown on Table 3.5-1.

Table 3.5-1 CPCS Auxiliary Trips

The CPCS (including the six computers, associated software, sensors, and trip outputs) is a safety grade system designed to take action as needed during certain transients. It is primarily designed to provide DNBR and LPD protection for anticipated operational occurrences, but is qualified also to assist the reactor protection system and the ESFAS in limiting the consequences of certain postulated limiting fault events. A list of typical CPCS design basis events are shown below:

a. Anticipated Operational Occurrences:

1. Uncontrolled Axial Xenon Oscillations.

2. CEA Misoperation Events from Critical Conditions, including Single CEA Withdrawal, Single CEA Drop or Insertion, Single CEA Sticking with Remainder of CEAs in that Group Moving, Subgroup Deviation (Insertion, Withdrawal or Drop), Out of Sequence Group Withdrawal or Insertion, Uncontrolled Sequential Group Withdrawal and Excessive Insertion of Part Length CEAs.

3. Excess Heat Removal (Excess Load) Due to Secondary System Malfunctions, including Excess Feedwater Flow, Excess Steam Flow Caused by Inadvertent Opening of Turbine Bypass Valves, Excess Steam Flow Due to Inadvertent Opening of Turbine Control Valves, and Decrease in Feedwater Enthalpy.

4. Loss of Forced Reactor Coolant Flow.

5. RCS Depressurization (Spray Malfunction).

6. Loss of Feedwater Flow.

7. Loss of External Load.

8. Complete Loss of AC Power to the Station Auxiliaries.

9. Uncontrolled Boron Dilution.

10. Asymmetric Steam Generator Transients due to Instantaneous Closure of One MSIV.

b. Postulated Accidents:

1. Reactor Coolant Pump Shaft Seizure.

2. Steam Generator Tube Rupture.

3. Steam Line Break Outside Containment.

The functional design of the CPCS is defined by the description of the algorithms which were created to meet the design basis of the system. The functional design is given in reference 32 for a CPC and in reference 33 for a CEAC. The functional design does not include the constants which are required by the algorithms. These constants are chosen based on analysis that is either generic (such as physical properties), reload fuel cycle independent (such as core configuration), or reload fuel cycle specific (such as power distribution related information).

The CPCS constants are divided into three types: data base, Reload Data Block (RDB), and addressable.

Data base constants are built into the CPCS software along with the CPCS algorithms. Data base constants cannot be changed e... ..pt by changing the software, which involves considerable testing and requires approval of the NRC prior to implementation. For a typical reload, the CPCS software is not changed and therefore, the data base constants are not changed.

RDB constants are located in protected memory of the CPCS and are separate from the algorithms and non RDB data base constants. RDB constants are loaded from a separate disk and can be changed without requiring a CPCS software change (currently all RDB disks are created by the fuel vendor).

Addressable constants can be changed by the plant operator under administrative controls. Addressable constants are changed at various frequencies from once per reload to daily during operation.

Since data base constants are not changed during a reload, the RDB and addressable constants are the only constants which are considered for change during the reload design. The first step in the CPCS reload design analysis is to determine the appropriate values of the RDB constants. Once the RDB constants have been determined, the addressable constants are calculated to ensure that the CPCS design basis is met. Some CPCS constants are defined by other reload design functional areas and other constants are determined based on cycle specific CPCS analyses. In the following sections, the information provided by other functional design areas that is used to determine CPCS constants will be presented first, followed by a discussion of the CPCS RDB and addressable constants analyses.

3.5.1 Physics Design Input to the CPCS Design

Typical physics design inputs to the CPCS reload design process were discussed in Section 3.1 and include a neutronics model, physics related data base constants, and physics related RDB constants. A neutronics model (such as FLAIR) is used as a reactor core simulator for CPCS constants analyses. The physics design also provides [

] This data is used to verify CPCS data base constants which cover for the drop of any single CEA. The RDB constants which are based directly on the physics design include constants such as [

] In most cases, the previous cycle values for these constants are verified as applicable to the upcoming cycle.

3.5.2 Core Thermal-Hydraulics Design Input to the CPCS Design

Core thermal-hydraulics design input to the CPCS reload design process includes a CETOP-D model, associated DNBR limit, and statistical data. As discussed in section 3.2, the CETOP-D model is the design core thermal-hydraulics code benchmarked to the TORC model over various operating ranges. The model is used initially to develop and tune the on-line core thermal-hydraulics algorithms in the CPCS and subsequently the model is used cycle-by-cycle for CPCS constants analyses. Typically, a cycle independent CETOP-D model is provided with a cycle specific penalty to be applied whenever the model is used. The DNBR limit and statistical data provided by the core thermal-hydraulics design includes the mean and standard deviation for the DNBR limit statistics (not including deterministic adjustments). The net DNBR limit with deterministic adjustments is calculated as described in reference 14. The mean and standard deviation is used with its 95/95 probability/confidence tolerance limit in the CPCS analyses as discussed in section 3.2. The DNBR limit with deterministic adjustments is used in calculating the CPC DNBR addressable constant trip setpoint.

3.5.3 Safety Design Input to the CPCS Design

The safety design verifies relevant RDB constants and provides input to calculations of several addressable constants, as described in sections 3.4.2.4 and 3.4.2.5. The values of these constants are selected to ensure that the CPCS will respond conservatively relative to the assumptions made in the analysis of postulated transients. In addition, the safety design defines the protective steps which must be taken to compensate for the potential degraded state of the CPCS and COLSS. Specifically the safety design provides the design margins which must be preserved if the COLSS is out-of-service and/or the CEACs are inoperable. These margins are typically different than the margins provided for normal operation of these systems. The difference is due to the fact that the ROPMs are determined based on more restrictive requirements (such as a reduced Technical Specification PDIL) when the COLSS is out-of-service and/or the CEACs are inoperable.

3.5.4 Fuel Performance Input to the CPCS Design

The fuel performance design provides the [ ] which are used in the CPCS addressable constants analyses. Additionally, the fuel performance design provides the [ ]

3.5.5 Other Input to the CPCS Design

Typically, the Reload Fuel vendor provides the flow measurement uncertainties included in the CPCS addressable constants analyses.

3.5.6 CPCS Constants Analyses

The purpose of the CPCS constants analyses is to verify reload specific data base constants and determine appropriate RDB and addressable constants which ensure that the CPCS design basis is met.

3.5.6.1 CPCS Data Base Constants Verification

For a typical reload the CPCS software is not changed, and therefore, the data base constants are not changed. Because of this, the data base constants were previously chosen to be bounding for all future cycles, if practical. In some cases, it was not possible to determine bounding values without adversely impacting operating space. As a result, some data base constants are evaluated each reload to ensure that they are conservative. One group of data base constants that are typically evaluated each cycle are the [

] These cycle specific values are compared to the penalty factors included in the CPCS data base and the addressable or RDB constants are adjusted as required. The remainder of the CPCS constants analyses involves calculation and/or verification of RDB and addressable constants.

3.5.6.2 Reload Data Block Constants Analyses

In addition to the RDB constants that are determined by the physics, core thermal-hydraulic, fuel performance or safety design, several RDB constants are verified based on other cycle specific requirements of the CPCS. The following design considerations are evaluated each cycle and appropriate RDB constants are adjusted as required.

[ ]

RDB constants change based on these design considerations. An adjustment can be made to addressable constants, if required. These analyses result in a list of RDB constants for use in CPCS design analyses. In addition, this list is used to generate the disk for installation in the CPCS prior to startup.

3.5.6.3 CPCS Addressable Constants Analyses

The CPCS addressable constants analysis is the final step in the CPCS reload design process. The CPCS addressable constants consist of constants measured during startup, calibration constants, trip setpoints, uncertainty factors and penalties. The CPCS addressable constants analyses provide the values for all addressable constants including initial values for the startup related constants. After reload startup testing is completed, final values of these constants are installed. The calibration constants such as flow and power calibration constants are routinely checked and adjusted during operation. The DNBR trip setpoint is determined by the thermal-hydraulic and safety design. The fuel performance design typically determines the trip setpoint for LPD.

The CPCS overall uncertainty analysis provides the final verification of the CPCS functional design and data base. This is done by calculating uncertainty factors which assure the CPCS conservatively calculates the MDNBR and LPD to at least a 95/95 probability/confidence level based on the values of the data base and cycle specific RDB constants. The calculation of the uncertainty factors and penalties is described in detail in reference 14.

After the uncertainty factors and penalties are calculated, the final values of the addressable constants are assembled based on the results of all constant and uncertainty analyses, along with input from other functional areas. The initial addressable constants are installed in the CPCS prior to startup.

Figure 3.5-1 Simplified CPC Functional Block Diagram

Figure 3.5-2 CEAC/CPC Block Diagram

3.6 CORE OPERATING LIMIT SUPERVISORY SYSTEM DESIGN

This section includes a brief description of the COLSS and the related analyses to be performed to support a typical reload design. The COLSS is a digital computer program in the PMS which assists the plant operator in maintaining Technical Specification LCOs. An overview description of the COLSS is provided in reference 34 and a simplified block diagram of the COLSS is shown on Figure 3.6-1. As shown on this figure, the COLSS program monitors the margin to the power operating limits as required by the Technical Specifications for:

a. Peak LHR (or LPD)

b. DNBR

c. Licensed power

Additionally, the COLSS monitors the core azimuthal tilt and ASI to assist the operator in maintaining the Technical Specification LCOs for these parameters. The COLSS design basis specifies that COLSS must provide information to the operator during normal operation. Therefore, there are no design basis events for which COLSS must be applicable. Instead, there are a number of AOOs and postulated events which are analyzed in order to provide initial margin requirements to the COLSS analyses. These events may vary from cycle to cycle due to changes in fuel management, fuel design, or other requirements. The input to the COLSS analyses is based on the choice of AOOs and postulated limiting fault events. The basis for the choice of events to analyze and the methodology for developing the input to COLSS is provided in section 3.4.2.2.

The functional design of the COLSS is defined as the description of the algorithms which were created to meet the design basis of the system. The functional design does not include the values of the constants which are required by the algorithms. These constants are listed in a COLSS data base document which is updated each reload as required.

Their are two types of COLSS constants, data base constants and addressable constants. The COLSS data base constants are specified in a data base document, and are implemented into the COLSS program in the form of a compiled data base file. The COLSS data base constants are typically specified once for each reload fuel cycle and implemented prior to startup. Addressable constants can be changed as required during operation. Similar to the CPCS data base, the COLSS data base contains default values of addressable constants. The default values are updated based on the results of the COLSS analyses and the startup testing. Section 3.6.1 describes the analysis required to determine the appropriate reload specific data base and addressable constants.

The first step in the COLSS reload design analysis is to determine the appropriate values of the data base constants. Once the data base constants have been determined, the addressable constants are calculated. Some of the COLSS constants are defined by other reload design functional areas and other constants are determined based on cycle specific COLSS analyses. The information provided by other functional design areas that is used to determine COLSS constants will be presented first, followed by a discussion of the addressable constants analyses.

3.6.1 Physics Design Input to the COLSS Design

Physics Design inputs to the COLSS reload design typically include a neutronics model and physics related data base constants. A neutronics model (such as FLAIR) is used in the COLSS analyses as a reactor core simulator for COLSS constants analyses. The physics design also provides [

] Data base constants which are based directly on the physics design include [

] In most cases, the previous cycle or implemented values for these constants are verified as applicable to the upcoming cycle.

3.6.2 Core Thermal-Hydraulics Design Input to the COLSS Design

Core Thermal-Hydraulics input to the COLSS reload design typically includes a CETOP-D model, associated DNBR limit, and statistical data. As discussed in section 3.2, the CETOP-D model is the design core thermal-hydraulics code benchmarked to TORC over various operating ranges. It is used initially to develop and tune the on-line core thermal-hydraulics algorithms in COLSS and is used cycle-by-cycle as a base core thermal-hydraulics model for COLSS analyses. Typically, a cycle independent CETOP-D model is provided with a cycle specific penalty to be applied to the core average heat flux whenever the model is used. The DNBR limit and statistical data includes the mean and standard deviation for the DNBR limit statistics (not including deterministic adjustments) along with the net DNBR limit with deterministic adjustments as described in reference 14.

3.6.3 Safety Design Input to the COLSS Design

The safety design provides data base constants, such as: (a) the LHR limit typically based on the LOCA design and (b) the constants related to the Required OverPower Margin (ROPM). The constants related to ROPM include the [

]

3.6.4 Other Inputs to the COLSS Design

Other inputs required for the COLSS reload analyses include [

] The reload fuel vendor currently provides verification of these constants.

3.6.5 COLSS Constants Analyses

This section provides an overview of the COLSS constants analyses for a typical reload design. These analyses verify data base constants and determine addressable constants which ensure that the COLSS design basis is met.

3.6.5.1 COLSS Data Base Constants Verification

Most COLSS data base constants do not change from cycle to cycle. The physics, thermal-hydraulic and safety design provide input to cycle dependent constants as described in the previous sections. This results in a list of data base constants for use in the COLSS addressable constants analyses. In addition, this list is used to generate the data base file for installation in the COLSS prior to startup.

3.6.5.2 COLSS Addressable Constants Analyses

The COLSS addressable constants analysis is the final step in the COLSS reload design process. The COLSS addressable constants consist of constants measured during startup, calibration constants, uncertainty factors and penalties. The COLSS addressable constants analyses provides the default values for all addressable constants including initial values for the startup related constants. After reload startup testing is completed, final values of these constants are installed.

The COLSS overall uncertainty analysis provides the final verification of the COLSS functional design and data base. This is done by calculating uncertainty factors which assure that the COLSS conservatively calculates the DNB-POL and LHR-POL to at least a 95/95 probability/confidence level based on the values of the data base constants. The calculation of the uncertainty factors and penalties is described in detail in reference 14.

After the uncertainty factors and penalties are calculated, the final values of the addressable constants are assembled based on the results of all constant and uncertainty analyses, along with input from other functional areas. The initial addressable constants are installed in the COLSS prior to startup.

Figure 3.6-1 Functional Diagram Of The Core Operating Limit Supervisory System

4.0 COMPARISON OF APS INDEPENDENT ANALYSIS TO CE ANALYSIS

As the final phase of the CE-APS Reload Technology Transfer program, APS prepared a reload design for PVNGS independent of CE. This "Independent Analysis" phase was based on the Unit 3 Cycle 3 core design. Both the APS and CE designs used the Unit 3 Cycle 3 fuel management pattern developed by APS as the starting point. However, the analyzed cycle lengths (burnup windows) were different. The CE analysis was based on a best estimate Cycle 2 schedule: Unit 3 Cycle 2 EOC occurring in the spring of 1991 with a cumulative burnup between 384 EFPD and 436 EFPD. The APS analysis was based on a contingency schedule: delaying the scheduled Unit 3 Cycle 2 EOC until the fall of 1991, with a termination burnup between 436 EFPD and 458 EFPD. This was done in case Unit 3 Cycle 2 did not enter the CE analysis shutdown window in time to avoid a refueling outage in the summer. Ultimately, the Unit 3 Cycle 2 EOC occurred on March 16, 1991, at 390.8 EFPD (within the CE analyzed burnup window).

After the decision was made to use the CE Cycle 3 design, the APS design calculations for the independent analysis were forwarded to CE for a technical review. This was done to verify the ability of APS to correctly implement CE models and methods. In addition to the CE review of the APS calculations, a comparison of the principal results of the two designs was performed by APS. The purpose of this section is to present these comparisons.

4.1 COMPARISON OF PHYSICS DESIGN PRINCIPAL RESULTS

This section compares the APS Unit 3 Cycle 3 physics design principal results to those of CE. Characteristic physics parameters for Cycle 3 are compared in Table 4.1-1.

Table 4.1-1 Unit 3 Cycle 3 Nominal Physics Characteristics

	Units	CE Cycle 3	APS Cycle 3
Dissolved Boron Concentration for Criticality CEAs withdrawn HFP Equilibrium Xenon, BOC	ppm	1153	1082
Boron Worth			
HFP, BOC	ppm/%$\Delta\rho$	119	118
HFP, EOC	ppm/%$\Delta\rho$	91	91
Moderator Temperature Coefficients HFP, Equilibrium Xenon			
BOC	$10^{-4}\Delta\rho/^{\circ}F$	-0.5	-0.7
EOC	$10^{-4}\Delta\rho/^{\circ}F$	-3.2	-3.2
Hot Zero Power, BOC	$10^{-4}\Delta\rho/^{\circ}F$	0.47	0.3
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/^{\circ}F$	-2.1	-2.3
HFP, BOC	$10^{-5}\Delta\rho/^{\circ}F$	-1.7	-1.8
HFP, EOC	$10^{-5}\Delta\rho/^{\circ}F$	-1.9	-2.0
Total Delayed Neutron Fraction, βeff			
BOC		.0062	.0061
EOC		.0051	.0050
Prompt Neutron Generation Time, l*			
BOC	10^{-6} sec	20.9	22.0
EOC	10^{-6} sec	28.2	28.4

Note that the values in Table 4.1-1 are intended to represent nominal core parameters and thus do not contain uncertainties or allowances. The physics data input to safety analyses contain uncertainties and allowances and also may contain extra conservatism to bound future operating cycles.

This comparison shows that the results of the CE and APS analyses were identical for some parameters and slightly different for others. All differences can be attributed to the different shutdown windows assumed in the analyses. For example:

a. The APS BOC HFP Critical Boron Concentration (CBC) was lower than the CE value due to the relatively lower reactivity of the fuel being held over from Cycle 2.

b. The APS BOC HFP Boron Worth was less than the CE value due to the lower HFP CBC.

c. The APS BOC MTCs were more negative than the CE values due to the lower CBCs.

d. The APS Doppler Coefficients were slightly more negative than the CE values due to the relatively higher burnup of the fuel being held over from Cycle 2.

e. The APS βeffs were slightly lower than the CE values due to the higher core average burnup as a result of the relatively higher burnup of the fuel being held over from Cycle 2.

f. The APS BOC prompt neutron lifetime (l*) was higher than the CE value due to the lower BOC CBC.

g. The EOC l*s were virtually the same.

Table 4.1-2 presents a comparison of the summary of CEA reactivity worths and allowances for the EOC3 full power Steam Line Break (SLB) transient.

Table 4.1-2 Unit 3 Cycle 3 Limiting Values of Reactivity Worths and Allowances for HFP Steam Line Break, $\%\Delta\rho$, EOC

	CE Cycle 3	APS Cycle 3
Worth of all CEAs Inserted	-17.5	-17.5
Stuck CEA Allowance	3.8	3.8
Worth of all CEAs Less Highest Worth CEA Stuck Out	-13.7	-13.7
Full PDIL CEA Worth	0.25	0.25
Calculated Scram Worth	-13.5	-13.5
Physics Uncertainty	1.3	1.3
Other Allowances	1.3	1.3
Net Available Scram Worth	-10.9	-10.9
Scram Worth Used in Safety Analyses	-10.2	-10.2

The full power SLB is typically chosen to illustrate differences in CEA reactivity worths for two different cycles. As is seen from this comparison, the EOC CEA reactivity worths were identical.

Table 4.1-3 shows a comparison of the reactivity worths of various CEA groups calculated at full power conditions.

Table 4.1-3 Unit 3 Cycle 3 Reactivity Worth of CEA Regulating Groups at HFP, %$\Delta\rho$

Regulating CEAs	BOC CE Cycle 3	BOC APS Cycle 3	EOC CE Cycle 3	EOC APS Cycle 3
Group 5	-0.33	-0.34	-0.41	-0.42
Group 4	-0.33	-0.33	-0.35	-0.34
Group 3	-0.93	-0.94	-1.13	-1.13

Note: Values shown assume sequential group insertion.

From this comparison, the CEA group reactivity worths were virtually identical.

Figures 4.1-1 and 4.1-2 compare the BOC and EOC ARO assembly Relative Power Densities (RPD) during Cycle 3 for the APS and CE designs. The one-pin Planar Radial Power Peaks (F_{xy}) presented in these figures represent the maximum values over the middle eighty percent of the core (axially). Time points at the BOC and EOC were chosen to display the variation in assembly and maximum planar radial peaking as a function of burnup. The maximum difference between the APS and CE calculated RPDs is 0.02 RPD Units. This maximum difference was typical of that seen due to the difference in previous cycle termination burnups.

Figures 4.1-3 and 4.1-4 compare the BOC and EOC RPDs for the bank 5 rodded configuration for the APS and CE designs. The maximum difference between the APS and CE calculated bank 5 RPDs is 0.03 RPD Units. This maximum difference was typical of that seen due to the difference in previous cycle termination burnups

The radial power distributions described in this section are calculated data which do not include any uncertainties or allowances. The calculations performed to determine these radial power peaks explicitly account for augmented power peaking, which is characteristic of fuel rods adjacent to the water holes.

In addition to the RPD and F_{xy}s the axial peaking factors of the APS and CE designs were compared. The nominal APS and CE calculated axial peaking factors range from a maximum of 1.22 at BOC3 to a minimum of 1.08 towards EOC3 and they agree well.

In summary, the comparison of the physics data for the APS and CE designs shows that the differences were minimal and could be attributed to the difference in previous cycle burnup.

Figure 4.1-1 ARO Assembly Relative Power Densities (BOC)

Assembly #
APS RPD
CE RPD
Difference

	APS	CE
F_{xy} =	1.53	1.52
Assembly =	47	47

					1 0.29 0.29 0.00	2 0.41 0.41 0.00	3 0.57 0.58 0.01	4 0.78 0.77 -0.01
			5 0.29 0.29 0.00	6 0.50 0.50 0.00	7 0.96 0.95 -.01	8 0.99 0.99 0.00	9 1.17 1.17 0.00	10 0.93 0.93 0.00
		11 0.47 0.47 0.00	12 0.77 0.78 0.01	13 1.16 1.16 0.00	14 1.12 1.13 0.01	15 1.27 1.27 0.00	16 0.98 0.98 0.00	17 1.24 1.24 0.00
	18 0.29 0.29 0.00	19 0.78 0.78 0.00	20 1.15 1.16 0.01	21 1.07 1.08 0.01	22 1.29 1.30 0.01	23 1.22 1.22 0.00	24 1.31 1.31 0.00	25 1.17 1.17 0.00
	26 0.50 0.50 0.00	27 1.16 1.16 0.00	28 1.07 1.08 0.01	29 1.25 1.25 0.00	30 1.07 1.07 0.00	31 1.34 1.34 0.00	32 1.12 1.12 0.00	33 1.30 1.30 0.00
34 0.29 0.29 0.00	35 0.96 0.95 -.01	36 1.12 1.13 0.01	37 1.29 1.30 0.01	38 1.07 1.07 0.00	39 0.96 0.96 0.00	40 1.07 1.07 0.00	41 1.29 1.28 -0.01	42 1.03 1.02 -0.01
43 0.41 0.41 0.00	44 0.99 0.99 0.00	45 1.27 1.27 0.00	46 1.22 1.22 0.00	47 1.34 1.34 0.00	48 1.07 1.07 0.00	49 1.33 1.32 -0.01	50 1.16 1.16 0.00	51 1.27 1.26 -0.01
52 0.57 0.58 0.01	53 1.17 1.17 0.00	54 0.98 0.98 0.00	55 1.31 1.31 0.00	56 1.12 1.12 0.00	57 1.29 1.28 -0.01	58 1.16 1.16 0.00	59 1.25 1.25 0.00	60 1.03 1.03 0.00
61 0.78 0.77 -0.01	62 0.93 0.93 0.00	63 1.24 1.24 0.00	64 1.17 1.17 0.00	65 1.30 1.30 0.00	66 1.03 1.02 -0.01	67 1.27 1.26 -0.01	68 1.03 1.03 0.00	69 0.92 0.91 -0.01

APS and CE Calculated
ARO Assembly Relative Power Densities
at HFP BOC3 with Eq. Xe.

Figure 4.1-2 ARO Assembly Relative Power Densities (EOC)

Assembly #
APS RPD
CE RPD
Difference

	APS	CE
F_{xy} =	1.53	1.51
Assembly =	63	63

					1 0.37 0.38 0.01	2 0.49 0.50 0.01	3 0.63 0.64 0.01	4 0.80 0.80 0.00
			5 0.36 0.37 0.01	6 0.55 0.56 0.01	7 1.00 1.00 0.00	8 0.99 0.99 0.00	9 1.19 1.18 -.01	10 0.96 0.96 0.00
		11 0.52 0.53 0.01	12 0.79 0.80 0.01	13 1.12 1.12 0.00	14 1.03 1.04 0.01	15 1.28 1.27 -.01	16 1.00 1.00 0.00	17 1.37 1.35 -.02
	18 0.36 0.37 0.01	19 0.79 0.81 0.02	20 1.13 1.14 0.01	21 0.98 0.99 0.01	22 1.11 1.10 -.01	23 1.11 1.11 0.00	24 1.35 1.33 -.02	25 1.16 1.15 -.01
	26 0.55 0.56 0.01	27 1.12 1.12 0.00	28 0.98 0.99 0.01	29 1.21 1.21 0.00	30 0.97 0.98 0.01	31 1.26 1.25 -.01	32 1.10 1.09 -.01	33 1.37 1.35 -.02
34 0.37 0.38 0.01	35 1.00 1.00 0.00	36 1.03 1.04 0.01	37 1.11 1.10 -.01	38 0.97 0.97 0.00	39 0.91 0.92 0.01	40 1.02 1.02 0.00	41 1.35 1.33 -.02	42 1.08 1.08 0.00
43 0.49 0.50 0.01	44 0.99 0.99 0.00	45 1.28 1.27 -.01	46 1.11 1.11 0.00	47 1.26 1.25 -.01	48 1.02 1.02 0.00	49 1.30 1.29 -.01	50 1.15 1.15 0.00	51 1.37 1.35 -.02
52 0.63 0.64 0.01	53 1.19 1.18 -.01	54 1.00 1.00 0.00	55 1.35 1.33 -.02	56 1.10 1.10 0.00	57 1.35 1.33 -.02	58 1.15 1.14 -.01	59 1.34 1.32 -.02	60 1.07 1.07 0.00
61 0.80 0.80 0.00	62 0.96 0.96 0.00	63 1.37 1.35 -.02	64 1.17 1.15 -.02	65 1.37 1.35 -.02	66 1.08 1.08 0.00	67 1.37 1.35 -.02	68 1.07 1.07 0.00	69 0.97 0.97 0.00

APS and CE Calculated
ARO Assembly Relative Power Densities
at HFP EOC3 with Eq. Xe.

Figure 4.1-3 CEA Group 5 Assembly Relative Power Densities (BOC)

Assembly # APS RPD CE RPD Difference					1 0.30 0.30 0.00	2 0.42 0.42 0.00	3 0.59 0.59 0.00	4 0.81 0.81 0.00
			5 0.31 0.31 0.00	6 0.52 0.52 0.00	7 1.02 1.02 0.00	8 1.04 1.05 0.01	9 1.22 1.22 0.00	10 0.96 0.96 0.00
		11 0.51 0.51 0.00	12 0.84 0.85 0.01	13 1.27 1.28 0.01	14 1.21 1.21 0.00	15 1.33 1.33 0.00	16 0.97 0.97 0.00	17 1.21 1.22 0.01
	18 0.31 0.31 0.00	19 0.85 0.85 0.00	20 1.28 1.28 0.00	21 1.17 1.17 0.00	22 1.40 1.40 0.00	23 1.24 1.24 0.00	24 1.23 1.23 0.00	25 1.02 1.02 0.00
	26 0.52 0.52 0.00	27 1.27 1.28 0.01	28 1.17 1.17 0.00	29 1.35 1.35 0.00	30 1.11 1.11 0.00	31 1.33 1.33 0.00	32 0.97 0.96 -0.01	33 0.75 0.75 0.00
34 0.30 0.30 0.00	35 1.02 1.02 0.00	36 1.20 1.21 0.01	37 1.40 1.40 0.00	38 1.11 1.11 0.00	39 0.95 0.94 -0.01	40 1.04 1.03 -0.01	41 1.17 1.17 0.00	42 0.85 0.85 0.00
43 0.42 0.42 0.00	44 1.04 1.05 0.01	45 1.33 1.33 0.00	46 1.24 1.24 0.00	47 1.33 1.33 0.00	48 1.04 1.03 -0.01	49 1.30 1.29 -0.01	50 1.11 1.10 -0.01	51 1.20 1.19 -0.01
52 0.59 0.59 0.00	53 1.22 1.22 0.00	54 0.97 0.97 0.00	55 1.23 1.23 0.00	56 0.97 0.96 -0.01	57 1.17 1.17 0.00	58 1.10 1.10 0.00	59 1.22 1.21 -0.01	60 1.00 0.99 -0.01
61 0.81 0.81 0.00	62 0.96 0.96 0.00	63 1.22 1.22 0.00	64 1.02 1.02 0.00	65 0.76 0.75 -0.01	66 0.85 0.85 0.00	67 1.20 1.19 -0.01	68 0.99 0.99 0.00	69 0.87 0.86 -0.01

	APS	CE
F_{xy} =	1.59	1.60
Assembly =	27	13

APS and CE Calculated
Assembly Relative Power Densities at HFP
BOC3 w/ Bk5 Inserted w/ ARO Eq. Xe.

Figure 4.1-4 CEA Group 5 Assembly Relative Power Densities (EOC)

Assembly # APS RPD CE RPD Difference

	APS	CE
F_{xy} =	1.49	1.48
Assembly =	63	17

					1 0.39 0.39 0.00	2 0.51 0.51 0.00	3 0.66 0.67 0.01	4 0.85 0.85 0.00
			5 0.39 0.39 0.00	6 0.60 0.60 0.00	7 1.11 1.09 -.02	8 1.06 1.05 -.01	9 1.27 1.25 -.02	10 1.00 1.00 0.00
		11 0.58 0.58 0.00	12 0.89 0.88 -.01	13 1.26 1.24 -.02	14 1.11 1.12 0.01	15 1.37 1.35 -.02	16 1.00 1.00 0.00	17 1.39 1.37 -.02
	18 0.39 0.39 0.00	19 0.89 0.89 0.00	20 1.30 1.28 -.02	21 1.07 1.08 0.01	22 1.17 1.17 0.00	23 1.10 1.12 0.02	24 1.27 1.26 -.01	25 1.00 1.00 0.00
	26 0.60 0.60 0.00	27 1.26 1.24 -.02	28 1.07 1.08 0.01	29 1.32 1.31 -.01	30 0.98 0.99 0.01	31 1.23 1.23 0.00	32 0.91 0.92 0.01	33 0.75 0.75 0.00
34 0.39 0.39 0.00	35 1.11 1.09 -.02	36 1.11 1.11 0.00	37 1.17 1.17 0.00	38 0.98 0.99 0.01	39 0.88 0.90 0.02	40 0.95 0.97 0.02	41 1.21 1.21 0.00	42 0.87 0.89 0.02
43 0.51 0.51 0.00	44 1.06 1.05 -.01	45 1.37 1.35 -.02	46 1.10 1.11 0.01	47 1.23 1.23 0.00	48 0.95 0.97 0.02	49 1.25 1.25 0.00	50 1.07 1.09 0.02	51 1.28 1.29 0.01
52 0.66 0.67 0.01	53 1.27 1.25 -.02	54 1.00 1.00 0.00	55 1.27 1.26 -.01	56 0.91 0.92 0.01	57 1.21 1.21 0.00	58 1.07 1.09 0.02	59 1.30 1.31 0.01	60 1.01 1.03 0.02
61 0.85 0.85 0.00	62 1.00 1.00 0.00	63 1.39 1.37 -.02	64 1.00 1.00 0.00	65 0.75 0.75 0.00	66 0.87 0.89 0.02	67 1.28 1.29 0.01	68 1.01 1.03 0.02	69 0.90 0.93 0.03

APS and CE Calculated
Assembly Relative Power Densities at HFP
EOC3 w/ Bk5 Inserted w/ ARO Eq. Xe.

4.2 COMPARISON OF CORE THERMAL-HYDRAULICS DESIGN PRINCIPAL RESULTS

This section compares the APS independent Unit 3 Cycle 3 core thermal-hydraulics design principal results to those of CE. Core thermal-hydraulics parameters for Cycle 3 are compared in Table 4.2-1.

Table 4.2-1 Unit 3 Cycle 3 Core Thermal Hydraulics Parameters at Full Power

General Characteristics	Units	CE Cycle 3	APS Cycle 3
Total Heat Output (Core only)	MWt 10^6 Btu/hr	3800 12,970	3800 12,970
Fraction of eat Generated in Fuel Rod		0.975	0.975
Primary System Pressure Nominal	psia	2250	2250
Inlet Temperature (Nominal)	°F	565.0	565.0
Total Reactor Coolant Flow++ (Minimum Steady State)	gpm 10^6 lb/hr	423,300 155.8	423,300 155.8
Coolant Flow Through Core (Minimum)	10^6 lb/hr	151.1	151.12
Hydraulic Diameter (Nominal Channel)	ft	0.039	0.039
Average Mass Velocity	10^6 lb/hr-ft^2	2.49	2.49
Pressure Drop Access Core (Minimum Steady State Flow Irreversible ΔP Over Entire Fuel Assembly)	psi	14.5	14.5
Total Pressure Drop Across Vessel (Based On Nominal Dimensions and Minimum Steady State Flow)	psi	51.3	51.3
Core Average Heat Flux (Accounts for Fraction of Heat Generated in Fuel Rod and Axial Densification Factor)	Btu/hr-ft^2	185,900*	185,900*
Total Heat Transfer Area (Accounts for Axial Densification Factor)	ft^2	68,000*	68,000*
Film Coefficient at Average Conditions	Btu/hr-ft^2°F	6100	6100
Average Film Temperature Difference	°F	30.5	30.5
Average LHR of Undensified Fuel Rod (Accounts for Fraction of Heat Generated in Fuel Rod)	kw/ft	5.44	5.44
Average Core Enthalpy Rise	Btu/lb	85.9	85.9

Table 4.2-1 Unit 3 Cycle 3 Core Thermal Hydraulics Parameters at Full Power

General Characteristics	Units	CE Cycle 3	APS Cycle 3
Maximum Clad Surface Temperature	°F	656.0	656.7
Engineering Heat Flux Factor		1.03+	1.03+
Engineering Factor on Hot Channel Heat Input		1.03+	1.03+
Rod Pitch, Bowing and Clad Diameter Factor		1.05+	1.05+
Fuel Densification Factor (Axial)		1.002	1.002

Notes:

- * Based on 2352 poison rods.
- \+ These factors have been combined statistically with other uncertainty factors as described in reference 14 to define overall uncertainty penalty factors applied in the DNBR calculations in COLSS and CPC. These factors were used in conjunction with the appropriate DNBR limit to provide assurance at the 95/95 probability/confidence level that the hot pin would not experience DNB.
- ++ Technical Specification minimum flow rate.

This comparison shows that the results of the CE and APS analyses were identical for all parameters with the exception of maximum clad surface temperature, which was due to numerical round-off. The same CETOP-D model from PVNGS Unit 1 Cycle 2 was used by both CE and APS for these analyses, with the calculated penalties on overpower as listed in Table 4.2-2:

[

Table 4.2-2 Overpower Penalties

]

The results were essentially the same. The slight discrepancy in the wide range overpower penalty results from a lower range of flow for the CE analysis at 74% design flow versus the 75% design flow used in the APS calculation.

[

]

4.3 COMPARISON OF FUEL PERFORMANCE DESIGN PRINCIPAL RESULTS

This section compares the APS independent Unit 3 Cycle 3 fuel performance analysis results to those of CE. The hot pin average temperature, centerline temperature, and pellet to clad gap conductance are shown on Table 4.3-1 and are plotted on Figures 4.3-1 through 4.3-3. All APS and CE results for Table 4.3-1 were taken from the same time in life (BOC) and peak power of 12.16 kw/ft.

Table 4.3-1 Hot Pin Data at Peak Power (BOC)

Avg Pin Burnup	Tavg @ Peak Power (°F)		Tcl @ Peak Power (°F)		Hgap @ Peak Power (Btu/hr-ft^2°F)	
(MWd/MTU)	APS	CE	APS	CE	APS	CE
0	1930	1928	2988	2983	1561	1560
50	1940	1938	2997	2992	1513	1511
100	1956	1953	3013	3008	1457	1454
500	2005	2004	3062	3059	1275	1275
1000	2014	2012	3070	3065	1244	1242
2000	2011	2009	3064	3059	1247	1245
4000	**2019**	**2017**	**3070**	**3066**	**1224**	**1220**
6000	1985	1983	3030	3026	1315	1311
8000	1934	1933	2970	2966	1461	1455

The comparison is made at BOC because the pin temperatures were highest early in life. The maximum temperature results occurred at the same pin average burnup of 4000 MWd/MTU for APS and CE analyses. As can be seen from the comparison, the gap conductance and temperature results were nearly the same.

Table 4.3-2 shows extreme values of fuel average and centerline temperatures and gap conductance as a function of power for the average pin. The maximum fuel average temperature and centerline temperatures and minimum gap conductance values occurred early in life. As can be seen from this comparison, the gap conductance and temperature results at various power for the average pin were comparable.

Table 4.3-2 Extreme Values for Average Pin (BOC)

Power(kw/ft)		Max Tavg (°F)		Max Tcl (°F)		Min Hgap (Btu/hr-ft^2°F)	
APS	CE	APS	CE	APS	CE	APS	CE
4.45	4.53	1293	1301	1584	1599	510	515
5.03	5.09	1305	1311	1638	1649	524	528
6.90	6.91	1526	1526	2033	2034	719	719
7.36	7.41	1531	1534	2075	2082	737	741
8.03	8.09	1619	1624	2233	2244	842	850
8.33	8.43	1619	1627	2258	2274	845	857
8.58	8.61	1662	1664	2330	2335	910	915
8.75	8.81	1657	1660	2338	2347	905	914
8.93	8.93	1686	1686	2391	2391	938	939

Table 4.3-3 and Figure 4.3-4 show the rod internal pressures for the hot pin as a function of average pin burnup starting with 25,000 MWd/MTU. From 0 to 24,000 MWd/MTU, the F_r was at 1.00 for both APS and CE analyses, which resulted in comparable pin internal pressure. The comparison shows that the APS calculated pin internal pressure was higher than the CE calculated pressure except for the burnup intervals of 25,000 to 30,000 MWd/MTU and 37,000 to 40,000 MWd/MTU. This was a result of the following differences:

1. The CE radial fall-off curve (normalized to the peak at each time point in the cycle as a function of rod average burnup) was based on both a different end-point of the previous cycles and improved cross sections which tended to flatten radial peaks.

2. APS conservatively delayed the final drop of the radial fall-off curve to 46,000 MWd/MTU versus 44,000 MWd/MTU for CE.

3. The above effect was amplified since the gas release at high burnup is sensitive to power level due to the grain size restriction. Despite the large differences in the rod internal pressure at 46,000 MWd/MTU, the APS result was within the design limit.

Table 4.3-3 Rod Internal Pressure for Hot Pin

AVERAGE PIN BURNUP	PIN INTERNAL PRESSURE (PSIA)	
(MWD/MTU)	APS	CE
25,000	1582	1607
27,000	1360	1432
30,000	1404	1488
32,000	1438	1416
35,000	1491	1439
37,000	1450	1482
40,000	1533	1613
42,000	1626	1578
44,000	1758	1652
46,000	2059	1406
48,00	1646	1506
49,000	1719	1645

In addition to the fuel pin temperatures and pressure comparisons described above, the following comparisons were made.

a. Both APS and CE calculations showed that the power to fuel centerline melt was greater than 21.0 kw/ft.

b. The generic maximum and minimum case average gap conductance values for CE 16x16 fuel were appropriate for use in both the APS and CE Unit 3 Cycle 3 safety design.

c. The engineering factor on LHR was 1.03 and the axial densification factor was less than 1.002.

Figure 4.3-1 Fuel Pin Average Temperature at Peak Power versus Average Pin Burnup

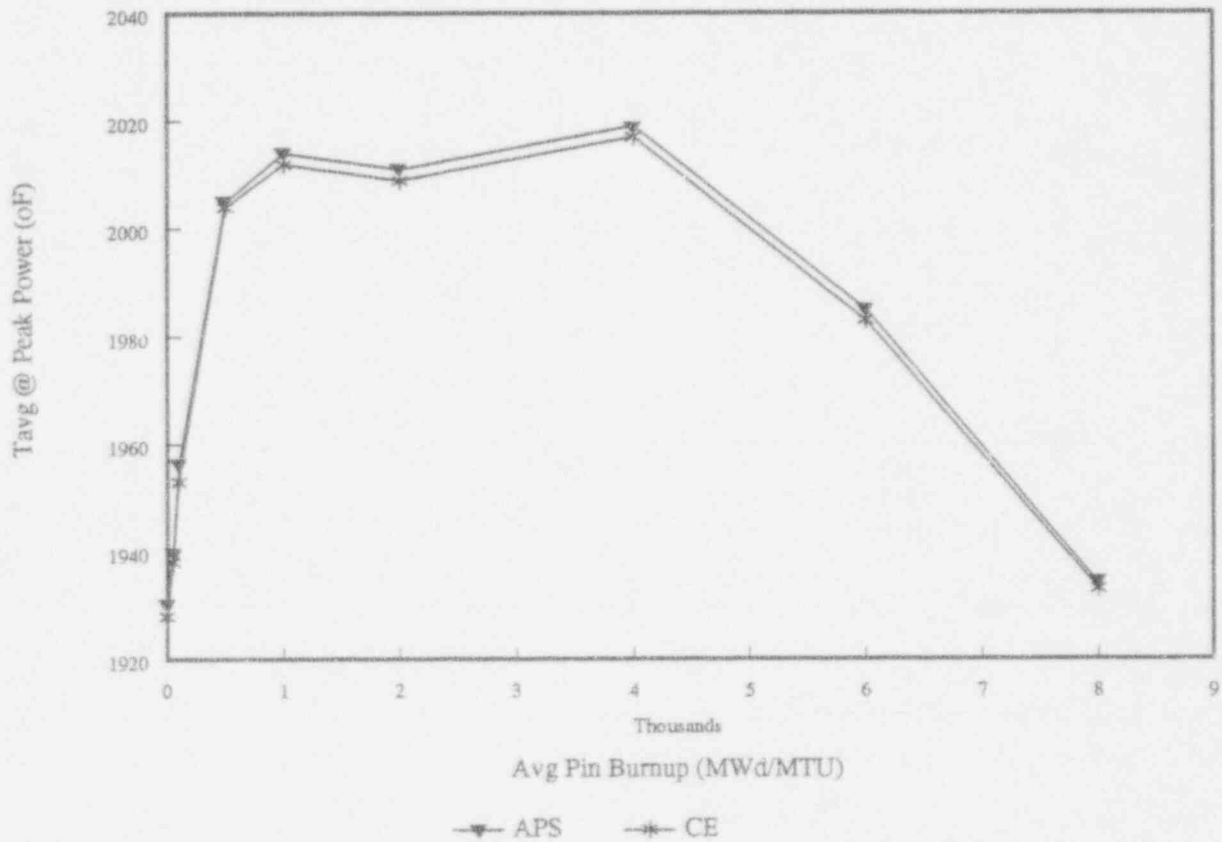


Figure 4.3-2 Fuel Pin Centerline Temperature at Peak Power versus Average Pin Burnup

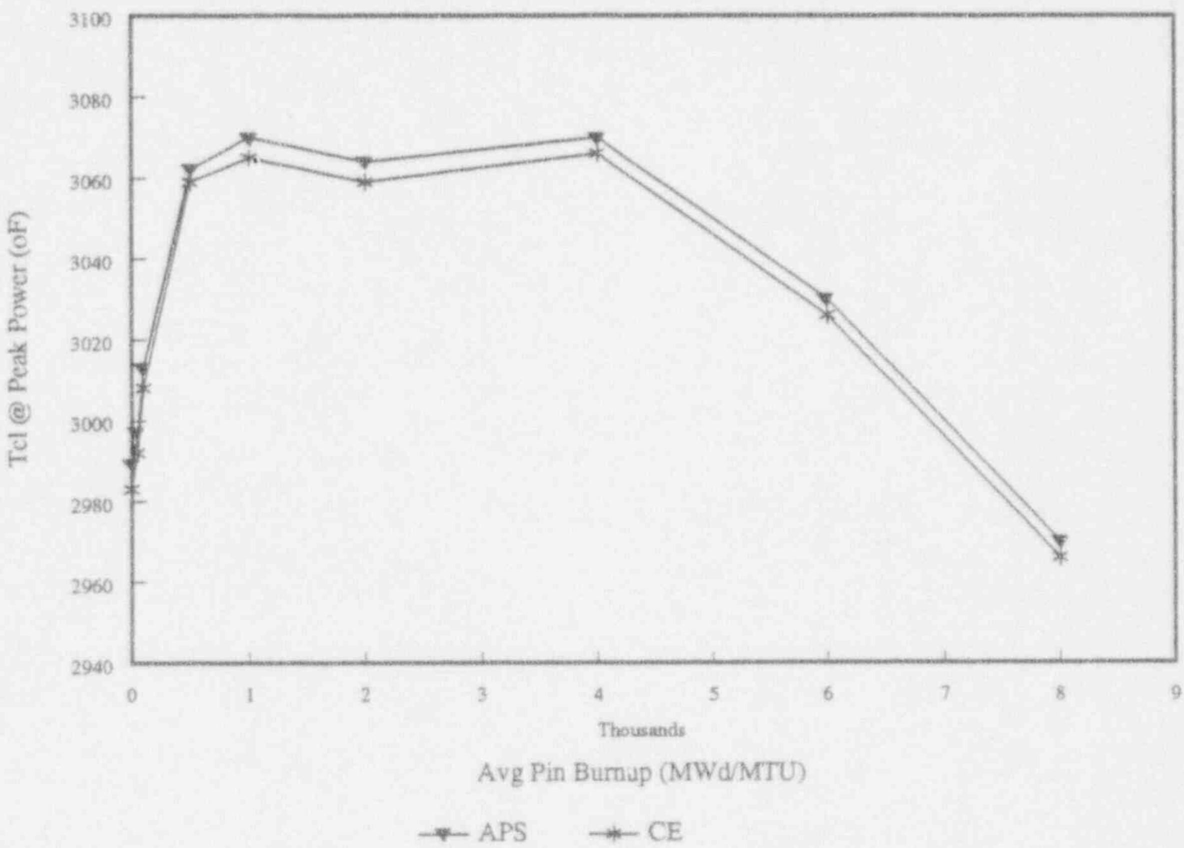


Figure 4.3-3 Fuel Pin Minimum Gap Conductance at Peak Power versus Average Pin Burnup

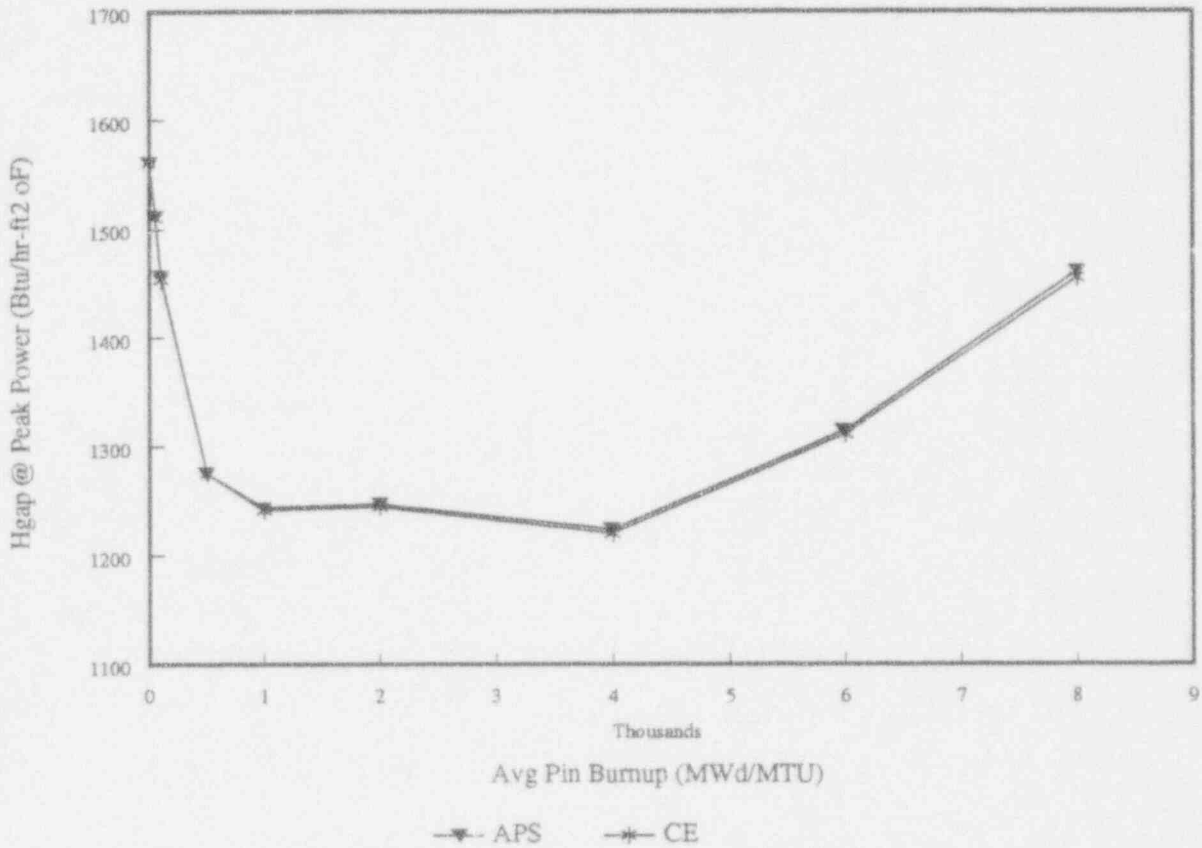
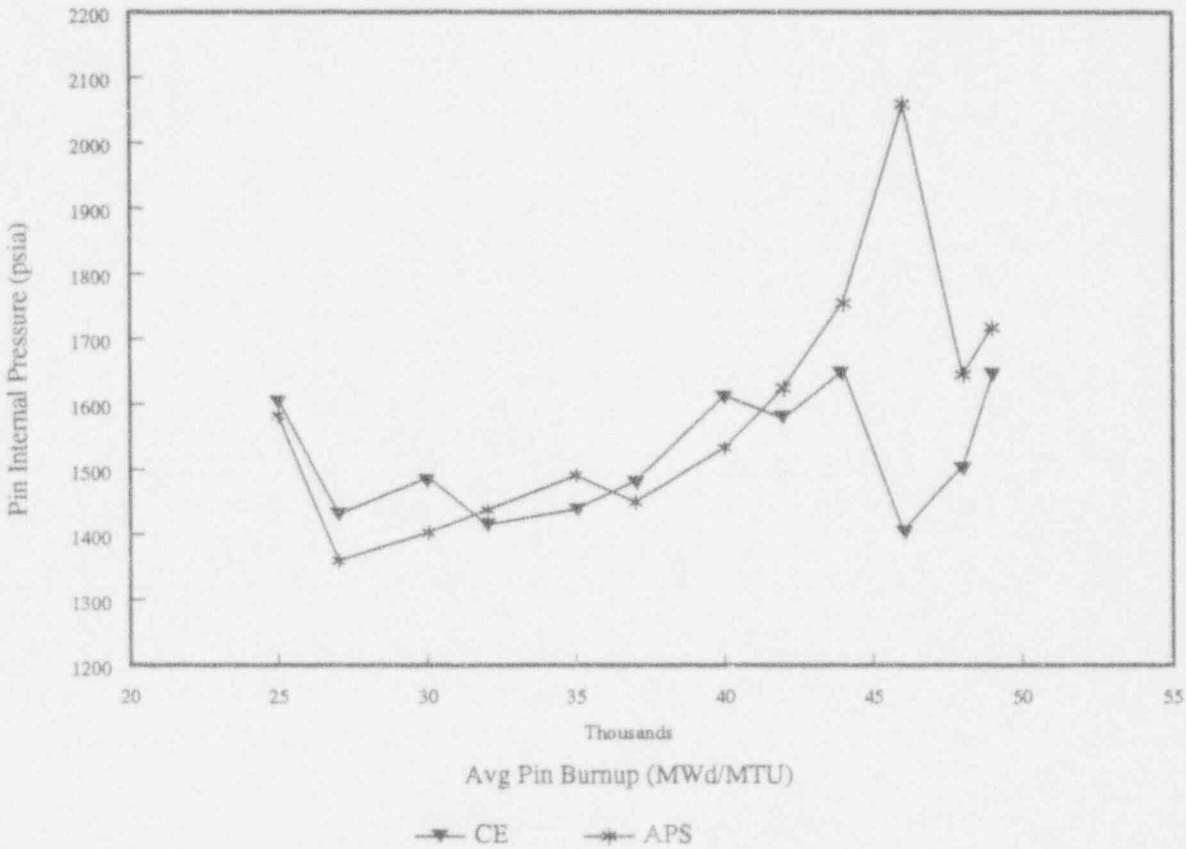


Figure 4.3-4 Fuel Pin Internal Pressure versus Average Pin Burnup



4.4 COMPARISON OF SAFETY DESIGN PRINCIPAL RESULTS

The following section compares the principal results of the APS safety design for the PVNGS Unit 3 Cycle 3 reload analyses with those of CE. For the purpose of comparison, this section is divided into two categories: fuel failure assessment and margin assessment.

4.4.1 Fuel Failure Events

The analysis methods presented in section 3.4.2 were used to determine the fuel failure for the limiting events for this cycle. The fuel failure for these events is presented in Table 4.4-1.

Table 4.4-1 Comparison of Fuel Failure Results for Unit 3 Cycle 3

	APS Design		CE Design	
Event Type	%Fuel Failure	COLSS ROPM	%Fuel Failure	COLSS ROPM
IOSGADV + LOAC	1.73		1.43	
SLB	0.07		0.14	
Single RCP SR	2.7		3.94	
CEA Ejection	9.04		8.9	

For both the APS and CE designs the fuel failure was within the reported numbers for the previous cycles. The differences in fuel failure between the APS and CE designs were basically due to the initial ROPMs assumed for the four events and due to different pin censi.

4.4.2 Margin Setting Events

The objective of the margin assessment analyses was to determine the minimum steady state ROPM which must be maintained by the COLSS. The initial margin requirements were determined based on fuel failure events (discussed in section 4.4.1) and margin setting events. The safety analysis design for margin setting events is discussed in section 3.4.2.2. The required initial margin results obtained by these analyses were used to generate COLSS constants and the DNBR limit lines for COLSS out-of-service conditions.

4.4.3 Technical Specification Setting Event

The Inadvertent Boron Dilution Event for APS and CE analysis followed slightly different approaches to obtain the loss of shutdown margin for the All Rods In (ARI) and N-1 (with the most reactive rod stuck out) CEA configuration. Both designs used an increase in the lower breakpoint temperature dependent shutdown margin from 3.5% $\Delta\rho$ to 4% $\Delta\rho$ at 350°F for modes 1, 2, 3, 4 and 5. Thus, this necessitated a change in the Figure 3.1-1A of Technical Specification 3.1.1.2. In the APS calculation, a minimum time to criticality for 'Mode 5 (drained)' of 58.8 minutes was obtained. The CE analysis, using physics input for the N-1 CEA configuration, concluded that Unit 3 Cycle 2 results were bounding by determining the time to lose 1% $\Delta\rho$ subcriticality. The minimum time for loss of shutdown margin, again for 'Mode 5 (drained)' operation for the CE Unit 3 Cycle 2 calculation was 55.5 minutes. Therefore, there was good agreement for this case.

Technical Specification Tables 3.1-2 through 3.1-5 provide the required boron monitoring frequencies in the event one or both startup channel flux alarms are inoperable. CE analyses required changes to Tables 3.1-2, 3.1-3 and 3.1-5 since the calculated monitoring times were less than those required for the Unit 3 Cycle 2 Technical Specifications. The APS analyses justified the existing Unit 3 Cycle 2 Technical Specifications for these tables.

4.4.4 Degraded Performance of CPCS & COLSS Category

4.4.4.1 COLSS in-service and at least one CEAC operable

Sections 3.4.2.1 and 3.4.2.2 analyses were used to calculate the ROPM for the COLSS in-service and CEACs operable condition. As discussed in sections 3.4.2.1 and 3.4.2.2, the required initial margin was preserved by COLSS through the use of both ROPM and UFF. UFF was used to protect the Loss of Flow (LOF) event only and all other events were protected by using ROPM. The UFF results from the LOF analyses for APS and CE designs are given in Table 4.4-2 and the agreement was excellent.

Table 4.4-2 Comparison of UFF for LOF Analyses

Based on the ROPM requirements, the final conservative ROPMs selected for the APS and CE designs are presented in Table 4.4-3. A comparison of these values demonstrates that there is very good agreement between the two results.

Table 4.4-3 Comparison of ROPM, COLSS In-Service and at least one CEACs Operable

4.4.4.2 COLSS in-service and both CEACs inoperable

Table 4.4-4 Comparison of ROPM with COLSS In-Service and both CEACs Inoperable

4.4.4.3 COLSS out-of-service and at least one CEAC operable

a. The COLSS in-service UFF versus ASI results given in section 4.4.4.1 also applied for COLSS out-of-service.

b. The ROPM as a function of power is presented in Table 4.4-5.

Table 4.4-5 Comparison of ROPM with COLSS Out-of-Service and at Least One CEAC Operable

The APS and CE ROPMs were minimum margins required for this category from the analyses. The two sets of numbers compare favorably.

4.4.4.4 COLSS out-of-service and both CEACs inoperable

a. The COLSS in-service and CEACs operable UFF versus ASI results given in section 4.4.4.1 remained valid for COLSS out-of-service and CEACs inoperable conditions.

b. For convenience in calculating the COLSS out-of-service and CEACs inoperable DNBR limit line, a maximum radial distortion factor was given for the ROPMs at each power level as shown in Table 4.4-6.

Table 4.4-6 Comparison of ROPM with COLSS Out-of-Service and Both CEACs Inoperable

As can be seen from this table, the APS and CE numbers agree at the 95% power level (CE presented the ROPM at 100% power and this ROPM has been used at the 95% power level). At the 65% power level, CE determined that the 100% ROPM was sufficient to protect for COLSS out-of-service and CEACs inoperable conditions at the 65% power level.

Thus, the results showed good agreement between the two sets of numbers.

4.4.5 Transient Related CPCS Constants

The transient offsets for thermal power and neutron flux power for APS and CE analyses are presented in Table 4.4-7.

Table 4.4-7 Comparison of Transient Offsets for APS and CE Analyses

The thermal transient offset determined by APS and CE was the same. The neutron flux power offset for APS analysis included 1% power penalty to limit the fuel failure consequences of IOSGADV+LOAC event, and an additional 1% penalty to accommodate the lag in CPC response to a decrease in RCS pressurizer pressure during an excess load event. The CE analysis did not require the additional 1% power penalty since the fuel failure was not more limiting than previously calculated.

4.5 COMPARISON OF CPCS DESIGN PRINCIPAL RESULTS

This section presents a comparison of the principal results of the APS CPCS design with those of CE. The purpose of the CPCS reload design is to verify or modify the CPCS constants required by the algorithms as necessary to ensure that the CPCS design basis is met. As noted in section 3.5, of the three types of CPCS constants (database, Reload Data Block and addressable) only the RDB and addressable constants are considered for change as part of a reload design. The comparisons presented in this section are divided into three major areas (a) Reload Data Block (RDB) constants comparison, (b) addressable constants comparison and (c) comparison of related Technical Specification requirements.

Table 4.5-1 compares the APS and CE calculated RDB constants for Unit 3 Cycle 3 which were different than the Unit 3 Cycle 2 values.

Table 4.5-1 Cycle Specific Reload Data Block Constants Comparison

Table 4.5-1 shows that the APS and CE RDB constants were identical except for the constants [ ] The difference in the values for constants [ ] was attributed to differences in the previous cycle shutdown windows and the amount of conservative round off included in the values. A comparison of the unadjusted values (before the application of uncertainty and round off) is shown on Table 4.5-2.

Table 4.5-2 Comparison of CPCS Constants [ ]

The differences in the radial peaking factors for the CEA group 5+4 [FPR(7)] and 5+4+PLR [FPR(8)] configuration were very small and are also due to a combination of previous cycle window and round off error.

Table 4.5-3 shows the addressable constants determined for the APS and CE designs.

Table 4.5-3 Comparison of CPCS Addressable Constants (Type II)

Table 4.5-3 Comparison of CPCS Addressable Constants (Type II)

Table 4.5-3 shows that the APS and CE addressable constants were indentical except for constants [ ] The difference in the values of all constants except for [ ] was small and could be attributed to the difference in cycle length and round off.

The CPC addressable constant [

]

As described in section 3.4.2.4, the Technical Specifications require that the CPCS be used to monitor the DNBR LCO when the COLSS is out-of-service. Part of the CPCS reload design was to determine the associated limit lines based on CPC DNBR to be included in the Technical Specifications. Figures 4.5-1 and 4.5-2 show the final COLSS out-of-service CPC DNBR limit lines for the conditions of (a) at least one CEAC operable and (b) both CEACs inoperable. In both the APS and CE related analyses it was shown that these limits were appropriate for Unit 3 Cycle 3.

Figure 4.5-1 DNBR Margin Operating Limit Based On Core Protection Calculators (COLSS Out-of-Service, CEACs Operable)

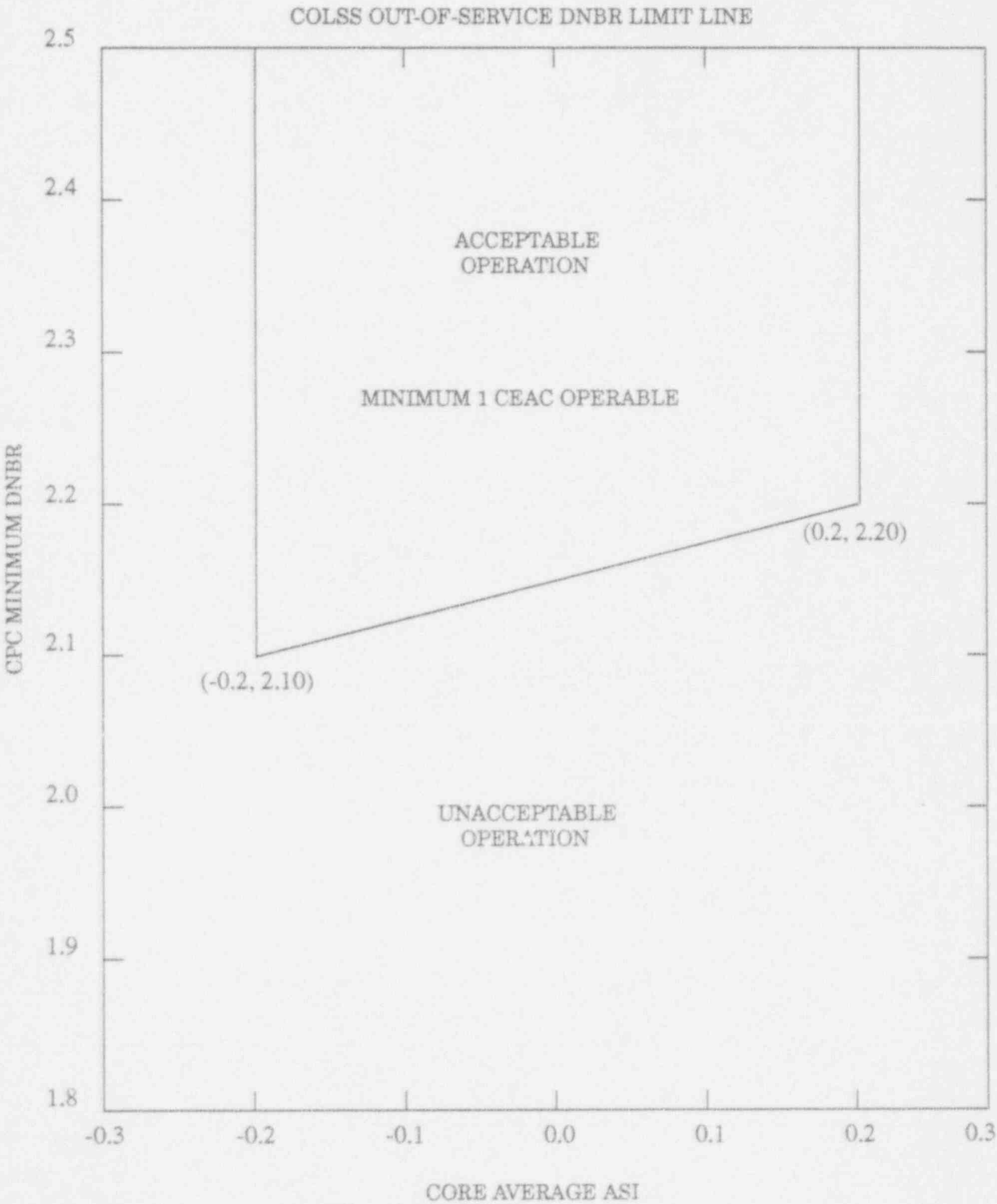
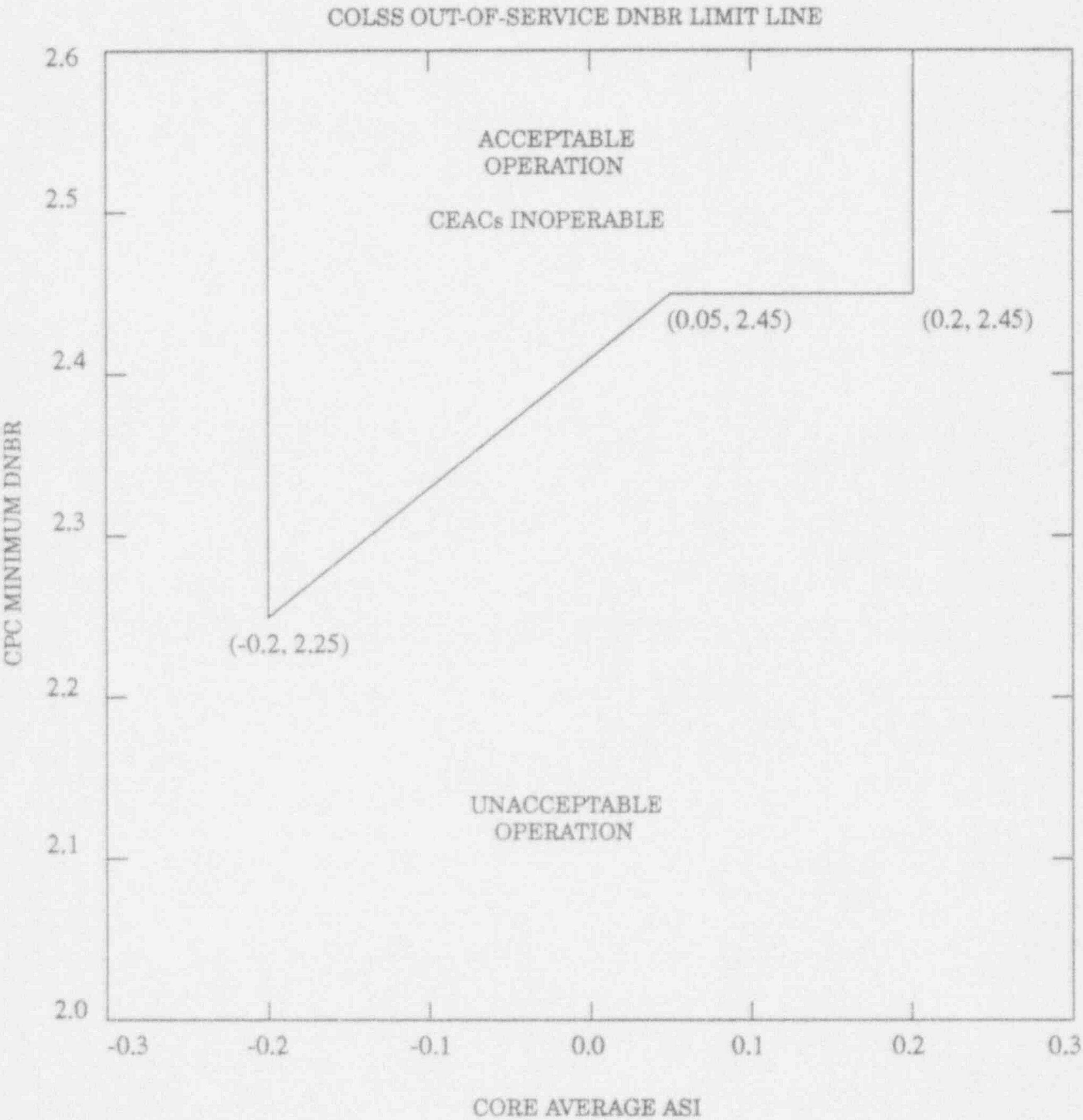


Figure 4.5-2 DNBR Margin Operating Limit Based On Core Protection Calculators (COLSS Out-of-Service Both CEACS Inoperable)



4.6 COMPARISON OF COLSS DESIGN PRINCIPAL RESULTS

This section presents a comparison of the principal results of the APS COLSS design with those of CE. The purpose of the COLSS reload design is to determine the COLSS constants necessary to ensure that the COLSS conservatively monitors the required Technical Specification LCOs. As noted in section 3.6, two types of COLSS constants (database and addressable) are considered for change as part of a reload design. The comparisons presented in this section are divided into two parts (a) data base constants comparison and (b) addressable constants comparison.

Table 4.6-1 provides a comparison of several key cycle specific data base constants typically evaluated for change during a reload design.

[

Table 4.6-1 Comparison of COLSS Database Constants

]

This table shows identical results for all constants except for the constants [

] The differences in the radial peaking factors were relatively small and could be attributed to round off error.

Table 4.6-2 shows the addressable constants determined for the APS and CE designs.

Table 4.6-2 Comparison of COLSS Addressable Constants

There were small differences between the values of [ ]
All these values were calculated as a function of core burnup. Typically the maximum value is conservatively rounded up to produce a final value. The maximum raw value of each of these values is given on Table 4.6-3.

Table 4.6-3 Comparison of COLSS Constant Raw Values

This table shows that the differences between the APS and CE values were small. All other addressable constant values shown on Table 4.6-2 were exactly the same.

5.0 CONCLUSION

The engineering staff of Arizona Public Service was trained by CE in the use of reload analysis methods through a structured reload technology transfer program. The program lasted approximately five years and involved extensive commitment of resources and engineer-hours by APS. In the final phase of the program, APS and CE performed a parallel and independent reload analysis to support the PVNGS Unit 3 Cycle 3 reload core design. The comparison of the principal safety and setpoint analysis results as calculated by APS and CE show excellent agreement. This comparison demonstrates the ability of APS to perform the analyses required for the licensing, operation and surveillance of a PVNGS reload fuel cycle.

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List of Definitions

ADV	Atmospheric Dump Valve
AOO	Anticipated Operating Occurrences
AOPM	Available Over Power Margin
APS	Arizona Public Service Co.
ARI	All Rods (CEAs) In
ARO	All Rods (CEAs) Out
ASGT	Asymmetric Steam Generator Transient
ASI	Axial Shape Index
BDAS	Boron Dilution Alarm System
BOC	Beginning of Cycle
CBC	Critical Boron Concentration
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEAC	Control Element Assembly Calculator
CEDM	Control Element Drive Mechanism
CEDMCS	Control Element Drive Mechanism Control System
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
COLSS	Core Operating Limit Supervisory System
CPCS	Core Protection Calculator System
CPCs	Core Protection Calculators
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EOC	End of Cycle
ESFAS	Emergency Safety Features Actuation System
F_r	Integrated Radial Peaking Factor

List of Definitions (Con't)

Fq	Maximum Value for the Core of the Ratio of Peak Pin Power in an Assembly to the Average Pin Power in the Core
FTC	Fuel Temperature Coefficient
Fxy	Planar Radial Peaking Factor
HFP	Hot Full Power
HPSI	High Pressure Safety Injection
IBW	Inverse Boron Worth
IOSGADV	Inadvertent Opening of Steam Generator Atmospheric Dump Valve
l*	Prompt Neutron Lifetime
LCO	Limiting Conditions for Operation
LHR	Linear Heat Rate
LOAC	Loss of Alternating Current Power
LOCA	Loss Of Coolant Accident
LOCV	Loss Of Condenser Vacuum
LOF	Loss Of coolant Flow
LPD	Local Power Density
LSSS	Limiting Safety Systems Setting
LTOP	Low Temperature Overpressure Protection
MC	Mesh Centered
MDNBR	Minimum allowable value for the Departure from Nucleate Boiling Ratio
MSCU	Modified Statistical Combination of Uncertainties
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OJT	On the Job Training

List of Definitions (Con't)

PDF	Probability Distribution Function
PDIL	Power Dependent Insertion Limit
PLR	Part Length Rod or CEA
PMS	Plant Monitoring System
POL	Power Operating Limit
PPM	Parts Per Million
PPS	Plant Protection System
PSV	Pressurizer Safety Valve
PVNGS	Palo Verde Nuclear Generating Station
RAR	Reload Analysis Report
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDB	Reload Data Block
ROCS	Reactor Operation and Control Simulator
ROPM	Required Over Power Margin
RPD	Relative Power Density
RPS	Reactor Protection System
SAFDL	Specified Acceptable Fuel Design Limits
SBCS	Steam Bypass Control System
SCS	Shutdown Cooling System
SCU	Statistical Combination of Uncertainties
SLB	Steam Line Break
SRP	Safety Review Plan
SR	Seized Rotor
SS	Sheared Shaft
UFF	Under Flow Fraction
UFSAR	Updated Final Safety Analysis Report
VOPT	Variable Over Power Trip
WRSO	Worst Rod (CEA) Stuck Out