THOMAS C. GEER
Vice President
Nuclear Engineering

Duke Energy Corporation
526 South Church St.
Charlotte, NC 28202

Mailing Address:
EC08H / PO Box 1006
Charlotte, NC 28201-1006

704 382 4712
704 382 7852 fax
tcgeer@duke-energy.com

July 30, 2007

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTENTION: Document Control Desk

Subject: Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke)
McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369, 50-370
Catawba Nuclear Station, Units 1 and 2, Docket Nos. 50-413, 50-414
Transmittal of DPC-NF-2010-A, Nuclear Physics Methodology for Reload Design, Revision 2

By letter dated June 24, 2003, the Nuclear Regulatory Commission (NRC) reviewed a revision to DPC-NF-2010, "Nuclear Physics Methodology for Reload Design," Revision 2. In that letter, the NRC determined that the change in the proposed report did not impact or change any of the conclusions made in the agency's most recent Safety Evaluation dated October 1, 2002.

Attached is a copy of the above cited report which includes the June 24, 2003 approval letter.

If there are any questions concerning this matter, please contact L. B. Jones at (704) 382-4753.

Sincerely,

Thomas C. Geer

Attachment

A001

NRR

xc: (without attachment)

W. D. Travers, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303-8931

J. F. Stang, Senior Project Manager (CNS & MNS)
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Mail Stop 0-8 G9A
Rockville, MD 20852-2738

J. B. Brady
NRC Senior Resident Inspector
McGuire Nuclear Station

A. T. Sabisch
NRC Senior Resident Inspector
Catawba Nuclear Station

DUKE POWER COMPANY
MCGUIRE NUCLEAR STATION
CATAWBA NUCLEAR STATION

Nuclear Physics Methodology For Reload Design

DPC-NF-2010-A
Revision 2

NRC Approval Letter Dated June 24, 2003

Nuclear Generation Department
Nuclear Engineering

Revision History

Revision	Description
DPC-NF-2010, Original Issue	Originally submitted to the NRC for approval in July 1984. Additional information was submitted to the NRC supplying responses to a request for additional information.
DPC-NF-2010-A, Original Issue	NRC approved version issued in May 1985.
DPC-NF-2010, Revision 1	Submitted to the NRC for approval in August 2001. This revision updates the report for completeness to indicate the use of NRC approved methods approved subsequent to the implementation of the original issue including the use of CASMO-3/SIMLUATE-3 reactor physics methods. This revision also removes unnecessary data including items not reviewed by the NRC in the original version and general fuel data not necessary for these methods. Also various editorial changes are made, including updating the Table of Contents. Changes associated with this revision are denoted by revision bars, except format changes.
DPC-NF-2010-A, Revision 1	NRC approved. SER issued October 1, 2002.
DPC-NF-2010, Revision 2	Submitted to the NRC for Approval in November 2002. This revision removed the description of the source of fuel temperature data for the reactor physics code. Changes associated with this revision are denoted by revision bars, and previous revision bars are removed.
DPC-NF-2010-A Revision 2	NRC approved. Approval letter issued June 24, 2003.

STATEMENT OF DISCLAIMER

This report was prepared by Duke Power Company ("Duke Power") for filing with the United States Nuclear Regulatory Commission ("USNRC") for the sole purpose of obtaining approval of Duke Power's PWR nuclear design methods at McGuire and Catawba. Duke Power makes no warranty or representation and assumes no obligation, responsibility, or liability with respect to the contents of this report or its accuracy or completeness. Any use of or reliance on the report or the information contained in this report is at the sole risk of the party using or relying on it.

ABSTRACT

This Technical Report describes Duke Power Company's Nuclear Design Methodology for the McGuire and Catawba Nuclear Station. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology Duke Power intends to use to perform design calculations and to provide operational support, and the development of statistical reliability factors.

TABLE OF CONTENTS

		Page
1.	INTRODUCTION	1-1
	1.1 General Nuclear Design Description	1-1
	1.2 Definition of Terms	1-3
2.	FUEL DESCRIPTION	2-1
3.	NUCLEAR CODE SYSTEM	3-1
	3.1 Introduction	3-1
	3.2 Sources of Input Data	3-1
	3.3 Cross Section Preparation	3-2
	3.3.1 Fuel Calculations	3-3
	3.3.2 Non-Fuel Calculations	3-3
	3.4 PDQ07 Models	3-4
	3.4.1 Colorset PDQ07 Modeling	3-4
	3.4.2 Quarter Core PDQ07 Model	3-5
	3.5 EPRI-NODE-P Model	3-7
	3.6 CASMO-3/SIMULATE-3 Model	3-8
4.	FUEL CYCLE DESIGN	4-1
	4.1 Preliminary Fuel Cycle Design - Initialization	4-1
	4.1.1 Review of Design Basis Information	4-1
	4.1.2 Determination of Cycle - Specific Operating Requirements	4-1
	4.1.3 Preliminary Loading Pattern and Reload Region Determination	4-2
	4.2 Final Fuel Cycle Design	4-2
	4.2.1 Fuel Shuffle Optimization and Cycle Depletion	4-3
	4.2.2 Rod Worth Calculations	4-3
	4.2.2.1 Control Rod Worths	4-4
	4.2.2.2 Shutdown Margin	4-4
	4.2.2.3 Rod Insertion Limit Verification	4-5
	4.2.2.4 Ejected Rod Analysis	4-5
	4.2.2.5 Dropped Rod Analysis	4-7
	4.2.3 Fuel Burnup Calculations	4-7

TABLE OF CONTENTS (CONT.)

		Page
4.2.4	Reactivity Coefficients and Deficits	4-7
4.2.4.1	Doppler Coefficient	4-8
4.2.4.2	Moderator Temperature Coefficient	4-8
4.2.4.3	Isothermal Temperature Coefficient	4-9
4.2.4.4	Power Coefficient and Power Defect	4-9
4.2.4.5	Miscellaneous Coefficients	4-11
4.2.4.6	Boron Related Parameters	4-11
4.2.4.7	Xenon Worth	4-11
4.2.4.8	Kinetics Parameters	4-12
4.2.5	Assessment of the Fuel Cycle Design	4-12
5.	NODAL ANALYSIS METHODOLOGY	5-1
5.1	Purpose and Introduction	5-1
5.2	Fuel Cycle Depletion	5-2
5.3	Rod Worth Analysis	5-2
5.3.1	Differential Rod Worth Analysis	5-3
5.3.2	Integral Rod Worth Analysis	5-3
5.4	Shutdown Margin Analysis	5-4
5.4.1	Shutdown Margin	5-4
5.4.2	Shutdown Boron Concentration	5-5
5.5	Rod Insertion Limit Assessment	5-5
5.6	Trip Reactivity Analysis	5-7
5.6.1	Minimum Trip Reactivity	5-7
5.6.2	Trip Reactivity Shape	5-8
5.7	Assessment of Nodal Analyses	5-8
6.	CALCULATION OF SAFETY RELATED PHYSICS PARAMETERS	6-1
6.1	Safety Analysis Physics Parameters	6-1
6.2	Core Power Distributions	6-1
6.3	Power Peaking and Reliability Factors	6-2
7.	3-D POWER PEAKING ANALYSIS	7-1
8.	RADIAL LOCAL ANALYSIS	8-1
8.1	Background	8-1

TABLE OF CONTENTS (CONT.)

		Page
8.2	Comparison of PDQ07 to CASMO-2 at Hot Full Power Condition	8-1
8.3	Comparisons of PDQ07 to Cold Criticals	8-2
8.4	BNL Benchmark Assembly Problem	8-2
8.5	Conclusion	8-3
9.	DEVELOPMENT OF CORE PHYSICS PARAMETERS	9-1
9.1	Startup Test Predictions	9-1
9.1.1	Critical Boron Concentrations and Boron Worths	9-1
9.1.2	Xenon Worth and Defect	9-2
9.1.3	Rod Worths	9-3
9.1.3.1	Group Worths	9-3
9.1.3.2	Stuck Rod Worth	9-3
9.1.3.3	Dropped Rod Worth	9-3
9.1.3.4	Ejected Rod Worth	9-4
9.1.4	Reactivity Coefficients	9-4
9.1.4.1	HZP Coefficients	9-4
9.1.4.2	HFP Coefficients	9-5
9.1.5	Power Distribution	9-5
9.1.6	Kinetics Parameters	9-5
9.2	Startup and Operation Report	9-6
10.0	PHYSICS TEST COMPARISONS	10-1
10.1	Introduction	10-1
10.2	Critical Boron Concentrations	10-2
10.2.1	Measurement Technique	10-2
10.2.2	Calculational Technique	10-2
10.2.3	Comparison of Calculated and Measured Results	10-2
10.2.3.1	Hot Zero Power Comparison	10-2
10.2.3.2	Hot Full Power Comparison	10-3
10.2.4	Summary	10-3
10.3	Control Rod Worth	10-3
10.3.1	Measurement Techniques	10-3
10.3.2	Calculational Techniques	10-3
10.3.3	Comparisons of Calculated and Measured Results	10-4
10.3.4	Summary	10-4

TABLE OF CONTENTS (CONT.)

		Page
10.4	Ejected Rod Worths	10-5
	10.4.1 Measurement Technique	10-5
	10.4.2 Calculational Technique	10-5
	10.4.3 Comparison of Calculated and Measured Results	10-5
10.5	Isothermal Temperature Coefficient	10-5
	10.5.1 Measurement Technique	10-6
	10.5.2 Calculational Technique	10-6
	10.5.3 Comparison of Calculated and Measured Results	10-6
	10.5.4 Summary	10-6
11.	POWER DISTRIBUTION COMPARISONS	11-1
11.1	Introduction and Summary	11-1
	11.1.1 Introduction	11-1
	11.1.2 Summary	11-1
11.2	Measured Data	11-2
	11.2.1 Measured Assembly Power Data	11-2
	11.2.2 Measurement System Description	11-2
11.3	EPRI-NODE-P Power Distribution	11-3
	11.3.1 EPRI-NODE-P Model	11-3
	11.3.2 Fuel Cycle Simulation	11-3
	11.3.3 Radial Power Methodology	11-5
	11.3.4 Assembly Peak Axial Power Methodology	11-5
	11.3.5 Conclusions	11-6
11.4	PDQ07 - Power Distribution Comparisons	11-6
11.5	Statistical Analysis	11-6
	11.5.1 Observed Nuclear Reliability Factor Derivation	11-6
	11.5.2 Normality Test Results	11-9
	11.5.3 Observed Nuclear Reliability Factors for EPRI-NODE-P	11-10
	11.5.4 Quantitative Comparison of EPRI-NODE-P to Measurement	11-11
	11.5.5 Relative Percent Differences	11-12
	11.5.6 Conclusions	11-13
12.	REFERENCES	12-1

TABLE OF CONTENTS (CONT.)

APPENDIX A: Code Summary

APPENDIX B: NRC/DPC Correspondence Regarding NRC Request for Additional Information

APPENDIX C: Original Issue NRC SER

APPENDIX D: Revision 1 NRC SER

APPENDIX E: Revision 2 NRC Approval Letter

LIST OF TABLES

Table		Page
3-1	EPRI-CELL/CASMO-2 Cross Sections Calculation by Composition	3-9
4-1	Nuclear Design Data for Reload Design	4-13
4-2	Shutdown Margin Calculation	4-14
4-3	Typical Boron Parameters	4-15
5-1	BOC Trip Reactivity Calculation	5-9
8-1	Characteristics of 1/8th Assembly Simulations	8-4
8-2	Peak Pin Power Comparison	8-5
8-3	Statistical Summary of Percent Differences between PDQ07 and CASMO-2 for Pins in Assemblies with Powers Greater Than or Equal to 1.000	8-6
9-1	Critical Boron Concentrations (ppmB)	9-7
9-2	Boron Worth (pcm/ppmB)	9-8
9-3	Typical Core Physics Data	9-9
10-1	McGuire Critical Boron Concentrations at Hot Zero Power, BOC	10-7
10-2	McGuire 1 Cycle 1-1A Hot Full Power Critical Boron Concentration	10-8
10-3	McGuire Control Rod Worths at Hot Zero Power, BOC	10-9
10-4	McGuire Control Rod Worths at Hot Zero Power, BOC Using Boron EndPoints	10-10
10-5	McGuire PDQ07 Calculated Control Rod Worths vs. Measured Rod Worths at HZP, BOC	10-11
10-6	McGuire 1 Cycle 1 Ejected Rod Worths	10-12
10-7	McGuire Isothermal Temperature Coefficients at Hot Zero Power, BOC	10-13
11-1	McGuire Unit 1 Cycle 1 State Points	11-15
11-2	McGuire Unit 1 Cycle 1A State Points	11-16
11-3	Sequoyah Unit 1 Cycle 1 State Points	11-17

LIST OF TABLES (CONT.)

Table		Page
11-4	McGuire Unit 1 Cycles 1 and 1A and Sequoyah Unit 1 Cycle 1 State Points for PDQ07 Calculated and Measured Data	11-18
11-5	Difference Distribution Normality Tests for (C, M $\geq$ 1.0) - 5% Level of Significance	11-19
11-6	Calculated ORNFs and Associated Data	11-20
11-7	Difference, Means, and Standard Deviations for Assembly Radial Powers (C, M $\geq$ 1.0)	11-21
11-8	Difference, Means, and Standard Deviations for Assembly Peak Axial Powers (C, M $\geq$ 1.0)	11-22
11-9	Percent Difference Means (C,M $\geq$ 1.0) - Assembly Radial Powers	11-23
11-10	Percent Difference Means (C,M $\geq$ 1.0) - Assembly Peak Axial Powers	11-24

LIST OF FIGURES

Figure		Page
3-1	Nuclear Flow Chart For EPRI-ARMP	3-10
3-2	17x17 Assembly PDQ07 Colorset Geometry	3-11
3-3	PDQ07 Quarter Core Model Assembly Geometry	3-12
3-4	PDQ07 Quarter Core 17x17 Geometry	3-13
8-1	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 1	8-7
8-2	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 2	8-8
8-3	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 3	8-9
8-4	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 4	8-10
8-5	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 5	8-11
8-6	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 6	8-12
8-7	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 7	8-13
8-8	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 8	8-14
8-9	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 9	8-15
8-10	CASMO-2 and PDQ-7 Rod Power Comparison BOL HFP No Xenon Case Number 10	8-16
9-1	Boron Letdown Curve HFP, ARO	9-10
9-2	Differential Boron Worth HZP, BOL	9-11
9-3	Inverse Boron Worth HFP, ARO	9-12
9-4	Equilibrium Xenon Worth HFP	9-13
9-5	Xenon Reactivity Defect BOL	9-14
9-6	Integral Rod Worth HZP, BOL	9-15
10-1	McGuire 1 Cycle 1 Boron Letdown Curves	10-14
10-2	McGuire 1 Cycle 1A Boron Letdown Curves	10-15

LIST OF FIGURES (CONT.)

Figure		Page
11-1	Instrumented Fuel Assemblies McGuire and Sequoyah	11-25
11-2	Control and Shutdown Bank Locations McGuire Unit 1 Cycle 1	11-26
11-3	Core Loading Pattern McGuire Unit 1 Cycle 1	11-27
11-4	Control and Shutdown Bank Locations Sequoyah Unit 1 Cycle 1	11-28
11-5	Core Loading Pattern Sequoyah Unit 1 Cycle 1	11-29
11-6 to 11-30	McGuire 1 Cycle 1 Assembly Radial Powers Calculated vs. Measured	11-30 to 11-54
11-31 to 11-55	McGuire 1 Cycle 1 Assembly Peak Axial Powers Calculated vs. Measured	11-55 to 11-79
11-56 to 11-60	McGuire 1 Cycle 1A Assembly Radial Powers Calculated vs. Measured	11-80 to 11-84
11-61 to 11-65	McGuire 1 Cycle 1A Assembly Peak Axial Powers Calculated vs. Measured	11-85 to 11-89
11-66 to 11-72	Sequoyah 1 Cycle 1 Assembly Radial Powers Calculated vs. Measured	11-90 to 11-96
11-73 to 11-79	Sequoyah 1 Cycle 1 Assembly Peak Axial Powers Calculated vs. Measured	11-97 to 11-103
11-80 to 11-83	McGuire 1 Cycle 1 PDQ07 Assembly Radial Powers Calculated vs. Measured	11-104 to 11-107
11-84 to 11-86	Sequoyah 1 Cycle 1 PDQ07 Assembly Radial Powers Calculated vs. Measured	11-108 to 11-110

1. INTRODUCTION

1.1 General Nuclear Design Description

A commercial Pressurized Water Reactor (PWR) is designed to hold a constant number of nuclear fuel assemblies which are generally identical mechanically, but differ in the amount of fissile material content. During cycles subsequent to the initial cycle, fuel assemblies differ in burnup as well. Refueling occurs at intervals appropriate for the power production needed, for example 12, 18, or 24 months. At refueling, a predetermined number of irradiated fuel assemblies are discharged and the same number are loaded as fresh (reload region) or possibly irradiated assemblies. The fuel management scheme determines the locations of all fresh and irradiated assemblies.

This report describes some of the various aspects of nuclear design with principal emphasis placed upon development of a core loading pattern and nuclear calculations performed to evaluate safety and operational parameters. The following sections provide detailed discussion, including descriptions, of design methods, analytical formulations, and calculational procedures involved in the various nuclear design tasks for the McGuire and Catawba Nuclear Stations. The nuclear design is essentially a series of analytical calculations with the objective of designing the reload core in such a manner that the reactor can be operated up to a specified power level for a specified number of days within acceptable safety and operating limits. It consists of the development of the basic specifications of the reload region (fuel enrichment, number of assemblies, uranium loading, etc.); it sets forth the number and identity of each residual fuel assembly, selects the location of each fuel assembly in the core for the new fuel cycle, and establishes the core characteristics. The nuclear design used in conjunction with the thermal hydraulic and safety analyses establishes the operating limits, control rod limits, and protection system setpoints.

In arriving at the final nuclear design, the designer tries to meet the requirements imposed by the operational considerations, fuel economics

considerations, and safety considerations. These requirements are called nuclear design criteria and are as follows:

1. Initial core excess reactivity will be sufficient to enable full power operation for the desired length of the cycle, with appropriate allowance for any planned coastdown.

2. The fuel assemblies to be discharged at the end of the fuel cycle will attain maximum permissible burnup so that maximum energy extraction consistent with the fuel mechanical integrity criteria is achieved.

3. Values of important core parameters (moderator temperature coefficient, Doppler coefficient, ejected rod worth, boron worth, control rod worth, maximum linear heat rate of the fuel pin at various elevations in the core, and shutdown margin) predicted for the cycle are conservative with respect to the values assumed in the safety analysis of various postulated accidents. If they are not conservative, acceptable reevaluation or reanalysis of applicable accidents is performed, or the core is redesigned.

4. The power distributions within the reactor core for all possible (or permissible) core conditions that could exist during the operation of the cycle will not lead to exceeding the thermal design criteria of the fuel or exceeding the LOCA-limited peaking factors.

5. Fuel management will produce fuel rod power and burnup consistent with the mechanical integrity analysis of the fuel rod.

The nuclear design process described in this report consists of mechanical properties used as nuclear design input, the nuclear code system and methodology Duke Power intends to use to perform design calculations and to provide operational support, and the development of statistical reliability factors.

The nuclear design calculations described in this report are covered by the Duke Power Quality Assurance program (Reference 21).

1.2 Definition of Terms

Presented below are terms which will be needed throughout the text of the report:

a/o	atom percent
ARI	all rods in
ARO	all rods out
axial offset (AO)	$\frac{P_T - P_B}{P_T + P_B}$, where P_T is the integrated power in the top half of the core, and P_B is the integrated power in the bottom half of the core
β_i	delayed neutron fraction for group i
β_{eff}	effective delayed neutron fraction in core
BOL	beginning of life
BP	burnable poison
BU	fuel burnup
C_B	Chemical shim boron concentration in the main coolant
CZP	cold zero power
EOL	end of cycle life
EQXE	equilibrium xenon condition
GWD/MTU	Gigawatt days per metric ton of initial uranium metal, 1 GWD/MTU is 1000 MWD/MTU
HFP	hot full power
HZP	hot zero power

$\bar{I}$	delayed neutron importance factor
ΔI or Axial Flux Difference (AFD)	flux difference between the top and bottom halves of the core; in this report, ΔI is a calculated value, rather than a difference between measured signals from the excore detectors
K(z)	F_Q^T normalized to the maximum value allowed at any core height
ℓ^*	prompt neutron lifetime
MOL	middle of cycle life
MWD/MTU	measure of energy extracted per unit weight of initial uranium metal fuel; is equal to 1 megawatt times 1 day, divided by 1 metric ton of uranium
pcm	percent mille (a reactivity change that equals 10^{-5} $\Delta\rho$)
ppm	parts per million by weight; which specifies the amount of chemical shim boron present by weight in the main coolant system
radial local	ratio of assembly maximum rod to assembly average x-y power
RCCA	rod cluster control assembly; the type of control rod assembly used in McGuire and Catawba. (All RCCA are full length absorbers for both plants.)
ρ	reactivity
$\Delta\rho$	$\frac{K_1 - K_2}{K_1 \times K_2}$, where K_1 and K_2 are eigenvalues obtained from two calculations where only one parameter was varied
shutdown margin	amount of negative reactivity (ρ) by which a reactor core is maintained in a HZP subcritical condition after a control rod trip
step	unit of control rod travel equal to 0.625 inch
TMOD	moderator temperature; defined as the temperature corresponding to the average water enthalpy of the core
T_{res}	resonance temperature of the fuel
w/o	weight percent

Power distributions will be quantified in terms of hot channel factors. These factors are a measure of the peak pellet power and the energy produced in the coolant. The factors are:

Power density	thermal power produced per unit volume of the core (KW/liter)
Linear Power Density	thermal power produced per unit length of active fuel (KW/ft)
Average Linear Power Density	total thermal power produced in the core divided by the total active fuel length of all fuel rods in the core
Local Heat Flux	local heat flux on the cladding surface (BTU/ft^2/hr)
Rod Power or Integral Power	is the length integrated linear power density in one rod (KW)

Various hot channel factors are:

F_Q^T, Heat Flux Hot Channel Factor, the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, including conservatisms for fuel pellet and rod dimensional uncertainties.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average linear power density, assuming nominal fuel rod and pellet dimensions.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter. Combined statistically the net effect is typically a factor of 1.03 to be applied to calculated KW/ft.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

2. FUEL DESCRIPTION

The reactor cores for McGuire and Catawba contain 193 fuel assemblies. Each fuel assembly consists of 264 fuel rods, 24 guide thimble tubes and 1 instrumentation thimble tube assembled in a square 17x17 lattice. The assembly structure consists of top and bottom nozzles and grid assemblies positioned axially along the fuel assembly. Each fuel rod contains a column of stacked fuel pellets.

Detailed design data for the fuel pellets, fuel rods, fuel assembly, and reactivity control components can be found in the Updated Final Safety Analysis Report[18,19] (UFSAR).

3. NUCLEAR CODE SYSTEM

3.1 Introduction

Nuclear design calculations performed for Westinghouse reactors employ the EPRI-ARMP code system[1] and the CASMO-2 code[2] or the CASMO-3/SIMULATE-3P code system. A summary description of each code is given in Appendix A. The ARMP/CASMO-2 and the CASMO-3/SIMULATE-3 code sequences have been reviewed and approved by the NRC for use in the design of reload cores for the McGuire and Catawba Nuclear Stations by Duke Power[28,32].

Presented in this section will be a description of the sequence, cross section preparation and parameterization, and reload design modeling procedures.

The nuclear calculational system enables the nuclear engineer to numerically model and simulate the reactor core. The ARMP/PDQ code system sequence used by Duke Power for McGuire and Catawba is outlined in Figure 3-1. The CASMO-3/SIMULATE-3 code system sequence is outlined in Reference 28.

3.2 Sources of Input Data

The determination of nuclear fuel loading patterns and core physics characteristics requires an accurate database consisting of:

1. Core operating conditions
2. Dimensional characteristics
3. Composite materials and mechanical properties
4. Nuclear cross sections

The UFSAR, supplemented by vendor reports and open literature, is the primary source of data for Items 1 to 3. These data are used as input to the cross section generators and core simulators

The cross section generators CASMO-2 and EPRI-CELL[8] use processed ENDF/B libraries unique to each code.

EPRI-CELL is a unit cell lattice code which is used to calculate few-group cross sections for fuel and non-fuel compositions as shown in Table 3-1.

CASMO-2 uses a processed version[9] of the ENDF/B-3 library. Group cross sections of σ_a, σ_f, $\nu\sigma_f$, σ_{tr}, scattering kernels, resonance integrals, and fission product data are among the data contained in this library. The 69 group library is divided into 14 fast, 13 resonance, and 42 thermal energy range groups. A 25 group version of this library is also used.

The EPRI-CELL library is derived from the ENDF/B-4 library[10]. The 97 energy groups are divided into 62 fast groups and 35 thermal groups.

CASMO-3 cross section development for SIMULATE-3 is described in Reference 28.

3.3 Cross Section Preparation

In order to model the neutronics of a reload core, it is necessary to generate a set of cross sections for use in a diffusion theory code. CASMO-3, CASMO-2, and EPRI-CELL are the cross section generators that may be used.

Inputs which are provided to these codes are: lattice materials and geometry, temperatures for fuel, clad, and moderator, effective resonance temperature, fuel enrichment, soluble boron concentration, number of depletion steps, length of depletion steps, etc. Table 3.1 shows the core materials or compositions which are parameterized by CASMO-2 and EPRI-CELL.

PDQ requires pin cell cross sections calculated as described Sections 3.3.1 and 3.3.2.

Reference 28 describes the cross section and nuclear data requirements for SIMULATE-3.

3.3.1 Fuel Calculations

Calculations for fuel regions employ fixed fuel and moderator temperatures for the cell depletion. Restart calculations are performed at various burnups to parameterize fuel cell cross sections at varying moderator and fuel temperatures.

The output of EPRI-CELL and CASMO-2 consists of sets of broad group cross sections which characterize the regions of interest. Cross sections are then formatted into PDQ07 tableset structure using either NUPUNCHER[11] (1-dimensional parameterization), or MULTIFIT[12] (2 and 3 dimensional parameterization or g-factors). Cross sections from CASMO-2 are similarly formatted using CHART[13].

3.3.2 Non-Fuel Calculations

Cross sections for empty control rod guide tubes, reflector, instrument thimble, and the water gap are calculated with either EPRI-CELL or CASMO-2. Separate cross section sets are generated for various moderator temperatures.

Strong absorbers such as RCCA and BP require reaction rate matching to obtain diffusion theory equivalent cross sections. Calculations using CASMO-2 are performed for these strong absorbers where first a transport theory method determines absorption rates, and then a series of diffusion theory iterations are performed to calculate a g-factor such that the absorption rates agree between both types of flux solutions. These g-factors are then incorporated in the tabulated cross sections.

RCCA cross sections are evaluated at BOL HFP conditions, while BP cross sections are evaluated with an HFP depletion calculation.

In both types of g-factor calculations, lattices with expected core average enrichments are used. Core baffle cross sections are also calculated with CASMO-2. A lattice geometry is employed, with the baffle material density

modified to reflect real versus modeled thickness in the quarter core PDQ07 discrete pin model.

3.4 PDQ07 Models

The PDQ07 few-group diffusion-depletion code is employed for core modeling. Two different models are used.

The first is the assembly colorset model, which is used for calculating k_∞ and M^2 data for EPRI-NODE-P 3-D simulations. The second model is the quarter-core model, which is used for X-Y power distribution calculations and for normalization of EPRI-NODE-P radial power distributions.

Aspects which are common to both PDQ07 models are:

1. Discrete pin representation
2. Two-group cross sections
3. Mixed Number Density thermal group constants
4. Improved Removal Treatment removal cross sections
5. Microscopic cross section parameterization for uranium, plutonium, burnable absorber, soluble boron, xenon, samarium, and lumped fission products
6. Thermally expanded geometry - pin pitch and assembly pitch

3.4.1 Colorset PDQ07 Modeling

The colorset PDQ07 model consists typically of four quarter assemblies arranged such that a representative neutron spectrum is obtained. Figure 3-2 shows a typical colorset geometry.

To accommodate asymmetric burnable poison rod loadings, full or half assembly geometries are used. The EPRI-ARMP PWR Procedures[14] are used for modeling, and most of the conventions and guidelines are employed.

Fuel types are determined according to enrichment and BPRA loading. k_∞ and M^2 data for each fuel type are calculated by performing the following operations:

I. BOL Cases - 0 MWD/MTU

A. k_∞ and M^2 - unrodded vs. T_{mod} (Inlet, Average, Outlet)

B. k_∞ and M^2 - rodded vs. T_{mod} (Inlet, Average, Outlet)

C. Boron worth

D. Doppler worth

II. Depletion Data - Exposure dependent data

A. Nominal HFP depletion at constant Tmod, Tfuel

B. Branch cases from depletion

1. Boron worth
2. Control rod worth
3. Equilibrium Xenon worth
4. Doppler worth
5. Moderator temperature worth

In the above PDQ07 branch calculations, only one parameter is varied, allowing a partial derivative of reactivity with respect to that parameter to be calculated.

The parameterization procedure involves approximately 150-200 cases, depending on the number of depletion steps.

The output from the PDQ07 colorset cases is written to PDQ07 integral files which in turn are processed by the linking codes EPRI-FIT[15] and SUPERLINX[16] to yield B-constant data for EPRI-NODE-P.

3.4.2 Quarter Core PDQ07 Model

Two-dimensional X-Y core simulations are performed with a discrete pin PDQ07 model. Assembly average and maximum pin powers are calculated, along with

critical boron concentrations and other reactivity parameters. Moderator and Doppler feedbacks are incorporated in this model.

The geometry employed utilizes thermally expanded dimensions. Figure 3-3 shows a geometry of a fuel assembly and water gaps. Figure 3-4 shows the complete quarter core mesh layout.

The plane of solution used in quarter core analyses is the axial midplane or the six foot level of the active fuel. Moderator and Doppler feedbacks are employed as described in Reference 17.

The depletion calculation is used to determine burnup dependent parameters. The soluble boron concentration is modified at each timestep such that the reactor is approximately critical.

Timesteps are taken using point depletion so that the core average exposure advances by: 150, 500, 1000, 2000, ..., N * 2000 MWD/MTU until the end of cycle is reached.

PDQ07 depletion calculations are used to determine the following parameters:

1. Assembly average and maximum pin powers
2. Core reactivity
3. Nuclide reaction rates: Fission and absorption
4. Nuclide inventories
5. Neutron flux distributions

Other calculations performed with the quarter core model may include:

1. RCCA bank worths
2. Boron and xenon worths
3. Power deficits
4. Moderator and Doppler temperature coefficients

Cases 2, 3, and 4 are usually performed with a nodal code; however, these are shown to demonstrate the quarter core model's flexibility.

3.5 EPRI-NODE-P Model

EPRI-NODE-P is the nodal code employed for three-dimensional analyses and reactivity studies. A summary description of EPRI-NODE-P is given in Appendix A. Typical calculations which are performed with the Duke Power EPRI-NODE-P model are:

1. Full core ejected rod worths
2. Power deficits
3. Differential rod worths
4. Axial xenon transients
5. Three-dimensional power distributions, etc.

The quarter core model uses one radial node per assembly and eighteen axial nodes.

Each unique combination of enrichment and BPRA loading comprises a separate fuel type. The fuel type is parameterized by sets of fitting coefficients which determine reactivity due to control rods, exposure, soluble boron, xenon, etc. Doppler and moderator feedbacks are explicitly treated.

EPRI-NODE-P radial power distributions are normalized near the beginning of cycle. Assembly average powers are adjusted to match quarter core PDQ07 calculations with radial albedoes - α_H and an internal leakage factor - g_H. The axial power distribution, is adjusted using vertical leakage factors α_V determined from comparisons of calculated and measured axial power distributions from benchmark core follow calculations.

Sections 5, 6, 7, and 9 discuss in depth calculational procedures of EPRI-NODE-P. Sections 10 and 11 address benchmarking of EPRI-NODE-P and PDQ07 calculations to measured power and reactivity data.

3.6 SIMULATE-3 Model

The SIMULATE-3 methodology used by Duke Power Company is described in Reference 28.

TABLE 3-1

EPRI-CELL/CASMO-2 Cross Sections
Calculation by Composition

1. EPRI-CELL

   a. Uranium Fuel
   b. Empty Control Rod Guide Tube/Instrument Tube
   c. Reflector
   d. Water Gap

2. CASMO-2

   a. Burnable Poison Rod Assembly
   b. Gadolinia doped Uranium Fuel
   c. Control Rod - AgInCd or B_4C
   d. Baffle

FIGURE 3-1

Nuclear Flow Chart for EPRI-ARMP

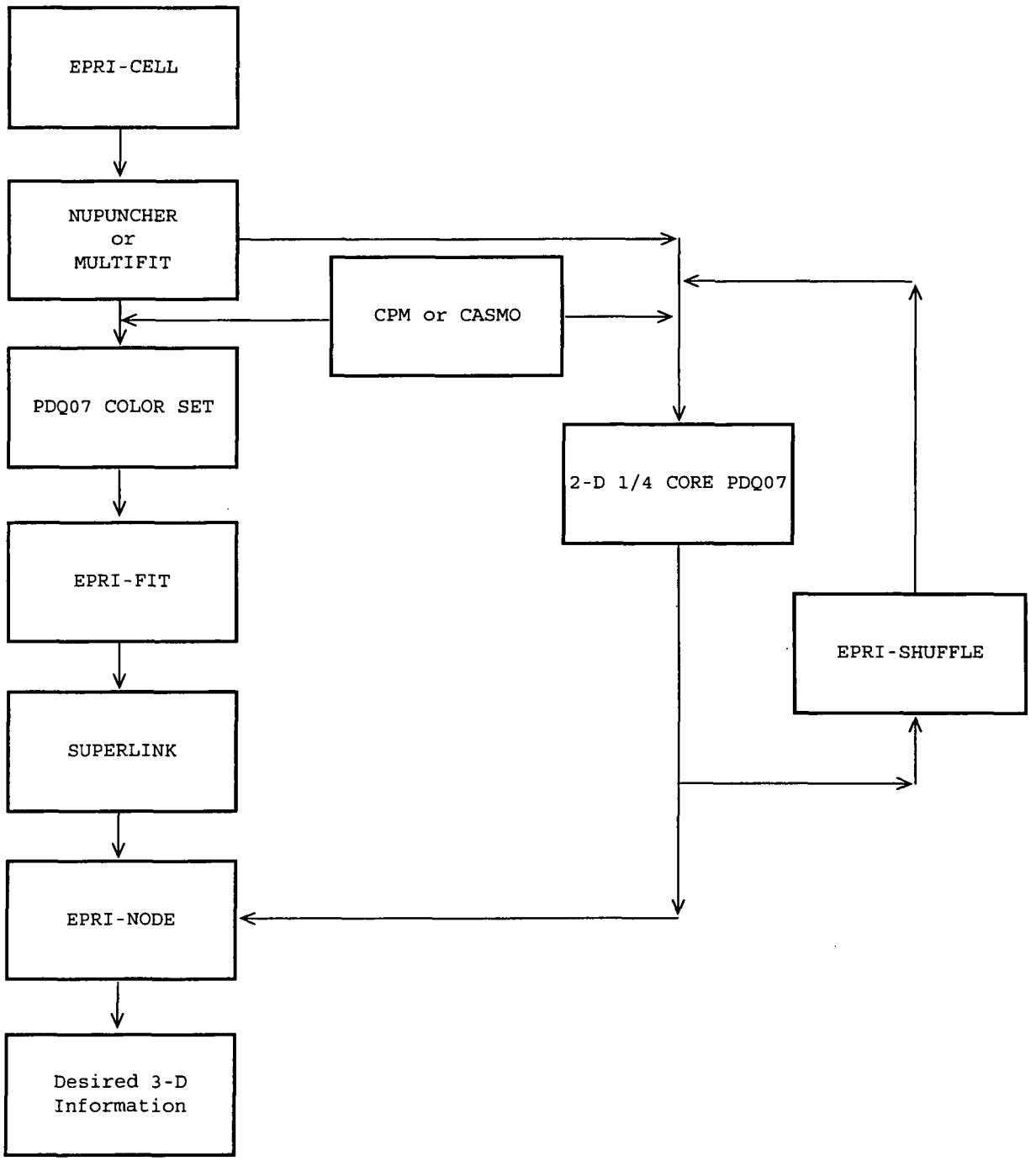
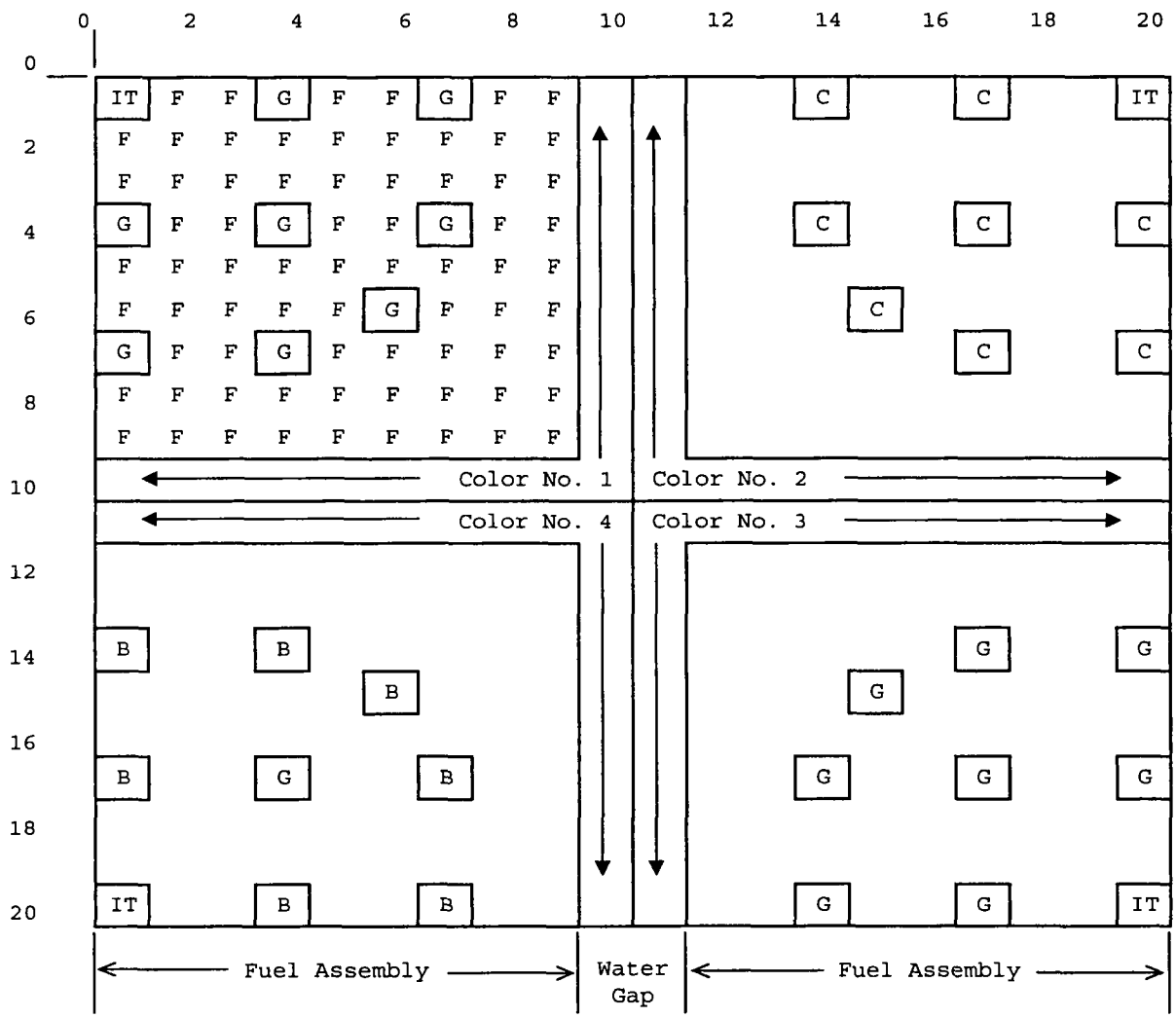


FIGURE 3-2

17 x 17 Assembly PDQ07 Colorset Geometry



LEGEND:

IT Instrument Tube

C Guide Tube With Control Rod

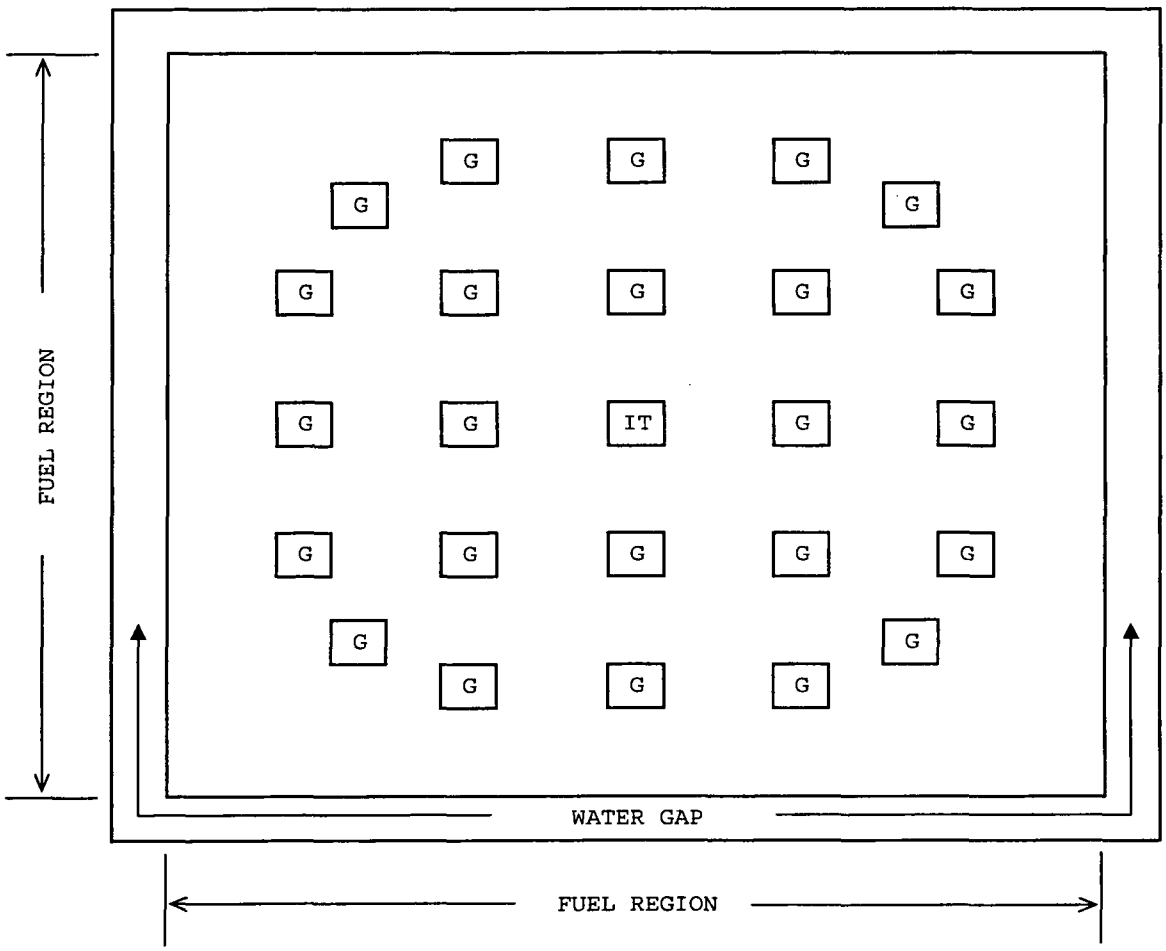
B Guide Tube With Burnable Poison Rod

G Empty Guide Tube

F Fuel Rod

FIGURE 3-3

PDQ07 Quarter Core Model Assembly Geometry



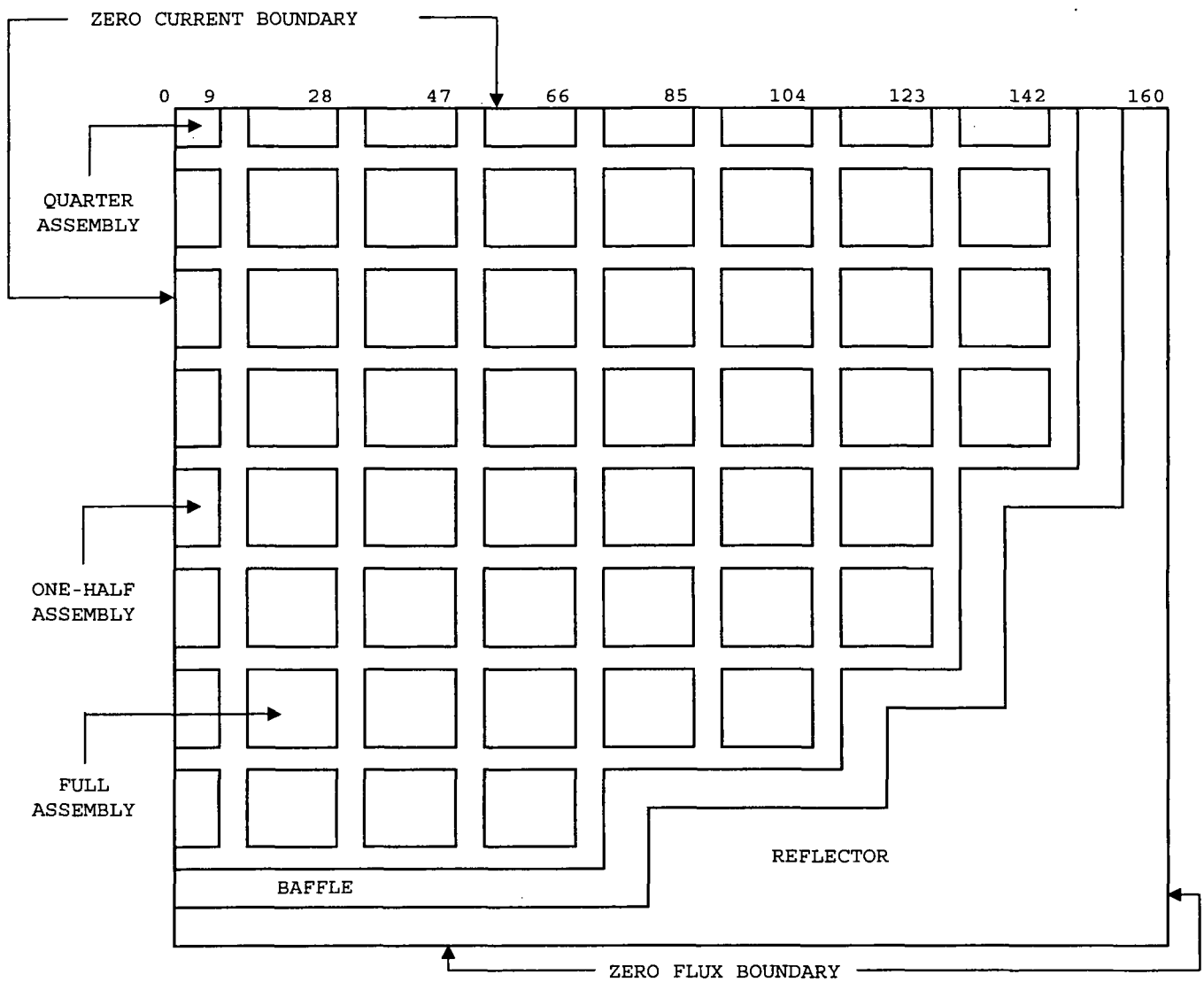
LEGEND:

IT Instrument Tube

G Control Rod Guide Tube

FIGURE 3-4

PDQ07 Quarter Core 17 x 17 Geometry



4. FUEL CYCLE DESIGN

4.1 Preliminary Fuel Cycle Design - Initialization

To commence the design of a reload, core operation requirements along with planned changes in reactor primary or secondary systems are assembled. A preliminary loading pattern is designed which meets operational requirements. Physics data from the preliminary design are compared with core operating requirements to determine the adequacy of the reload design. Likewise, physics data are compared to Technical Specifications to verify that the preliminary design will conform to existing limits.

4.1.1 Review of Design Information

The preliminary design procedure requires assembly of design information which in turn will determine the cycle's operational capabilities. Typical design data are shown on Table 4-1.

Table 4-1 and other pertinent nuclear design data are assembled and reviewed for consistency with previous sets of design data.

4.1.2 Determination of Cycle-Specific Operating Requirements

Design data from Table 4-1 uniquely determines expected operating requirements and capabilities. For instance, a longer than annual cycle may require a low leakage loading pattern and the use of burnable absorber rods. A larger energy requirement than can be provided by normal operation with a given reload enrichment may require a planned power coastdown at end of cycle. Similarly, other design considerations will govern the rest of the cycle-specific operational characteristics.

4.1.3 Preliminary Loading Pattern and Reload Region Determination

The purpose of a preliminary loading pattern analysis is to determine the uranium and separative work requirements to meet a desired cycle lifetime. The cycle lifetime is confirmed by modeling the depletion of a reload core. If the number of new fuel assemblies and the enrichment are known, this analysis will yield an estimate of the cycle lifetime.

4.2 Final Fuel Cycle Design

Having determined the number and enrichment of the fuel assemblies during the preliminary fuel cycle design, the final fuel cycle design (FFCD) concentrates on optimizing the placement of fresh and burned assemblies and burnable poison assemblies (if any) to result in an acceptable fuel cycle design. It must meet the following design criteria with appropriate reductions to account for calculational uncertainties:

1. $F^N_{\Delta H}$ must meet the limits specified in the Technical Specifications.
2. Moderator Temperature Coefficient must meet the limits specified in the Technical Specifications.
3. Maximum fuel burnup must be less than the limits applicable for the type of fuel being used.
4. Shutdown Margin must meet limits specified in the Technical Specifications.
5. Maximum linear rod power must meet the limit specified in the Technical Specifications.
6. UFSAR Chapter 15 related physics parameters must be validated.

A preliminary verification of the above is made in the FFCD. Final verification of the above is made in fuel mechanical performance analyses, thermal and thermal-hydraulic analyses, safety-analysis physics parameters analyses, and maneuvering analyses.

4.2.1 Fuel Shuffle Optimization and Cycle Depletion

The preliminary fuel shuffle scheme is modified to minimize power peaking. This is accomplished by a trial and error type search until an acceptable BOC power distribution results.

The reload design's "burnup window" is assessed to ensure that applicable safety criteria are met.

The cycle is then depleted to various times in the cycle to verify that power peaking versus burnup remains acceptable. The shuffling variations include rearranging the location of the burned or fresh fuel assemblies, BP placement, and rotation of the spent fuel assemblies. These calculations are typically performed assuming quarter core symmetry.

The core neutronic model resulting from the FFCD is the core model used for other nuclear design calculations.

The shuffle pattern determined in the FFCD may later need to be modified based upon results obtained in the remaining nuclear calculations.

4.2.2 Rod Worth Calculations

Control rods serve several functions in the McGuire and Catawba reactors. The primary function is to provide adequate shutdown capability during normal and accident conditions. They are also used to maintain criticality during power maneuvers and to maintain the Axial Flux Difference (AFD) within Technical Specification limits. Since the presence of control rods influences both power distributions and criticality, it is necessary in many calculations to evaluate not only the reactivity effect but also the perturbation that a given rod configuration has on the power distribution.

McGuire and Catawba are typically operated in the ARO or feed and bleed mode. All RCCA have full length absorber rods. During full power operation, Control

Bank D is typically inserted about six inches (215 steps withdrawn) in the active core. Control Bank D is used to control power during load follow maneuvers, and in conjunction with Control Banks B and C, to achieve criticality during startup.

Calculations of control rod worth and power peaking (F_Q) are used in the safety analysis of the reload core. The calculations discussed in subsequent sections include the following:

1. Control Rod Worths
2. Shutdown Margin
3. Ejected Rod Analysis
4. Dropped Rod Analysis

4.2.2.1 Control Rod Worths

RCCA bank locations in McGuire and Catawba usually are fixed and do not change from cycle to cycle. The worth of each control bank (A, B, C, D) is calculated at BOC and EOC, at HFP and HZP. The total rod worth (ARI) is calculated at BOC, EOC, and any limiting burnup at HZP only for use in the shutdown margin calculation.

4.2.2.2 Shutdown Margin

Searches for the highest worth stuck rod are performed at BOC, EOC, or any limiting burnup for HZP conditions using full core calculations.

Table 4-2 summarizes the results of a shutdown margin calculation. The total rod worth described in section 4.2.2.1 is shown as Item 1. Item 2 is the worth of the highest stuck rod. The total worth reduced by the stuck rod worth is shown as the net worth (Item 3). A calculational uncertainty of 10% is subtracted off in Step 4, and Step 5 shows the available rod worth.

The required rod worth is calculated next in Steps 6-9. The power deficit obtained by modeling the core at HFP and HZP (using constant boron and xenon) and subtracting the reactivities is shown as Item 6. This reactivity insertion accounts for Doppler and Moderator deficits. The maximum allowable inserted rod worth, Item 7, is obtained from the allowable rod insertion and the integral rod worth curve for that insertion. This accounts for the maximum allowed rod insertion at HFP. An axial flux redistribution occurs when the power level is reduced from HFP to HZP. This redistribution causes an increase in reactivity. If Item 6 is calculated using a 3-D model, no additional penalty is required. If Item 6 was calculated using a 2-D model, where redistribution effects are not modeled, an additional reactivity penalty is assessed as Item 8. The sum of these required worths (Item 9) is the total required worth.

Additional reactivity penalties are applied to both the power defect and the rod insertion allowance to account for xenon redistribution effects.

The shutdown margin is shown as Item 10 and is defined as the total available worth minus the total required worth. Shutdown margin requirements are specified in the Technical Specifications.

4.2.2.3 Rod Insertion Limit Verification

As part of the reload design procedure, the Rod Insertion Limits are verified for applicability in the reload core (see Section 5.5).

4.2.2.4 Ejected Rod Analysis

The UFSAR[18,19] presents the limiting criteria for the ejected rod accident. The accident has been analyzed at HFP and HZP conditions at BOL and EOL.

Ejected rod calculations are performed on a cycle-specific basis to verify that UFSAR accident analysis values are not exceeded.

Calculational limits are established using the methodology described in References 30 and 31.

To verify that the ejected rod parameters are within calculational limits, ejected rod calculations are performed at BOC and EOC or at other limiting times in cycle life at both HFP and HZP.

The HZP ejected rod calculations are performed using full core geometry with Control Banks B and C at their insertion limits in the core and with Control Bank D fully inserted. Single rods in Control Banks D, C, and B are removed in subsequent cases and the worth of the ejected rod is calculated by subtracting the reactivities of the cases before and after the rod was removed. The fuel and moderator temperature is held constant and equal to the HZP moderator temperature for these calculations. The highest worth calculated by the above procedure is the worst ejected rod at HZP. If the ejected rod worth exceeds the calculational limit, one of the following is performed: an evaluation, a revision to the rod insertion limits, a reanalysis of the Chapter 15 REA analysis, or a redesign of the core loading pattern.

The HFP ejected rod calculations are performed in a similar manner to the HZP calculations with the exceptions that only Control Bank D is inserted at the HFP insertion limit and that the fuel temperature and moderator temperatures correspond to those of HFP conditions. The HFP ejected rod worths are determined without thermal feedback to be conservative. If the ejected rod worth exceeds the calculational limit, one of the following is performed: an evaluation, a revision to the rod insertion limits, a reanalysis of the Chapter 15 REA analysis, or a redesign of the core loading pattern.

A parallel analysis, addressing core peaking, is performed at the same time as the rod worth analyses. Additional discussion of the rod ejection accident analysis methodology is contained in Reference 30.

4.2.2.5 Dropped Rod Analysis

The UFSAR[18,19] presents the limiting criteria for the dropped rod accident. The calculational limits are established using the methodology described in Reference 30.

A core model is used that evaluates pre-drop and post-drop physics parameters for possible dropped rod combinations. The physics parameters important to the dropped rod analysis are presented in Reference 30.

4.2.3 Fuel Burnup Calculations

The reload design must meet fuel burnup limits. This is confirmed during the final fuel cycle design. Depletion calculations yield core, assembly average, single fuel rod burnups, and peak local burnups which can be compared to the design limits.

4.2.4 Reactivity Coefficients and Defects

Reactivity coefficients define the reactivity insertion for small changes in reactor parameters such as moderator temperature, fuel temperature, and power level. These parameters are input to the safety analysis and used in modeling the reactor response during accidents and transients. Whereas reactivity coefficients represent reactivity effects over small changes in reactor parameters, reactivity defects usually apply to reactivity inserted from larger changes typical of HFP to HZP. An example of a reactivity deficit is the power defect from HFP to HZP used in the shutdown margin calculation. A different way of looking at the terms is that the coefficient when integrated over a given range yields the defect, or the coefficient is the partial derivative of reactivity with respect to one specific parameter.

Coefficients of reactivity are calculated using the core model. First a nominal case is established at some reference conditions. Then one parameter of interest is varied up and/or down by a fixed amount in another calculation

and the resulting change in core reactivity divided by the parameter change is calculated as the reactivity coefficient.

4.2.4.1 Doppler Coefficient

The Doppler Coefficient is the change in core reactivity produced by a small change in fuel temperature.

The major component of the Doppler Coefficient arises from the behavior of the Uranium-238 and Plutonium-240 resonance absorption cross sections. As the fuel temperature increases, the resonances broaden increasing the chance that a neutron will be absorbed and thus decreasing the core reactivity.

If Case 1 represents the reference case with an effective fuel temperature T_1 (and K^1 effective) and Case 2 represents a second case where the fuel temperature has been increased or decreased by approximately 50°F and is T_2 (and K^2 effective), the Doppler Coefficient is mathematically calculated from the following equation:

$$\alpha_D = \frac{\left(\dfrac{K^1_{eff} - K^2_{eff}}{K^1_{eff} \times K^2_{eff}}\right)}{(T_1 - T_2)} \times 10^5 = \Delta\rho \ \text{(pcm/°F)}$$

Doppler Coefficients are calculated at various core conditions to validate safety analysis assumptions.

4.2.4.2 Moderator Temperature Coefficient

The Moderator Temperature Coefficient (MTC) is the change in reactivity produced by a small change in moderator temperature. In McGuire and Catawba, the average core moderator temperature increases linearly as power is escalated from 0 to 100% HFP. Therefore, for accident and transient analyses it is necessary to know the moderator temperature coefficient over a range of moderator temperatures from CZP to HFP.

These analyses are performed by modeling changes in the core average moderator temperature. Cases are run changing the moderator temperature from the reference temperature. If the cases and resulting $K_{effective}$'s are identified as Case 1 (T_{MOD1}, K^1_{eff}) and Case 2 (T_{MOD2}, K^2_{eff}) the moderator temperature coefficient is calculated from the following equation:

$$\alpha_{T_{MOD}} = \frac{\left(\frac{K^1_{eff} - K^2_{eff}}{K^1_{eff} \times K^2_{eff}}\right)}{\left(T_{MOD1} - T_{MOD2}\right)} \times 10^5 = \Delta\rho \ \ (\text{pcm}/^\circ\text{F})$$

Since the reload core is designed with a predetermined flexibility (burnup window), the MTC is verified to be within its design limit for the current cycle considering the burnup window of the previous cycle.

4.2.4.3 Isothermal Temperature Coefficient

The fractional change in reactivity due to a small change in core temperature is defined as the isothermal temperature coefficient (ITC) of reactivity. This is equal to the sum of the moderator and Doppler temperature coefficients and may be explicitly calculated at HZP for isothermal conditions (TFUEL=TMOD) by changing both the fuel and moderator temperatures from the reference HZP moderator temperature.

4.2.4.4 Power Coefficient and Power Defect

The power coefficient of reactivity is the core reactivity change resulting from an incremental change in core power level. The power defect is usually the total reactivity change associated with a power level change from HZP to HFP.

The power coefficient is defined by the following equation:

$$\alpha_P = \frac{\left(\frac{K^1_{eff} - K^2_{eff}}{K^1_{eff} \times K^2_{eff}}\right)}{\left(P_1 - P_2\right)} \times 10^5 = \Delta\rho \ \ (\text{pcm}/\%\text{FP})$$

where: K^1_{eff} is K-effective for the core at power P_1 (%)

K^2_{eff} is K-effective for the core at power P_2 (%)

Neglecting second order effects this equation is equivalent to the following:

$$\alpha_P = \alpha_{TMOD}\frac{\Delta TMOD}{\Delta P} + \alpha_D\frac{\Delta TFUEL}{\Delta P}$$

where: α_{TMOD} is the moderator temperature coefficient and α_D is the Doppler temperature coefficient.

Since the power coefficient should include flux redistribution effects resulting from axial variations in burnup and isotopics as well as non-uniform fuel temperature distributions, it should be performed using a 3-D simulator with thermal hydraulic feedback. If the calculation is performed using a 2-D model then it should be corrected for the 3-D effects.

A typical power coefficient calculation for HFP would proceed in the following manner: The HFP case is run, and the core Keff is calculated (K^1_{eff}). Then a second case is run with the core power level reduced while holding control rods, boron, and xenon constant. The K_{eff} from this case, K^2_{eff}, is used along with the results from the reference case to calculate the power coefficient:

$$\alpha_P = \frac{\left(\frac{K^1_{eff} - K^2_{eff}}{K^1_{eff} \times K^2_{eff}}\right)}{(P_1 - P_2)} \times 10^5 = \Delta\rho \ \text{(pcm/\%FP)}$$

The power defect is calculated for use in the shutdown margin calculation (see Section 4.2.2.2) and is the reactivity change from HZP to HFP. This calculation should be performed in three dimensions to satisfactorily model the axial flux redistribution, however, a two dimensional calculation may be performed and corrected for this flux redistribution phenomenon. These calculations are usually performed at BOC and EOC.

Both HFP and the HZP cases should have the equilibrium xenon concentration corresponding to HFP. The power defect is calculated from the following equation:

$$\text{Power Defect (HFP to HZP)} = \left(\frac{K^1_{eff} - K^2_{eff}}{K^1_{eff} \times K^2_{eff}}\right) \times 10^5 = \Delta\rho \text{ (pcm)}$$

where: K^1_{eff} is K-effective at HZP and

K^2_{eff} is K-effective at HFP

4.2.4.5 Miscellaneous Coefficients

For reload design, certain other coefficients of reactivity are not routinely calculated. These include moderator density coefficient, moderator pressure coefficient, and moderator void coefficient. These coefficients can be calculated in an analogous manner by varying the appropriate core reactivity parameters.

4.2.4.6 Boron Related Parameters

Critical boron concentrations for various core conditions during cycle lifetime are calculated using the core model. Table 4-3 lists typical conditions that critical boron concentrations and boron worths are calculated. In addition to these, an ARO critical boron letdown curve is generated for HFP EQXE.

4.2.4.7 Xenon Worth

Xenon worth calculations are performed to support plant operation (e.g. startup after trip), rather than as a safety parameter. Xenon worth is calculated as a function of burnup. The equilibrium xenon worth is calculated as the difference in reactivities between the equilibrium and no xenon cases. The peak xenon worth is calculated as the difference between the peak and no xenon cases. The peak xenon worth is determined at approximately 8 hours following a reactor shutdown from HFP, equilibrium xenon conditions.

4.2.4.8 Kinetics Parameters

The kinetics behavior of the nuclear reactor is often described in terms of solutions to the kinetics equation for six effective groups of delayed neutrons. Transient and accident analyses often involve kinetic modeling of the reactor core. The rate of change in power from a given reactivity insertion can be calculated by solving the kinetics equations if the six group effective delayed neutron fractions, the six group precursor decay constants, and the prompt neutron lifetime are known.

PDQ07 and DELAY may be used to calculate these parameters[20]. PDQ07 is used to obtain spatially averaged isotopic fission rates as a function of burnup and DELAY calculates kinetics parameters and then uses these parameters to solve the Inhour equation and thereby relate the stable reactor period to the reactivity insertion. CASMO-3 data libraries contain delayed neutron data, and SIMULATE-3 is capable of calculating the core averaged kinetics parameters of interest.

Calculations are performed at BOL and EOL. The sum of the six group $\beta_i$effective, βeffective, for the new reload cycle is compared to those values used in the UFSAR.

4.2.5 Assessment of the Fuel Cycle Design

Once the FFCD calculations are performed, the resultant data are assessed for validity and consistency with core operation requirements as well as fuel design and safety analysis limits.

Design criteria for a reload design are outlined in Section 4.2. A preliminary verification of these criteria or parameters important to these criteria is made in the FFCD. Additional calculations that validate a reload design are described in Sections 5 - 7 of this report.

TABLE 4-1

Nuclear Design Data
For Reload Design

1. Power operation mode: load follow or base load.

2. Vessel internal or core component modifications.

3. Expected minimum and maximum cycle burnups.

4. Feed enrichment (if already contracted for).

5. Number and design of feed assemblies.

6. RCS hydraulic conditions.

TABLE 4-2

Shutdown Margin Calculation

		BOC, % Δρ
Available Rod Worth		
1.	Total rod worth, HZP	6.46
2.	Maximum stuck rod, HZP	-1.39
3.	Net Worth	5.07
4.	Less 10% uncertainty	.51
5.	Total available worth	4.56
Required Rod Worth		
6.	Power defect, HFP to HZP	.88
7.	Max allowable inserted rod worth	1.36
8.	Flux/Xenon redistribution	.63
9.	Total required worth	1.87
10.	Shutdown Margin (total avail. worth minus total required worth)	2.69

NOTE: Required shutdown margin is specified in the Technical Specifications.

TABLE 4-3

Typical Boron Parameters

Critical Boron - ppm

HZP, ARO, BOC, No Xenon

HZP, Bank D inserted, BOC, No Xenon

HZP, Bank D + C inserted, BOC, No Xenon

HFP, ARO, EQXE vs exposure

Boron Worth - ppm/%$\Delta\rho$

HFP, EQXE, ARO vs. exposure

HZP, NOXE, ARO vs. exposure

Boron Worth Versus Boron Concentration - HZP, NOXE

BOC

EOC

5. NODAL ANALYSIS METHODOLOGY

5.1 Purpose and Introduction

Nodal analysis allows for modeling of the reactor core in three-dimensions and for performing calculations which because of either code restraints or economic restraints cannot be performed by any other means. Examples of nodal code capabilities include:

1. Calculations which need a three-dimensional geometry such as differential rod worths, axial xenon transients and three-dimensional power distributions.

2. Calculations which need a full-core geometry such as stuck and ejected rod worths.

This section addresses the role of a nodal code in performing cycle depletions, generating rod worth data, determining shutdown margins and shutdown boron concentrations, setting control rod insertion limits, and determining trip reactivity worths and shapes.

A nodal code is also used to calculate many of the startup test parameters and core physics parameters described in Section 9 of this report.

The nodal codes used for McGuire and Catawba analyses are EPRI-NODE-P and SIMULATE-3. (See descriptions in Section 3 and Appendix A).

EPRI-NODE-P can be run with either a quarter-core or a full-core geometry. The McGuire and Catawba models utilize one radial node per assembly and twelve to eighteen axial nodes. EPRI-NODE-P radial powers are normalized to the two-dimensional PDQ07 assembly powers near the beginning of each cycle.

The SIMULATE-3 model is described in Reference 28.

5.2 Fuel Cycle Depletion - Nodal Code

A fuel cycle depletion is performed for each cycle using nodal analysis. For EPRI-NODE-P, the nodal radial powers are normalized to the two-dimensional quarter core PDQ07 or SIMULATE-3 powers at various conditions. SIMULATE-3 does not require normalization. The nodal core model is then depleted from BOC to EOC at appropriate burnup intervals. This depletion is typically performed in the critical boron search mode, with nominal rod insertion (usually 215 SWD) and equilibrium xenon.

Data files may be saved at each burnup step throughout the cycle depletion. These files contain records of the power, exposure, and xenon concentration for each node in the core.

As a result of the nodal core depletion, the following data is obtained:

1. Two and three-dimensional power distributions at each burnup step.

2. A boron letdown curve, i.e., critical boron concentrations as a function of burnup.

3. Axially-dependent parameters such as offset or axial flux difference as a function of burnup.

4. Assembly exposures as a function of core-averaged burnup.

5.3 Rod Worth Analysis

Nodal analysis is used to calculate various rod worths which require three-dimensional capabilities. These calculations include differential rod worths and integral rod worths.

5.3.1 Differential Rod Worth Analysis

Differential rod worths are calculated as a function of rod insertion. The differential rod worth is defined as the change in reactivity associated with a small change in rod position. This rod worth is determined by running two cases at different rod insertions with all other parameters held constant (power, burnup, xenon, boron) and then by dividing the reactivity difference by bank height difference.

Differential rod worths for the control banks are calculated at HZP and HFP, at BOC and EOC, and at no xenon, equilibrium xenon, and peak xenon conditions. The rod banks are inserted both sequentially and in 50% overlap.

5.3.2 Integral Rod Worth Analysis

Integral rod worths are defined as the integral of the differential rod worth data. Integral rod worths are determined by summing up the reactivities resulting from the differential rod worth analysis. Total integral rod worths for a rod bank can be calculated either with a two-dimensional or three-dimensional code by subtracting the reactivities resulting from cases where the rod bank is out and then in (other parameters held constant). However, in order to get the integral rod worth as a function of rod position, i.e., the shape of the rod worth curve, the three-dimensional nodal code is used.

Integral rod worth calculations for the control banks are performed at HZP and HFP, at BOC and EOC, and at no xenon, equilibrium xenon, and peak xenon conditions. The rod banks are inserted sequentially with 50% overlap. The total rod worth (ARI) is calculated at BOC, EOC, and any limiting burnup at HZP for use in the shutdown margin calculation.

5.4 Shutdown Margin Analysis

5.4.1 Shutdown Margin

Shutdown margin calculations are described in Section 4.2.2.2. Table 4-2 summarizes the results of a shutdown margin calculation.

The calculation consists of:

1. The total rod worth (ARI) at HZP, BOC, and EOC (Item 1 in Table 4-2). This worth is determined by running cases at ARO and ARI (with constant boron and xenon) and subtracting the reactivities.

2. The maximum stuck rod worth at HZP, BOC, and EOC (Item 2 in Table 4-2). Utilizing full-core capabilities, the worth of the worst stuck rod is determined by subtracting the reactivities between two cases, one with ARI, the other with ARI and the stuck rod out.

3. The power deficit from HFP to HZP, at BOC and EOC (Item 6 in Table 4-2). This deficit is determined by running cases at HFP and HZP (with constant boron and xenon) and subtracting the reactivities. This reactivity insertion accounts for Doppler and Moderator deficits, and for axial flux redistribution.

4. The maximum allowable inserted rod worth at HFP, BOC, and EOC (Item 7 in Table 4-2). This worth is obtained by reading the integral rod worth curve at the rod insertion limits (See Section 5.3.2).

Additional reactivity penalties are applied to both the power defect and the rod insertion allowance to account for xenon redistribution effects.

5.4.2 Shutdown Boron Concentration

The shutdown boron concentration is another parameter that is determined using three-dimensional nodal analysis. Since the shutdown margin is determined based on the worst case stuck rod out of the core with all other rods in, the full-core capability of the nodal code is needed.

The nodal code is first used to determine the worst case stuck rod by calculating the worth of various rods in the core. After the worse case stuck rod is determined, a boron search case is performed at the ARI-stuck rod out conditions. This boron concentration is adjusted based on boron worth results until the core reactivity reflects the appropriate margin (1.3% $\Delta\rho$ for temperatures greater than 200°F, 1.0% $\Delta\rho$ for temperatures less than or equal to 200°F). The resulting boron concentration is the shutdown boron concentration required for the conditions modeled in the nodal code. This calculated boron concentration is conservatively increased by a boron equivalent of 10% of the ARI-stuck rod out worth and by at least an additional 100 ppm.

A shutdown boron concentration can be determined for any moderator temperature provided the input cross sections remain valid. Typical average moderator temperatures for which shutdown boron concentrations are provided are 68°F, 200°F, 500°F, and the HZP average moderator temperature (approximately 557°F).

5.5 Rod Insertion Limit Assessment

Control rod insertion limits define how deep the control rods may be inserted into the core during normal operation as a function of the power level. It is a Technical Specification requirement that the rods not be inserted deeper than the established limits. This analysis is usually a verification that the Rod Insertion Limits from cycle N-1 are adequate for cycle N.

The control rod insertion limits are determined based on:

1. Maintaining the required minimum shutdown margin, as specified in the Technical Specifications, throughout the cycle life.

2. Maintaining the maximum calculated power peaking factors within the limit specified in the Technical Specifications.

3. The acceptability of the UFSAR Chapter 15 accident analyses.

Determining control rod insertion limits involves an iterative process based on satisfying the above criteria. This process begins with insertion limits from the previous cycle.

The first requirement for insertion limits is that of satisfying the reactivity constraints, i.e., maintaining the required shutdown margin. The insertion limits from the previous cycle, along with integral rod worth curves for control banks in ~50% overlap for the current cycle, are used to calculate the maximum allowable inserted rod worth for input into the shutdown margin calculation. The shutdown margin is calculated at BOC, EOC, and any limiting burnup in order to determine if the control rod insertion limits are acceptable. If the shutdown margin criteria is not satisfied, the insertion limits are adjusted until satisfactory margin is obtained or the core is redesigned.

The insertion limits also have to satisfy the peaking factor constraints. For ARMP methods, the nodal powers are synthesized with discrete pin PDQ07 pin powers to give values of power peaking at various power levels from HZP to HFP. For SIMULATE-3, power peaking factors are calculated directly. The power peaking values are then compared to the Technical Specification limits. If the Technical Specification limits are not satisfied, the control rod insertion limits are adjusted until satisfactory power peaking values are obtained, or the core is redesigned

In addition to satisfying reactivity and peaking factor constraints during normal operation, the control rod insertion limits may need to be modified based on the worst case consequences of an ejected RCCA, a dropped RCCA or a statically misaligned RCCA. Evaluations are performed with the nodal code to identify the worst case rod configuration during a withdrawal or misalignment event, that is, to identify the single RCCA which produces the maximum peaking factor (control rods held at insertion limits). The results of the three-dimensional nodal analysis with these worst case rod configurations are compared to the design criteria associated with each event. The acceptability of the control rod insertion limits is dependent on the criteria being satisfied.

5.6 Trip Reactivity Analysis

The minimum trip reactivity and the shape of the trip reactivity insertion curve (inserted rod worth as a function of rod position) are both generated using nodal analysis. These parameters are needed to perform the safety analysis for various UFSAR Chapter 15 accidents/transients.

5.6.1 Minimum Trip Reactivity

The minimum trip reactivity is the minimum amount of reactivity available to be inserted into the core in the event of a reactor trip. It is evaluated for each reload core to ensure that the previously set limits are still valid.

The minimum trip reactivity is calculated at BOC and EOC at HFP and HZP conditions. The minimum trip reactivity is the total rod worth reduced by (1) the most reactive stuck rod worth, (2) a 10% uncertainty on available rod worth, and (3) the rod insertion allowance including applicable penalties to account for xenon redistribution. The rod insertion allowance is the amount of reactivity associated with the control rod insertion limits. It is the difference in reactivity between an ARO case and one with control rods at their insertion limits. A sample BOC calculation is shown in Table 5-1.

5.6.2 Trip Reactivity Shape

The shape of the trip reactivity insertion curve defines the inserted rod worth as a function of rod position. The most limiting shape is the one which defines the minimum inserted rod worth as a function of rod position. This most limiting shape is evaluated each reload cycle to ensure that the values for the minimum inserted rod worth vs. rod position used in the safety analysis are still applicable.

The most limiting trip reactivity shape typically corresponds to the most bottom-skewed axial power shape. HFP axial power distributions are examined from BOC to EOC, with control rods at the full power rod insertion limits and the most reactive rod stuck out of the core. After the most limiting power shape is found, the N-1 control rods are inserted into the core in a stepwise manner. The results of this insertion yield the minimum inserted rod worth vs. position curve.

5.7 Assessment of Nodal Analyses

Once the nodal calculations are performed, the resultant data are assessed for validity and consistency with core operation requirements and safety limits.

TABLE 5-1

Example BOC Trip Reactivity Calculation

Trip Reactivity	HFP % $\Delta\rho$
Minimum Available N-1 Rod Worth	6.18
10% Rod Worth Uncertainty	-0.62
Total Available Rod Worth	5.56
Rod Insertion Allowance	-1.13
Xenon Redistribution Penalty	-0.08
Minimum Trip Reactivity	4.35

6. CALCULATION OF SAFETY RELATED PHYSICS PARAMETERS

6.1 Safety Analysis Physics Parameters

With a reload of fresh fuel, a reactor core's physics characteristics are altered in three major areas:

1. Power distribution
2. Control rod worths
3. Kinetics

Each of the above has its own subset of specific parameters. These core physics parameters are considered in the UFSAR Chapter 15 safety analyses. The core physics parameters whose values have an important influence on the course or the consequences of the accidents analyzed in the UFSAR are designated as safety analysis physics parameters. Reference 30 identifies these parameters, describes the approach for calculating the values of these parameters, and discusses the manner in which the reload values are evaluated for acceptability with respect to the accident analysis assumptions.

6.2 Core Power Distributions

As part of the reload design, detailed analyses of the core power distributions are performed for core conditions of normal operation and anticipated transient conditions. These analyses are performed:

1. to confirm that the power peaking factors assumed as initial conditions for certain accidents remain valid,
2. to verify that certain transient induced power peaks will be acceptable for the fuel design thermal limits, and
3. to facilitate the selection of the operating limits and protection system setpoints.

The methodology for performing these analyses is presented in Reference 29.

6.3 Power Peaking Factors and Reliability Factors

Power peaking factors to be compared to a design limit are conservatively increased by total peaking reliability (or uncertainty) factors. The peaking reliability factor is determined by statistically combining manufacturing and calculation uncertainties. Additional potential power peaking uncertainties may be included for things such as rod bowing, etc.

The general formulation for the peaking reliability factor is:

$$\text{Peaking Reliability Factor} = 1+\text{BIAS}+\sqrt{(UC^2+Ux1^2+Ux2^2+...)} \qquad (6\text{-}1)$$

Where:

UC - Calculation Uncertainty

For the Pin Total Peak (F_Q): $UC^2 = UT^2 + URL^2$

For the Pin Radial Peak ($F_{\Delta H}$): $UC^2 = UR^2 + URL^2$

For the Assembly Axial Peak (F_Z): $UC^2 = UA^2$

UT - Total Peaking Uncertainty

URL - Assembly Radial Local (or Pin) Power Peaking Uncertainty

UR - Assembly Radial Power Peaking Uncertainty

UA - Assembly Axial Power Peaking Uncertainty

Uxi - Additional Uncertainties, e.g. manufacturing tolerance, rod bow, etc.

BIAS - Calculation Bias

For the PDQ07 code methodology, Section 8 contains the calculation of the radial local uncertainty factor. For the EPRI-NODE-P code based on ARMP methodology, Section 11 contains the calculation of the assembly and pin total peak and assembly and pin radial peak uncertainty factors. When the EPRI-NODE-P code is based on the CASMO-3/SIMULATE-3 methodology, Reference 28 presents the uncertainty factors. For the CASMO-3/SIMULATE-3 code methodology, Reference 28 contains the calculation of the uncertainty factors.

7. 3D POWER PEAKING ANALYSIS

As part of the reload design, detailed analysis of the core power distribution for normal operation and anticipated transient conditions are made. These analyses are performed (1) to confirm that the initial condition power peaking factors for certain accidents remain valid, (2) to verify that certain transient induced power peaks will satisfy the fuel design limits, and (3) to facilitate the selection of operating limits and RPS setpoints. The methods for performing these analyses are outlined in Reference 29.

8. RADIAL LOCAL ANALYSIS

8.1 Background

The radial local is an important factor in fuel cycle design because of its significant influence on LOCA and DNB analysis. The premise for performing this analysis is to evaluate the ability of PDQ07 to predict the radial local. The radial local is defined as the ratio of the maximum pin power, to the assembly average planar (x-y) power. It is used to calculate pin power by combining assembly power (F_Q or $F_{\Delta H}$) from the nodal analysis with the radial local factor by the equation shown below.

Pin Power = Nodal Power x Radial Local Factor

In the ARMP methodology, PDQ07 and CASMO-2 may be used to calculate radial local factors. PDQ07 is a 1, 2, or 3 dimensional two neutron energy group diffusion theory code, whereas CASMO-2 is a 2-dimensional multigroup transport theory code, which utilizes transport probabilities in the solution of the transport equation. The 2-dimensional PDQ07 code is the primary calculational tool used to model reactor cores (for additional information concerning the use of this code, refer to Section 3.4). Energy and burnup dependent Mixed Number Density (MND) cross sections used by PDQ07 are developed in accordance with ARMP[14] procedures. CASMO-2 is used primarily to generate multigroup constants (i.e., control rod and burnable absorber cross sections), and as a benchmark code.

SIMULATE-3 is capable of calculating pin power peaking directly and does not require the use of radial local factors (Reference 28).

8.2 Comparison of PDQ07 to CASMO-2 at Hot Full Power Condition

The predictive capability of PDQ07 was assessed by performing a series of eighth assembly calculations using both PDQ07 and CASMO-2. A typical Westinghouse 17x17, 3.2 w/o Uranium-235 optimized fuel assembly was modeled

using these codes.

All simulations were performed at beginning of life (BOL), hot full power (HFP), no xenon conditions, for at this time severe pin power peaking is most prominent. Simulations were performed for a variety of burnable absorber loadings and soluble boron concentrations. Table 8-1 contains a summary of the cases that were investigated.

Figures 8-1 through 8-10 contain 1/8 assembly pinwise power comparisons between PDQ07 and CASMO-2. Results from these comparisons indicate that PDQ07 conservatively overpredicts the maximum CASMO-2 pin power. This overprediction ranges from 0.86% to 2.26%. PDQ07 also correctly identifies the location of the CASMO-2 maximum pin power. Comparisons between PDQ07 and CASMO-2 maximum pin powers for each case are tabulated in Table 8-2.

The predictive capability of PDQ07 was assured by performing a statistical analysis over all pins in the problem and for pins with powers greater than or equal to 1.000. The average and average absolute differences and respective standard deviations are presented in Table 8-3 for all cases investigated.

8.3 Comparisons of PDQ07 to Cold Criticals

The ability of PDQ07 to predict pin powers at cold conditions was assessed by performing a series of simulations based on the B&W uranium criticals. In all simulations, PDQ07 conservatively and accurately predicted the maximum pin power. For additional specifics concerning the comparisons of PDQ07 to the B&W uranium criticals, refer to Reference 3.

8.4 BNL Benchmark Assembly Problem

The BNL evaluation of the Duke solution to the BNL benchmark assembly problem determined that PDQ07 methods overpredict the peak pin power by just over 1% at BOC and underpredict the peak pin power by approximately 1% at 40,000 MWD/MTU with the cross over occurring at approximately 15,000 MWD/MTU.

8.5 Conclusion

Comparisons between PDQ07 and CASMO-2 at HFP conditions indicate that PDQ07 conservatively predicts maximum pin powers. PDQ07 comparisons to B&W cold criticals also indicate that PDQ07 conservatively predicts maximum pin powers. However, the solution to the BNL benchmark problem shows an underprediction at high burnups. Therefore, an uncertainty of 2% is applied to the predicted pin peaking factors (see Appendix B).

TABLE 8-1

Characteristics of 1/8th Assembly Simulations

CASE	ENRICHMENT W/O U-235	BURNABLE ABSORBER LOADING	BORON CONCENTRATION (PPMB)
1	3.2	0	0
2	3.2	0	950
3	3.2	4	0
4	3.2	4	950
5	3.2	12	0
6	3.2	12	950
7	3.2	16	0
8	3.2	16	950
9	3.2	20	0
10	3.2	20	950

TABLE 8-2

Peak Pin Power Comparison

CASE	PDQ07 PEAK PIN POWER	CASMO PEAK PIN POWER	DIFFERENCE PDQ07-CASMO	% DIFFERENCE (P-C)/C
1	1.053	1.042	0.011	1.056
2	1.051	1.039	0.012	1.155
3	1.055	1.046	0.009	0.860
4	1.053	1.043	0.010	0.959
5	1.152	1.131	0.021	1.857
6	1.137	1.119	0.018	1.609
7	1.188	1.163	0.025	2.150
8	1.170	1.149	0.021	1.828
9	1.178	1.152	0.026	2.257
10	1.164	1.140	0.024	2.105

TABLE 8-3

Statistical Summary of Percent Differences between PDQ07 and CASMO-2 for Pins in Assemblies with Powers Greater Than or Equal to 1.000

CASE	$\overline{D}$ *	$\overline{ABS\ (D)}$	STANDARD DEVIATION (D)	S.D. [ABS (D)]
1	0.4450	0.5566	0.5524	0.4339
2	0.4657	0.5554	0.5627	0.4697
3	0.2151	0.3470	0.4098	0.3010
4	0.2109	0.3595	0.4215	0.2987
5	0.8916	1.0620	0.8705	0.6396
6	0.7936	0.9548	0.7733	0.5503
7	0.9321	1.1509	1.0832	0.8311
8	0.8057	1.0241	0.9635	0.7107
9	0.7130	0.8202	0.8885	0.7851
10	0.6458	0.7530	0.8109	0.7069

Statistical Summary of Percent Differences between PDQ07 and CASMO-2 For All Pins Within An Assembly

CASE	$\overline{D}$ *	$\overline{ABS\ (D)}$	STANDARD DEVIATION (D)	S.D. [ABS (D)]
1	0.0030	0.6395	0.7867	0.4463
2	0.0066	0.6572	0.8119	0.4648
3	-0.0255	0.4606	0.6328	0.4281
4	-0.0066	0.4520	0.6280	0.4298
5	0.0616	1.0682	1.2511	0.6310
6	0.0394	0.9801	1.1449	0.5713
7	0.0585	0.9416	1.2120	0.7499
8	0.0398	0.8436	1.0926	0.6819
9	0.0268	0.8776	1.1696	0.7604
10	0.0293	0.8059	1.0732	0.6972

* NOTE: D = [(PDQ07 - CASMO-2)/CASMO-2] *100

$$\overline{D} = \sum_{i=1}^{N} D_i/N$$

FIGURE 8-1

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 1

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	0	0
K-INFINITY	1.3486	1.3479
*MAX ROD POWER	1.053	1.042

0.0 0.0								
1.020 1.024	1.002 1.009							
1.021 1.024	1.002 1.009	1.002 1.010						
0.0 0.0	1.024 1.025	1.026 1.027	0.0 0.0					
1.020 1.023	1.002 1.009	1.004 1.013	1.033 1.037	1.023 1.045				
1.018 1.021	1.000 1.007	1.002 1.011	1.033 1.038	* 1.042 1.053	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	1.018 1.017	1.020 1.019	0.0 0.0	1.030 1.037	1.014 1.010	0.976 0.975		
1.008 1.006	0.990 0.993	0.990 0.994	1.011 1.006	0.986 0.989	0.970 0.964	0.959 0.947	0.955 0.940	
0.986 0.982	0.985 0.979	0.985 0.979	0.985 0.980	0.979 0.973	0.972 0.961	0.966 0.953	0.967 0.954	0.981 0.971

FIGURE 8-2

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 2

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	0	0
K-INFINITY	1.2122	1.2077
*MAX ROD POWER	1.051	1.039

0.0 0.0								
1.017 1.021	1.000 1.007							
1.018 1.021	1.000 1.007	1.000 1.008						
0.0 0.0	1.021 1.022	1.023 1.024	0.0 0.0					
1.018 1.021	1.000 1.007	1.002 1.011	1.030 1.035	1.020 1.042				
1.016 1.020	0.999 1.006	1.001 1.010	1.030 1.035	* 1.039 1.051	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	1.017 1.016	1.019 1.018	0.0 0.0	1.029 1.036	1.014 1.011	0.979 0.977		
1.007 1.006	0.990 0.993	0.990 0.994	1.011 1.007	0.988 0.991	0.973 0.966	0.963 0.950	0.960 0.945	
0.987 0.983	0.987 0.980	0.986 0.980	0.987 0.982	0.982 0.975	0.975 0.964	0.971 0.957	0.972 0.959	0.986 0.976

FIGURE 8-3

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 3

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	4	4
K-INFINITY	1.2978	1.2956
*MAX ROD POWER	1.055	1.046

0.0 0.0								
1.011 1.013	0.988 0.991							
1.006 1.003	0.975 0.972	0.933 0.915						
0.0 0.0	0.989 0.977	0.893 0.890	0.0 0.0					
1.011 1.009	0.980 0.979	0.940 0.925	0.905 0.908	0.964 0.963				
1.022 1.027	0.998 1.006	0.989 0.992	1.011 1.008	1.032 1.040	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	1.034 1.036	1.033 1.034	0.0 0.0	* 1.046 1.055	1.036 1.039	1.005 1.011		
1.033 1.037	1.014 1.024	1.015 1.023	1.037 1.036	1.014 1.022	1.000 1.000	0.992 0.987	0.990 0.983	
1.015 1.018	1.015 1.015	1.015 1.014	1.016 1.016	1.011 1.010	1.005 1.001	1.002 0.996	1.004 0.999	1.019 1.018

FIGURE 8-4

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 4

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	4	4
K-INFINITY	1.1757	1.1672
*MAX ROD POWER	1.053	1.043

0.0 0.0								
1.012 1.014	0.990 0.993							
1.007 1.004	0.977 0.975	0.937 0.919						
0.0 0.0	0.991 0.980	0.898 0.896	0.0 0.0					
1.011 1.010	0.982 0.981	0.943 0.928	0.909 0.912	0.965 0.965				
1.021 1.026	0.998 1.006	0.989 0.993	1.010 1.009	1.030 1.039	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	1.031 1.035	1.031 1.032	0.0 0.0	* 1.043 1.053	1.034 1.037	1.005 1.010		
1.031 1.035	1.013 1.022	1.013 1.021	1.035 1.034	1.013 1.020	1.000 0.999	0.992 0.987	0.991 0.983	
1.013 1.016	1.013 1.013	1.013 1.013	1.015 1.015	1.010 1.009	1.005 1.001	1.002 0.996	1.005 1.000	1.020 1.018

FIGURE 8-5

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 5

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	12	12
K-INFINITY	1.2073	1.2029
*MAX ROD POWER	1.152	1.131

0.0 0.0								
* 1.131 1.152	1.108 1.132							
1.123 1.144	1.099 1.123	1.088 1.109						
0.0 0.0	1.109 1.123	1.093 1.102	0.0 0.0					
1.093 1.109	1.063 1.081	1.036 1.045	1.019 1.014	0.946 0.937				
1.076 1.085	1.038 1.047	0.980 0.970	0.905 0.906	0.864 0.855	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	1.043 1.041	0.930 0.934	0.0 0.0	0.860 0.849	0.876 0.867	0.932 0.908		
1.055 1.062	1.020 1.028	0.967 0.956	0.907 0.906	0.930 0.910	0.950 0.934	0.973 0.959	0.996 0.989	
1.035 1.046	0.026 1.032	1.004 1.000	0.984 0.976	0.983 0.973	0.992 0.984	1.007 1.000	1.024 1.023	1.048 1.054

FIGURE 8-6

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 6

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	12	12
K-INFINITY	1.1010	1.0941
*MAX ROD POWER	1.137	1.119

0.0 0.0								
* 1.119 1.137	1.097 1.118							
1.112 1.130	1.090 1.110	1.080 1.099						
0.0 0.0	1.100 1.112	1.086 1.094	0.0 0.0					
1.086 1.100	1.058 1.074	1.032 1.041	1.018 1.013	0.949 0.941				
1.070 1.079	1.035 1.044	0.981 0.970	0.909 0.911	0.871 0.864	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	1.041 1.039	0.933 0.937	0.0 0.0	0.868 0.858	0.884 0.876	0.938 0.915		
1.052 1.059	1.019 1.027	0.969 0.958	0.912 0.912	0.936 0.917	0.956 0.940	0.977 0.964	0.999 0.992	
1.034 1.044	1.026 1.030	1.005 1.001	0.987 0.980	0.987 0.978	0.996 0.988	1.010 1.004	1.025 1.024	1.049 1.054

FIGURE 8-7

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 7

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	16	16
K-INFINITY	1.1629	1.1576
*MAX ROD POWER	1.188	1.163

0.0 0.0								
* 1.163 1.188	1.138 1.166							
1.150 1.173	1.125 1.152	1.111 1.137						
0.0 0.0	1.123 1.138	1.110 1.124	0.0 0.0					
1.072 1.080	1.051 1.065	1.040 1.052	1.035 1.034	0.966 0.962				
0.950 0.952	0.975 0.962	0.966 0.953	0.915 0.918	0.885 0.879	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	0.909 0.906	0.904 0.902	0.0 0.0	0.882 0.874	0.905 0.900	0.966 0.946		
0.922 0.919	0.949 0.934	0.948 0.933	0.916 0.918	0.955 0.937	0.983 0.971	1.010 1.001	1.036 1.035	
0.992 0.989	0.993 0.993	0.995 0.993	0.996 0.993	1.009 1.004	1.027 1.023	1.046 1.046	1.066 1.071	1.093 1.105

FIGURE 8-8

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 8

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	16	16
K-INFINITY	1.0655	1.0581
*MAX ROD POWER	1.170	1.149

0.0 0.0								
* 1.149 1.170	1.126 1.150							
1.138 1.157	1.114 1.137	1.102 1.125						
0.0 0.0	1.113 1.127	1.102 1.115	0.0 0.0					
1.067 1.074	1.047 1.059	1.037 1.048	1.033 1.033	0.968 0.965				
0.952 0.953	0.976 0.963	0.968 0.955	0.919 0.923	0.890 0.886	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	0.913 0.911	0.909 0.907	0.0 0.0	0.888 0.881	0.910 0.906	0.969 0.950		
0.927 0.925	0.953 0.939	0.953 0.938	0.921 0.923	0.959 0.943	0.986 0.974	1.011 1.002	1.035 1.033	
0.995 0.993	0.996 0.996	0.997 0.997	0.999 0.996	1.011 1.006	1.027 1.024	1.045 1.044	1.064 1.068	1.088 1.099

FIGURE 8-9

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 9

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	20	20
K-INFINITY	1.1206	1.1148
*MAX ROD POWER	1.178	1.152

0.0 0.0								
* 1.152 1.178	1.121 1.148							
1.132 1.151	1.094 1.112	1.036 1.034						
0.0 0.0	1.084 1.087	0.967 0.977	0.0 0.0					
1.061 1.064	1.026 1.033	0.971 0.958	0.899 0.898	0.903 0.877				
0.952 0.956	0.972 0.959	0.951 0.931	0.891 0.887	0.873 0.865	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	0.923 0.924	0.916 0.915	0.0 0.0	0.898 0.891	0.929 0.928	0.998 0.984		
0.948 0.950	0.976 0.964	0.975 0.962	0.943 0.948	0.985 0.971	1.017 1.010	1.048 1.046	1.077 1.084	
1.025 1.027	1.027 1.030	1.029 1.031	1.031 1.032	1.045 1.046	1.065 1.069	1.087 1.095	1.109 1.124	1.136 1.160

FIGURE 8-10

CASMO-2 AND PDQ-7
ROD POWER COMPARISON
BOL HFP NO XENON
3.2 w/o U-235 OPT 17x17 FA
CASE NUMBER 10

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	20	20
K-INFINITY	1.0315	1.0238
*MAX ROD POWER	1.164	1.140

0.0 0.0								
* 1.140 1.164	1.112 1.136							
1.123 1.140	1.087 1.103	1.032 1.029						
0.0 0.0	1.078 1.082	0.967 0.977	0.0 0.0					
1.058 1.062	1.025 1.031	0.973 0.960	0.905 0.904	0.909 0.885				
0.954 0.958	0.974 0.962	0.954 0.936	0.897 0.894	0.880 0.873	0.0 0.0	CASMO-2 PDQ-7		
0.0 0.0	0.927 0.928	0.921 0.920	0.0 0.0	0.903 0.898	0.932 0.932	0.998 0.984		
0.951 0.953	0.978 0.967	0.978 0.966	0.946 0.951	0.986 0.973	1.016 1.010	1.045 1.043	1.072 1.077	
1.025 1.028	1.027 1.031	1.029 1.031	1.031 1.031	1.044 1.044	1.062 1.065	1.082 1.089	1.102 1.115	1.127 1.149

9. DEVELOPMENT OF CORE PHYSICS PARAMETERS

Upon completion of the Final Fuel Cycle Design, physics parameters such as boron concentrations and worths, power distributions, etc. are calculated primarily for HFP and some HZP conditions. The purpose of this stage of developing core physics parameters is to provide additional calculations to supplement those already performed. The results of these calculations are used for startup test predictions and core physics parameters throughout the cycle.

9.1 Startup Test Predictions

After each refueling, the reactor undergoes a startup test program aimed at verifying that the reactor core is correctly loaded and to verify reactor behavior is as predicted by the nuclear simulators which were used in generating the data used in the plant's safety analysis.

9.1.1 Critical Boron Concentrations and Boron Worths

Critical boron concentrations and boron worths at a variety of rod configurations, at HZP and HFP, as a function boron concentration, at different xenon concentrations, and at different times in the fuel cycle are calculated. EPRI-NODE-P and SIMULATE-3 are capable of performing critical boron search calculations. The method used for PDQ07 is to correct the input boron concentration to the critical boron concentration using a calculated boron worth and the calculated reactivity.

Table 9-1 shows some of the critical boron calculations normally performed for startup physics tests. These calculations are performed after the sequential insertion of each control or shutdown bank and are sometimes referred to as boron endpoints.

Critical boron concentrations at HZP and HFP with all rods out are also calculated as a function of cycle burnup. An example of how boron changes

with burnup is shown in Figure 9-1. These curves are referred to as boron letdown curves.

The boron worths are usually calculated by running two identical cases except that the soluble boron concentration is different. The differential boron worth is calculated by subtracting the reactivities and dividing by the boron difference. Differential boron worths are usually quoted in PCM/PPMB. The inverse boron worth is the inverse of the differential boron worth and is usually quoted in PPMB/%Δρ.

Table 9-2 shows the soluble boron worths usually performed for startup physics tests. Similar to critical boron concentrations, these worths are calculated with sequential bank insertions.

Differential boron worth (or inverse boron worth) can also be calculated as a function of boron concentration and as a function of cycle burnup. Figures 9-2 and 9-3 show the results of a typical differential boron worth calculation vs. boron concentration and vs. burnup, respectively.

9.1.2 Xenon Worth and Defect

Xenon worth is calculated as a function of cycle burnup. The nominal HFP depletion cases with equilibrium xenon are used as input to a second set of cases where the xenon concentration is set to zero (or the xenon cross sections are set to zero). The difference in reactivities between the equilibrium xenon and no xenon cases equals the equilibrium xenon worth at HFP. Figure 9-4 shows the results of a typical equilibrium xenon worth calculation.

Xenon worth can also be presented as a function of power level. Worths presented in this manner are usually referred to as the equilibrium xenon reactivity defect and are quoted in either pcm or %Δρ. Figure 9-5 shows the results of a typical xenon defect calculation.

9.1.3 Rod Worths

9.1.3.1 Bank Worths

The worth of the shutdown and control banks are calculated at BOC HZP for use in the zero power physics testing. The rod banks are sequentially inserted or withdrawn assuming no control rod overlap. The bank worth is the difference in reactivity between the fully inserted case and the fully withdrawn case.

Integral rod worth curves are calculated at BOC HZP for Control Banks B, C and D. The rod banks are inserted both sequentially and with 50% overlap. Figure 9-6 shows the results of a typical integral rod worth calculation.

Control bank worths with sequential insertion and integral rod worth curves with 50% overlap are calculated at HFP equilibrium xenon both at BOC and EOC.

9.1.3.2 Stuck Rod Worth

The maximum worth of a single control rod stuck out of the reactor core at HZP is calculated. The worth of the stuck rod is used by the site engineers in the reactivity balance procedures to guarantee shutdown margin. If the stuck rod worth is to be measured during the startup test program, then a recalculation of the worth is performed simulating the test conditions. This worth would then be provided as a startup test prediction.

9.1.3.3 Dropped Rod Worth

The maximum worth of a single control rod dropped into the reactor core is calculated. If this parameter is to be measured during the startup test program, then a recalculation of the worth is performed simulating the test conditions. This worth would then be provided as a startup test prediction.

9.1.3.4 Ejected Rod Worth

The maximum ejected control rod worth is calculated. If this parameter is to be measured during the startup test program, then a recalculation of the worth is performed simulating the test conditions. This worth would then be provided as a startup test prediction.

9.1.4 Reactivity Coefficients

9.1.4.1 HZP Coefficients

At HZP the isothermal temperature coefficient is measured by varying the average moderator temperature and determining the corresponding reactivity change. The calculations for predicting the isothermal temperature coefficient should be performed by changing the average moderator temperature in the core model. The resulting reactivity change is then divided by the temperature change to yield the HZP isothermal temperature coefficient.

The Doppler or fuel temperature coefficient at HZP can be calculated by varying the fuel temperature while maintaining the no load moderator temperature. The resulting reactivity change divided by the change in fuel temperature is the Doppler coefficient at HZP.

The predicted moderator coefficient is calculated by subtracting the Doppler coefficient from the isothermal coefficient. It is compared to the (inferred) measured moderator coefficient obtained by subtracting the predicted Doppler coefficient from the measured isothermal coefficient.

Alternately, the moderator temperature coefficient can also be explicitly calculated.

9.1.4.2 HFP Coefficients

Both a temperature coefficient of reactivity and a power Doppler coefficient of reactivity are calculated at HFP. The temperature coefficient is calculated by running one equilibrium HFP case at BOC (4 EFPD) and a second case where the moderator temperature is changed. The difference in reactivity divided by the temperature change is the temperature coefficient. To calculate the power Doppler coefficient, a third case is performed where the power level is reduced. All other parameters are kept at the HFP equilibrium values. The difference in reactivity between the HFP and the reduced power cases divided by change in power is the power Doppler coefficient.

9.1.5 Power Distribution

Power distributions, both assembly radial and total peaking factors, are measured at various power levels as identified in the test procedures for McGuire/Catawba reload startups. Calculations are performed at these power levels and nominal conditions to provide predicted power distributions for comparison.

9.1.6 Kinetics Parameters

Kinetics parameters are calculated using the methodology and codes as discussed in Section 4.2.4.8. These parameters include the six group βi effective and λi, total β effective and ℓ^*. These kinetics parameters are generated for both BOC HZP and BOC HFP conditions with ARO. A second set of delayed neutron data may be generated at EOC.

9.2 Startup and Operation Report

The purpose of the Startup and Operation Report is to document the predicted behavior of the reactor core as a function of burnup and power level. It is intended to be used for operator guidance and to aid the site engineer.

This report will include sufficient information to calculate reactivity balance throughout the cycle. Table 9-3 lists items typical of what will be calculated for this report.

TABLE 9-1

Critical Boron Concentrations (ppmB)

HZP, NOXE, 0 EFPD

ARO

Bank D in

Banks D + C in

Banks D + C + B in

Banks D + C + B + A in

Banks D + C + B + A + S_E in

Banks D + C + B + A + S_E + S_D in

Banks D + C + B + A + S_E + S_D + S_C in

Banks D + C + B + A + S_E + S_D + S_C + S_B in

Banks D + C + B + A + S_E + S_D + S_C + S_B + S_A in

HFP, NOXE, 0 EFPD

ARO

HFP, EQXE, 4 EFPD

ARO

Bank D in

HFP, EQXE, EOC

ARO

TABLE 9-2

Boron Worth (pcm/ppmB)

HZP, NOXE, 0 EFPD

ARO

Bank D in*

Banks D + C in

Banks D + C + B in

Banks D + C + B + A in

ARI

HFP, EQXE, 4 EFPD

ARO

HFP, EQXE, EOC

ARO

* Note: When bank worths are determined using interchange (swap) with the reference control bank, the boron worth with the reference bank only inserted is evaluated in place of sequential insertions.

TABLE 9-3

Typical Core Physics Data

A. Critical Boron Concentrations
   1. ARO HFP Versus Burnup
   2. ARO HZP Versus Burnup

B. Shutdown Boron Concentrations Required for Shutdown with Highest Worth Rod Stuck Out (NoXe)
   1. HZP Versus Burnup
   2. 500°F, 200°F and 68°F Versus Burnup

C. Differential Boron Worth HFP, HZP Versus Burnup

D. Power Distributions from the Cycle Depletion

E. Rod Worths BOC, EOC, HFP and HZP

F. Xenon Worth Versus Power Level

G. Xenon Worth Versus Burnup

H. Reactivity Coefficients Versus Temperature, Power Level and Burnup

FIGURE 9-1

Boron Letdown Curve
HFP, ARO

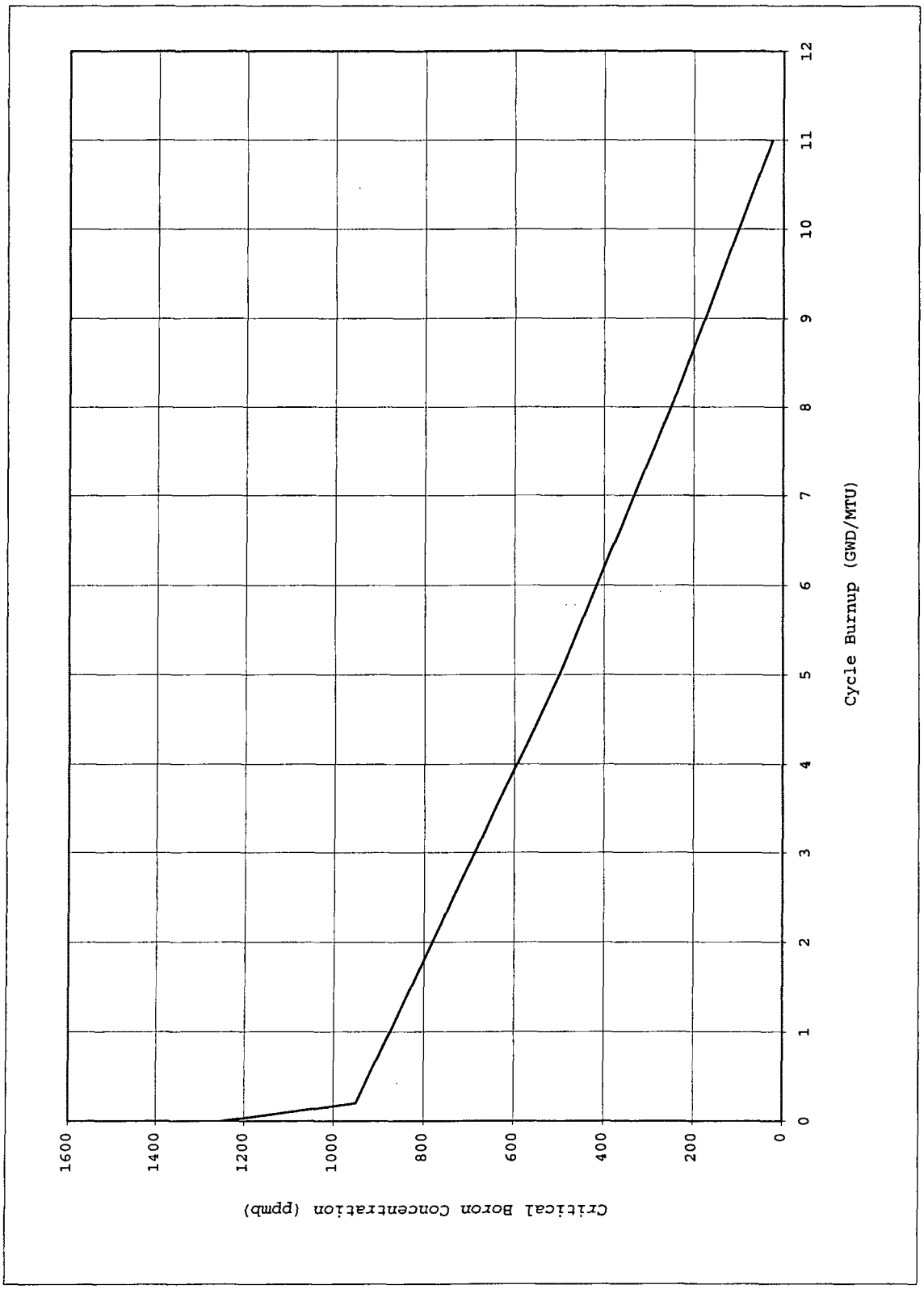


FIGURE 9-2

Differential Boron Worth
HZP, BOL

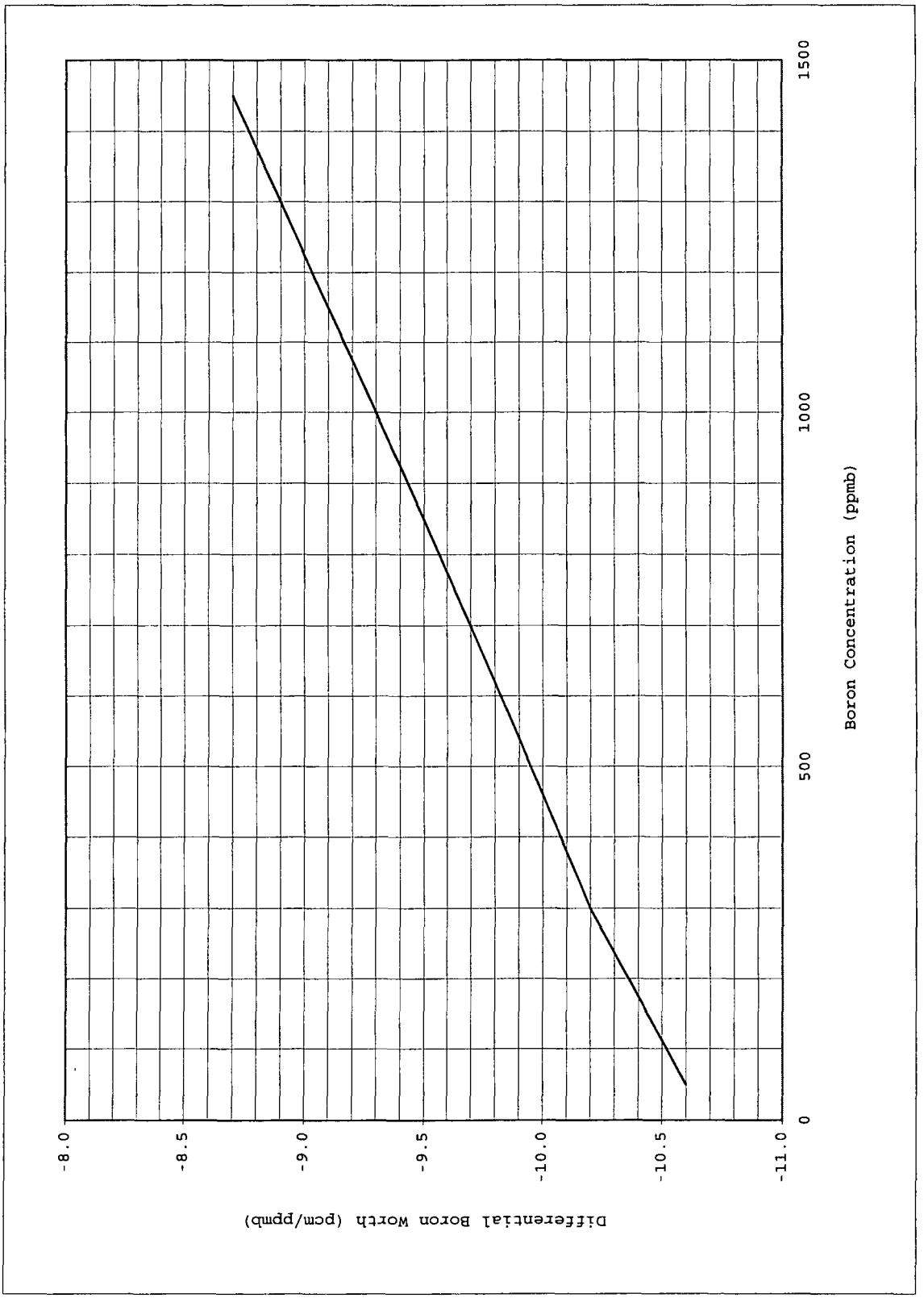


FIGURE 9-3

Inverse Boron Worth
HFP, ARO

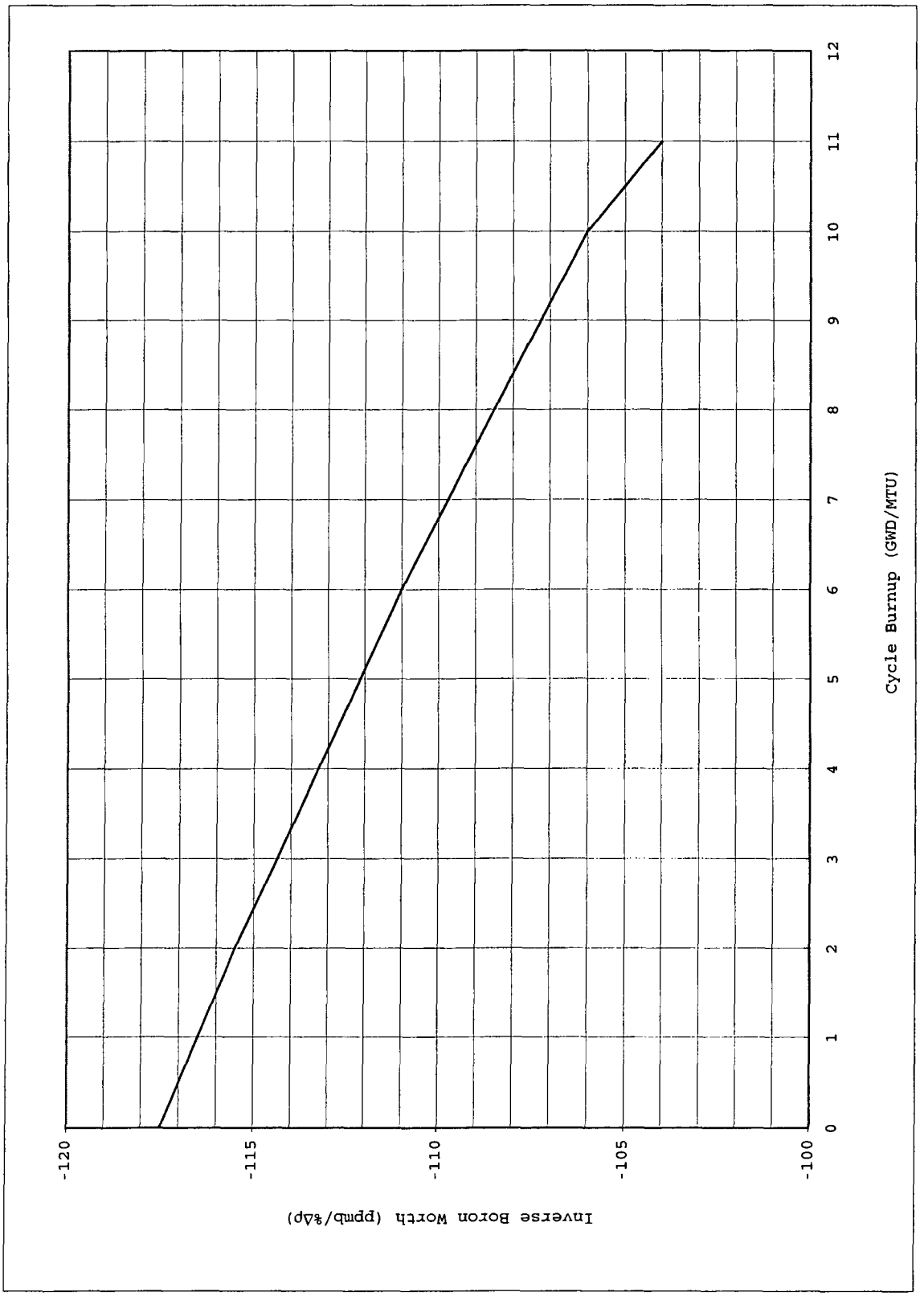


FIGURE 9-4

Equilibrium Xenon Worth
HFP

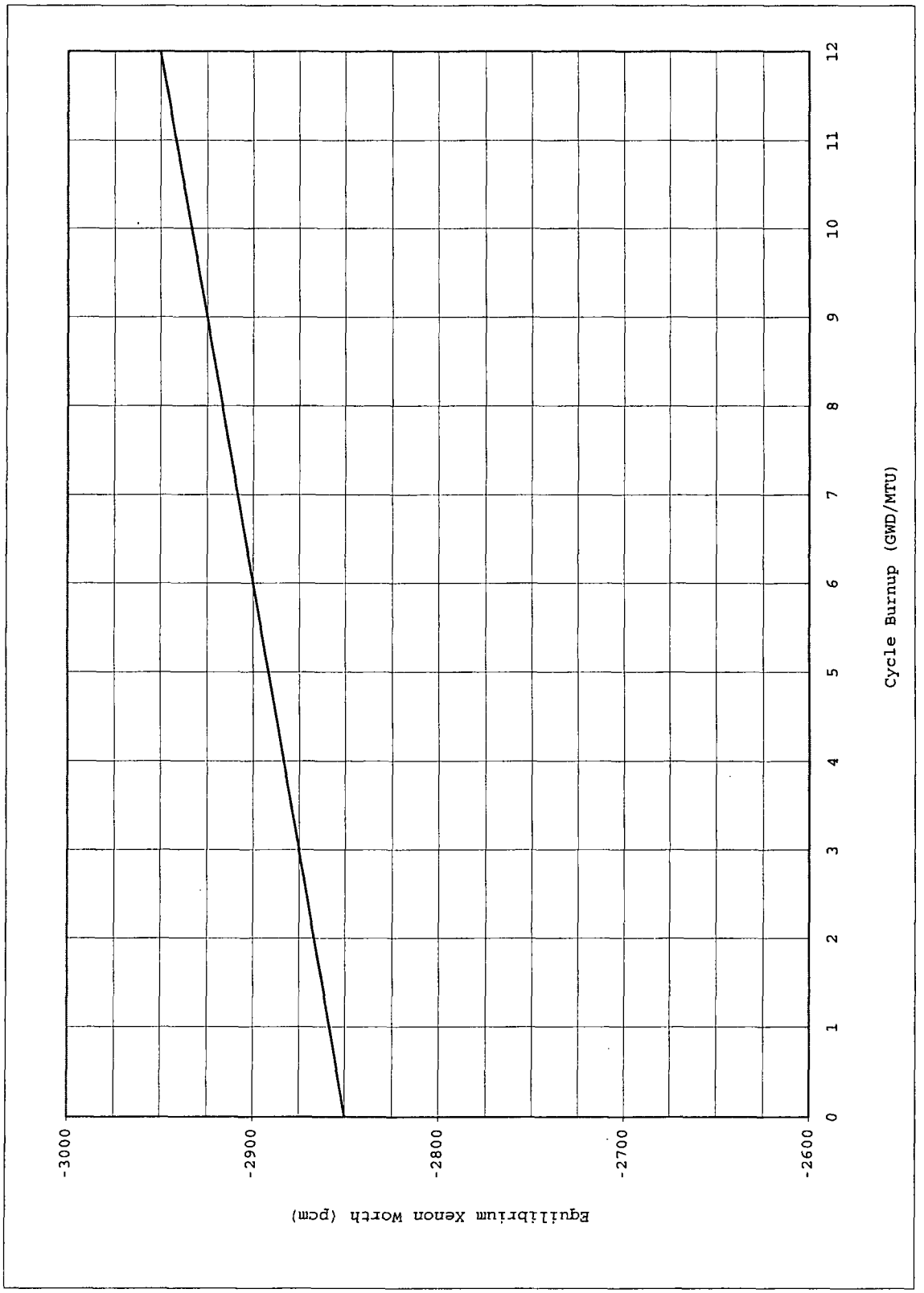


FIGURE 9-5

Xenon Reactivity Defect
BOL

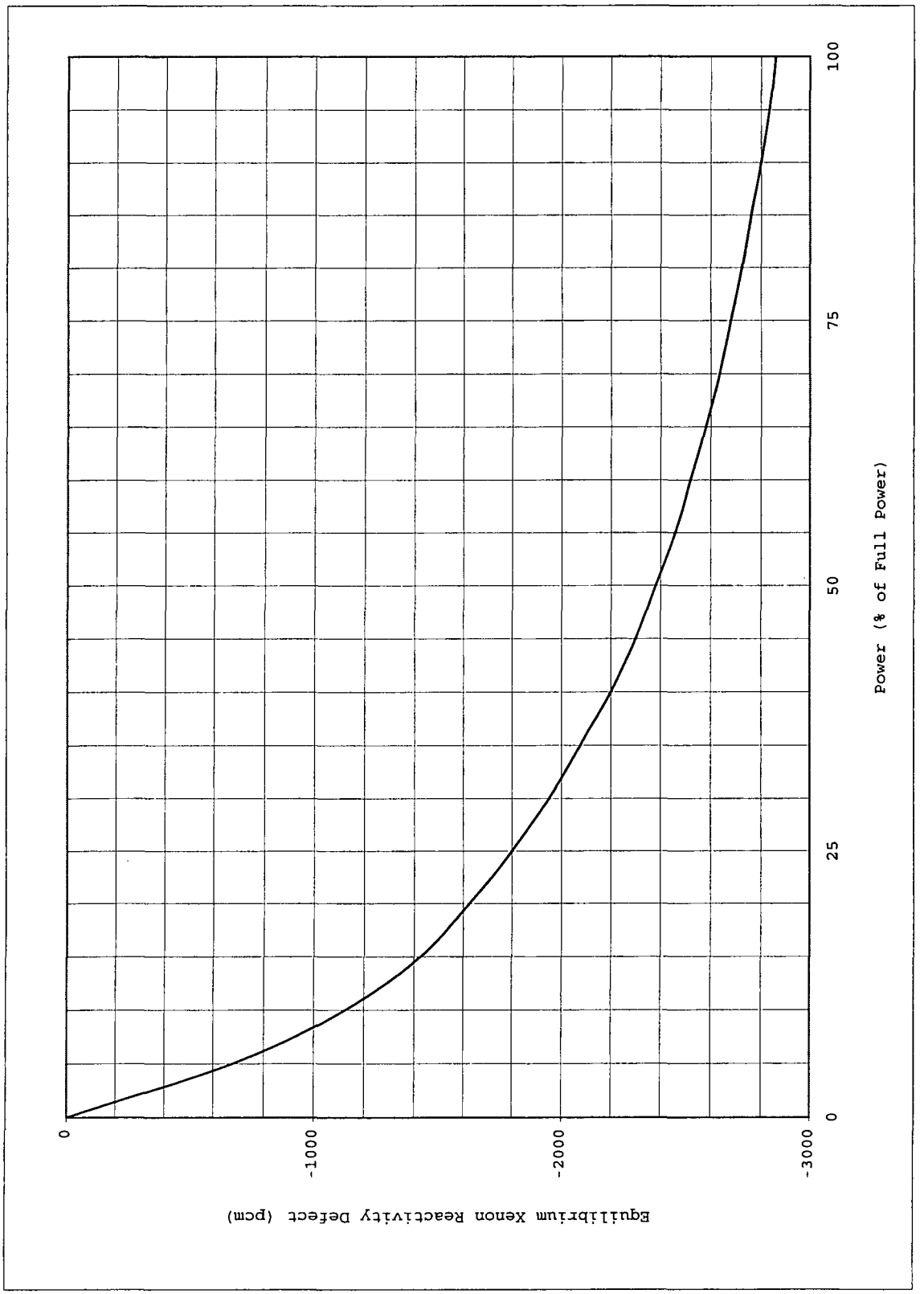
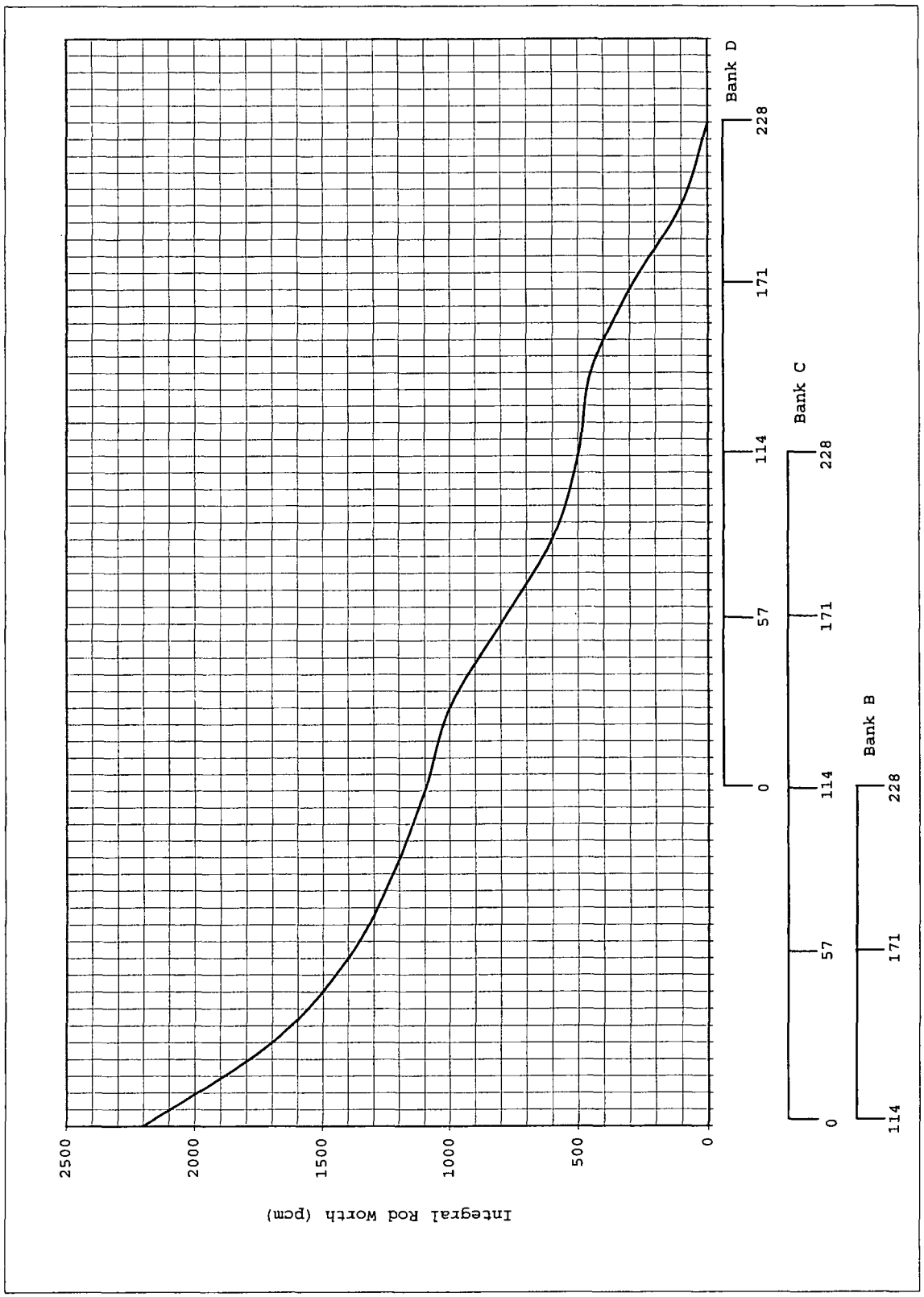


FIGURE 9-6

Integral Rod Worth
HZP, BOL



10. PHYSICS TEST COMPARISONS

Startup physics testing is described in UFSAR Chapter 14. The startup testing information contained in this section is presented to provide the perspective on the benchmarking of measured to predicted results. The predicted results presented are based on the PDQ07/EPRI-NODE-P models. Reference 28 presents the benchmarking results for the CASMO-3/SIMULATE-3 models.

10.1 Introduction

This section presents measurement and calculational techniques and comparisons of calculated and measured results for some key core physics parameters. The physics parameters include hot zero power (HZP) and hot full power (HFP) critical boron concentrations, HZP control rod worths and ejected rod worths, and HZP isothermal temperature coefficients.

The measured data is from the McGuire Nuclear Station, Unit 1 Cycles 1 and 1A, and Unit 2, Cycle 1. (Broken hold down springs on some Burnable Poison rods were found during an outage on McGuire Unit 1 at 191.5 EFPD. During this outage, 94 of 96 Burnable Poison Rod Assemblies were removed from the core. Cycle 1A is the continuation of Cycle 1 but without the Burnable Poison Rods.) The measurement techniques discussed are those currently used at the station. The HZP measurements were taken at beginning-of-cycle (BOC) during the Zero Power Physics Testing. The HFP boron concentration measurements were taken at various time steps throughout the cycles. All calculations were performed with EPRI-NODE-P.

The comparisons of calculated and measured results present the means of the differences between the measured and calculated data and the corresponding standard deviations. The mean and standard deviation are defined as follows:

$$\text{Mean} = \bar{x} = \frac{\Sigma x_i}{n}$$

$$\text{Standard Deviation} = S = \sqrt{\frac{\Sigma(\bar{x} - x_i)^2}{n-1}}$$

where: x_i = value for the ith observation

n = number of observations.

10.2 Critical Boron Concentrations

10.2.1 Measurement Technique

Critical boron concentrations are measured at HZP and HFP by an acid-base titration of a reactor coolant system sample.

The measurement uncertainty for critical boron concentrations is due to (1) error in the titration method and (2) error due to differences between the sample concentration and the core average concentration. Based on conservative estimates of these errors, the total uncertainty associated with the critical boron concentration measurements is less than 20 ppmb.

10.2.2 Calculational Technique

Critical boron concentrations are calculated at HZP and HFP using EPRI-NODE-P in the boron search mode. Since the search does not yield an exactly critical value, fixed boron runs using EPRI-NODE-P are also made to calculate a boron worth, which is then used to correct the calculated boron concentration to exactly critical.

10.2.3 Comparison of Calculated and Measured Results

10.2.3.1 Hot Zero Power Comparison

The calculated and measured critical boron concentrations at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and Unit 2, Cycle 1 are compared in Table 10-1. Each entry corresponds to a different control rod position. The mean

of the differences for these three cycles was found to be -7 ppmb with a standard deviation of 16 ppmb.

10.2.3.2 Hot Full Power Comparison

The calculated and measured critical boron concentrations at HFP for McGuire Unit 1, Cycles 1 and 1A, are compared in Table 10-2. The mean of the differences for these cycles is -41 ppmb with a standard deviation of 11 ppmb.

The data displayed in Table 10-2 can be visualized better by examining plots of soluble boron concentration as a function of burnup. These boron letdown curves are shown in Figures 10-1 and 10-2.

10.2.4 Summary

The comparison between EPRI-NODE-P and measured critical boron concentrations at HZP and HFP indicate EPRI-NODE-P can adequately predict soluble boron concentrations.

10.3 Control Rod Worth

10.3.1 Measurement Techniques

Individual control rod bank worths are measured by the boron swap technique. This technique involves a continuous decrease in boron concentration together with an insertion of the control rods in small, discrete steps. The change in reactivity due to each insertion is determined from reactivity computer readings before and after the insertion. The worth of each rod bank is the sum of all the reactivity changes for that bank. Measured bank worths in ppmb can be determined independent of the reactivity computer by using the measured boron endpoints.

10.3.2 Calculational Techniques

Individual and total controlling rod bank worths in terms of reactivity are

calculated by making two EPRI-NODE-P runs. The first is a boron search run with the rod bank(s) out. The boron concentration found in this run is then used in a fixed boron run with the rod bank(s) in. The difference in reactivity between these two runs with constant boron concentration is the rod bank(s) worth.

Bank worths were also calculated using the calculated Boron endpoints. These bank worths are in terms of ppmB.

10.3.3 Comparison of Calculated and Measured Results

A comparison of calculated and measured control rod worths in terms of reactivity is shown in Table 10-3. This table compares the worths of control banks: D, C, B, and A and shutdown banks: E, D, and C at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and McGuire Unit 2, Cycle 1. A comparison of calculated and measured control rod worths in terms of ppmB is shown in Table 10-4. This table also compares the worths of control banks: D, C, B, and A and shutdown banks: E, D, and C at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and McGuire Unit 2 Cycle 1. Table 10-5 is a comparison of PDQ07 calculated and measured control rod worths.

PDQ07 calculated bank worths agree well to measured with an average difference of 2.7% and a standard deviation of 3.3%. EPRI-NODE-P calculated bank worths similarly agreed well with an average difference of -4.5% and a standard deviation of 5.1%. Rod worths calculated using boron endpoints also agreed well, with an average difference of -2.2% and a standard deviation of 7.9%.

10.3.4 Summary

The comparisons between the calculated and measured control rod worths at HZP indicate that EPRI-NODE-P can adequately predict control rod worths. Tables 10-3 and 10-4 indicate consistent agreement using either reactivity or boron endpoint measurement techniques.

10.4 Ejected Rod Worths

Ejected rod worth is defined here as the measured worth of the worst case ejected rod. No error adjustments have been included.

10.4.1 Measurement Technique

Ejected rod worths are measured by boron swap. The boron swap method is similar to the method used to measure control rod worth. It involves maintaining criticality by varying the boron concentration to compensate for the ejection of the worst case rod. The control rod positions are held constant. As was done for control rod worth, the ejected rod worth is determined from the reactivity computer.

10.4.2 Calculational Techniques

Ejected rod worths are calculated using EPRI-NODE-P to simulate boron swap. A boron search run is first performed to determine the critical boron concentration at the rod group position. The boron concentration as calculated in the EPRI-NODE-P run should be corrected for exact criticality. Using this corrected boron concentration and a constant rod group position, the reactivity is determined with the worst case rod first in and then out. The ejected rod worth is the difference in reactivity between the worst case rod in and out.

10.4.3 Comparison of Calculated and Measured Results

A comparison of calculated and measured ejected rod worth for McGuire Unit 1, Cycle 1, is given in Table 10-6.

10.5 Isothermal Temperature Coefficients

The isothermal temperature coefficient is defined as the change in reactivity per unit change in moderator temperature at hot zero power, i.e.,

$$\alpha_T = \frac{\Delta\rho}{\Delta T}$$

10.5.1 Measurement Techniques

The isothermal temperature coefficient is measured by executing an average moderator temperature ramp to +5°F and then a ramp down to the initial equilibrium critical conditions. During each change, reactivity is measured on the reactivity computer and other pertinent data is measured. After each change, steady state conditions are established. The isothermal temperature coefficient is determined as the change in reactivity between plateaus divided by the change in temperature. Since two different temperature ramps are executed, two coefficients can be determined. The reported isothermal temperature coefficient is an average of these two coefficients.

10.5.2 Calculational Technique

The isothermal temperature coefficient at HZP is calculated using EPRI-NODE-P. Two cases with the same boron concentration and rod positions but different moderator temperatures are run. The isothermal temperature coefficient is the difference in reactivity between the two cases divided by the difference in the moderator temperatures.

10.5.3 Comparison of Calculated and Measured Results

A comparison of calculated and measured isothermal temperature coefficients at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and Unit 2, Cycle 1 is presented in Table 10-7. The mean of all the differences was found to be 1.38 pcm/°F with a standard deviation of 1.87 pcm/°F.

10.5.4 Summary

The comparison between calculated and measured isothermal temperature coefficients indicates that EPRI-NODE-P is a good predictor of isothermal temperature coefficients.

TABLE 10-1

McGuire

Critical Boron Concentrations at Hot Zero Power, BOC

Unit	Cycle	Critical Boron Conc. PPM Calculated	Measured	Difference
1	1	1301	1310	-9
		1242	1248	-6
		1123	1128	-5
		1033	1029	4
		972	967	5
		888	891	-3
		822	819	3
		728	723	5
1	1A	1269	1310	-41
		1200	1242	-42
		1090	1125	-35
2	1	1280	1295	-15
		1221	1217	4
		1101	1097	4
		1002	997	5
		944	938	6
		861	860	1
		788	791	-3
		691	694	-3
Mean		----	----	-6.6
Standard Deviation		----	----	15.7

Difference = Calculated - Measured

TABLE 10-2

McGuire 1 Cycle 1-1A

Hot Full Power Critical Boron Concentrations

Unit	EFPD	Critical Boron Conc. PPM Calculated	Measured	Difference
1	24.6	860	880	-20
	34.4	846	865	-19
	39.2	838	864	-26
	49.2	823	862	-39
	82.2	761	801	-40
	90.4	745	790	-45
	99.0	729	771	-42
	101.2	724	762	-38
	126.0	667	724	-57
	154.4	600	650	-50
	180.7	531	591	-60
1A	203.7	782	831	-49
	217.5	713	751	-38
	227.8	673	719	-46
	232.9	653	696	-43
	238.5	631	677	-46
	255.2	566	615	-49
	279.8	473.	511	-38
	300.6	395	434	-39
	330.8	281	318	-37
Mean		----	----	-41.1
Standard Deviation		----	----	10.5

Difference = Calculated - Measured

TABLE 10-3

McGuire

Control Rod Worths at Hot Zero Power, BOC

Unit/Cycle	Bank	Rod Worth (PCM) Calculated	Measured	Difference (PCM)	Difference (%)
1/1	CD	606	669	-63	-9.4
	CC	1217	1250	-33	-2.6
	CB	925	996	-71	-7.1
	CA	654	695	-41	-5.9
	SE	884	840	44	5.2
	SD	668	755	-87	-11.5
	SC	961	1011	-50	-4.9
1/1A	CD	685	712	-27	-3.8
	CC	1100	1038	62	6.0
2/1	CD	604	664	-60	-9.0
	CC	1224	1283	-59	-4.6
	CB	1004	1105	-101	-9.1
	CA	618	678	-60	-8.8
	SE	862	853	9	1.1
	SD	738	771	-33	-4.3
	SC	992	1026	-31	-3.0
Mean		----	----	-37.6	-4.5
Standard Deviation		----	----	43.8	5.1

Difference (PCM) = Calculated - Measured

$$\text{Difference}(\%) = \frac{\text{Calculated} - \text{Measured}}{\text{Measured}} \times 100$$

TABLE 10-4

McGuire
Control Rod Worths at Hot Zero Power, BOC
Using Boron Endpoints

Unit/Cycle	Bank	Rod Worth (PPM) Calculated	Measured	Difference (PPM)	Difference (%)
1/1	CD	59	62	-3	-4.8
	CC	119	120	-1	-0.8
	CB	90	99	-9	-9.1
	CA	61	62	-1	-1.6
	SE	84	76	8	10.5
	SD	66	72	-6	-8.3
	SC	94	96	-2	-2.1
1/1A	CD	69	68	1	1.5
	CC	110	117	-7	-6.0
2/1	CD	59	78	-19	-24.4
	CC	120	120	0	0.0
	CB	99	100	-1	-1.0
	CA	58	59	-1	-1.7
	SE	83	78	5	6.4
	SD	73	69	4	5.8
	SC	97	97	0	0.0
Mean		----	----	-2.0	-2.2
Standard Deviation		----	----	6.3	7.9

Difference (PCM) = Calculated - Measured

$$\text{Difference}(\%) = \frac{\text{Calculated} - \text{Measured}}{\text{Measured}} \times 100$$

TABLE 10-5

McGuire

PDQ07 Calculated Rod Worths vs. Measured Rod Worths at HZP, BOC

Unit/Cycle	Bank	Rod Worth (PCM) Calculated	Measured	Difference (PCM)	Difference (%)
1/1	D	644	669	-25	-3.7
	C	1214	1250	-36	-2.9
	B	962	996	-34	-3.4
1/1A	D	667	712	-45	-6.3
	C	1088	1038	50	4.8
2/1	D	637	664	-27	-4.1
	C	1261	1283	-22	-1.7
	B	1090	1105	-15	-1.4
	A	638	678	-40	-5.9
Mean		----	----	-22	-2.7
Standard Deviation		----	----	28	3.3

Difference (PCM) = Calculated - Measured

$$\text{Difference}(\%) = \frac{\text{Calculated} - \text{Measured}}{\text{Measured}} \times 100$$

TABLE 10-6

McGuire 1 Cycle 1
Ejected Rod Worths

Cycle	Location	Worth (PCM) Calculated	Measured	Difference (PCM)
1	D-12	406	432	-26

Difference (PCM) = Calculated - Measured

TABLE 10-7

McGuire

Isothermal Temperature Coefficients at Hot Zero Power, BOC

Unit/Cycle	Control Rod Configuration	Temp. Coeff., (PCM/°F) Calculated	Measured	Difference (PCM/°F)
1	ARO	-1.03	-0.57	-0.46
	D in	-2.09	-2.02	-0.07
	C & D in	-6.03	-5.86	-0.17
	B, C, & D in	-6.08	-6.83	0.75
	A, B, C, & D in	-9.37	-9.72	0.35
1	ARO	-4.51	-1.13	-3.38
	D in	-5.86	-1.98	-3.88
	C & D in	-9.76	-4.83	-4.93
2	ARO	-2.34	-1.41	-0.93
	D in	-3.54	-2.73	-0.81
	C & D in	-7.70	-6.07	-1.63
Mean		----	----	-1.38
Standard Deviation		----	----	1.87

Difference (PCM/°F) = Calculated - Measured

FIGURE 10-1

McGuire 1 Cycle 1
Boron Letdown Curves

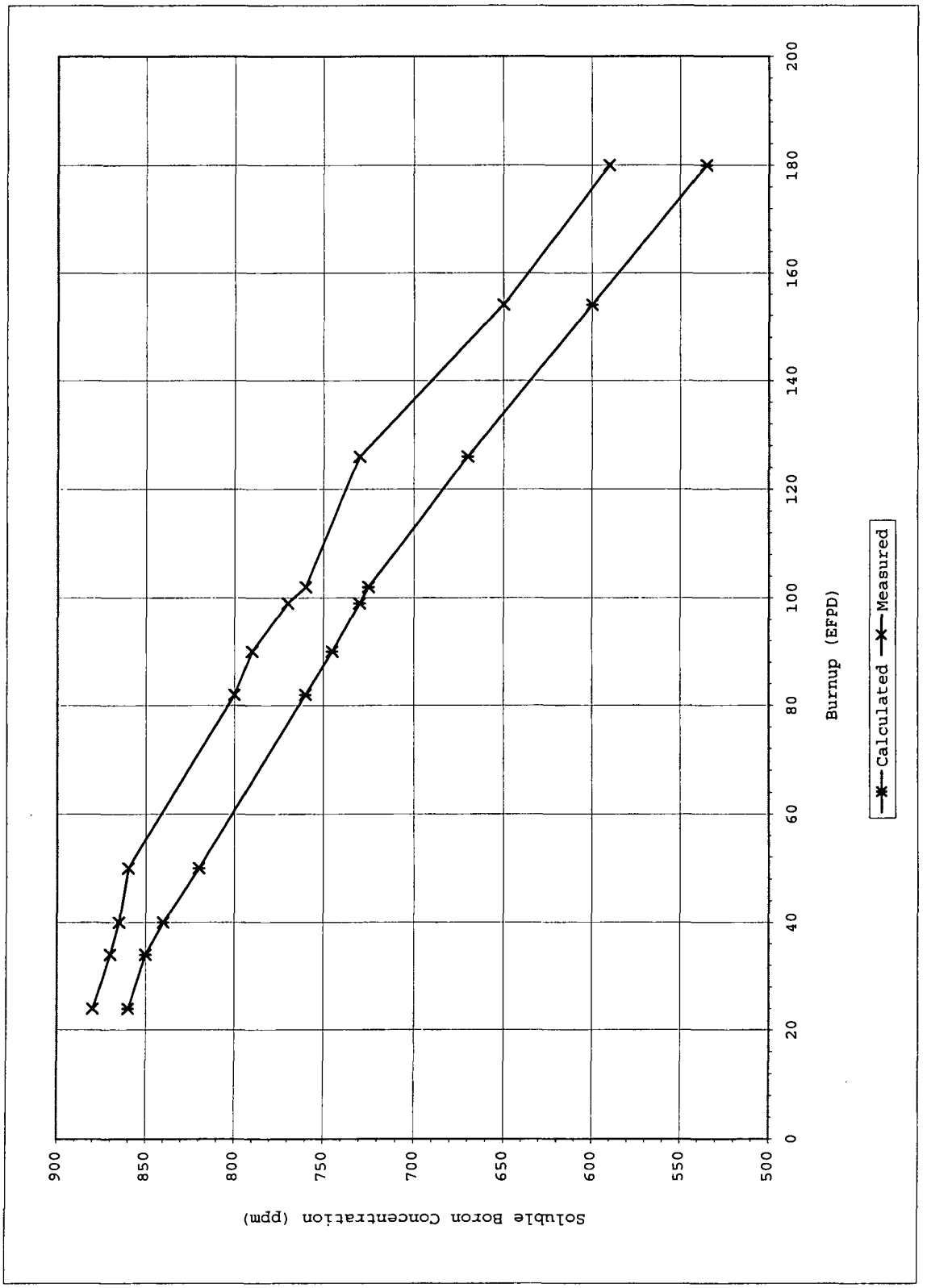
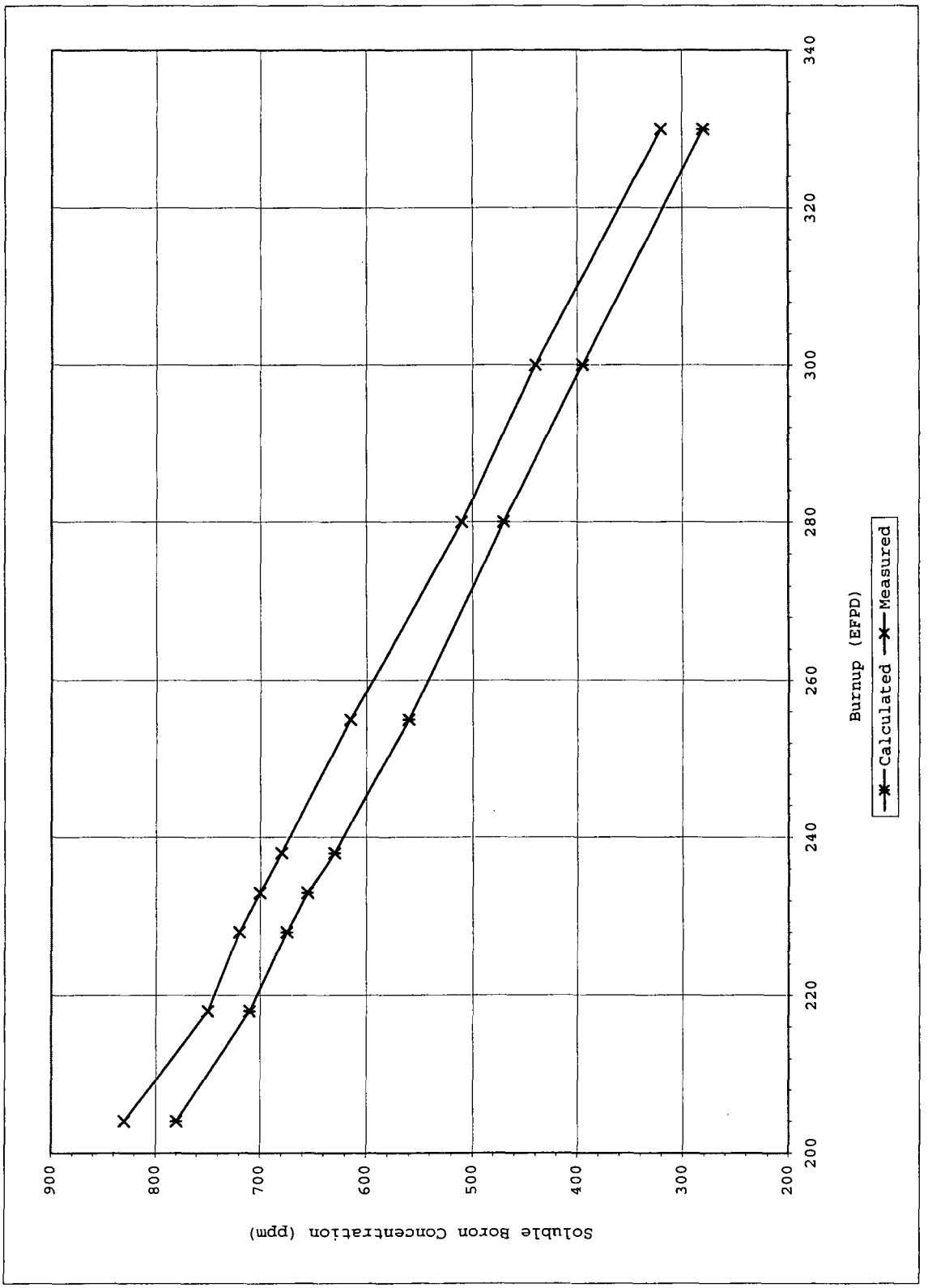


FIGURE 10-2

McGuire 1 Cycle 1A
Boron Letdown Curves



11. POWER DISTRIBUTION COMPARISONS

Power distribution information contained in this section is presented to provide the perspective on the benchmarking of measured to predicted results. The predicted results are based on the PDQ07/EPRI-NODE-P models. Reference 28 presents the benchmarking results for the CASMO-3/SIMULATE-3 models.

11.1 Introduction and Summary

11.1.1 Introduction

The nuclear code employed in this section to calculate three dimensional assembly power calculations is EPRI-NODE-P. Additional two dimensional calculations are performed with PDQ07. The EPRI-NODE-P code has been benchmarked against McGuire Unit 1 Cycle 1 and part of Cycle 1A. It has also been benchmarked against TVA's Sequoyah Unit 1 Cycle 1.

This work encompassed: derivation of measured power distributions for the above cycles, simulations of the above cycles using EPRI-NODE-P, development of fitting procedures for the calculated assembly peak axial powers, and development of a statistical basis for estimating the calculational accuracy of EPRI-NODE-P.

11.1.2 Summary

A data base consisting of McGuire Unit 1 Cycle 1 and part of Cycle 1A, and TVA's Sequoyah Unit 1 Cycle 1, measured and EPRI-NODE-P calculated fuel assembly powers was assembled. Calculated and measured powers were statistically combined to derive 95/95 Observed Nuclear Reliability Factors (ONRF) for EPRI-NODE-P. ONRF's were calculated for assembly radial powers, assembly peak axial powers, and assembly normalized axial powers. The assembly radial power ($F_{\Delta H}$) is defined as the ratio of assembly average power to core average power. The assembly peak axial power (F_Q) is defined as the maximum assembly x-y planar average power along the fuel assembly length

relative to the core average power. The assembly axial power (F_Z) is defined as the ratio of the assembly peak axial power and the assembly radial power ($F_Q/F_{\Delta H}$).

ORNFs of 1.03 for the assembly radial powers, 1.06 for the assembly peak axial powers, and 1.05 for the assembly normalized axial powers were determined.

11.2 Measured Data

11.2.1 Measured Assembly Power Data

The measured power data base comprises assembly power data from McGuire Unit 1, Cycle 1 and part of Cycle 1A, and TVA's Sequoyah Unit 1, Cycle 1. All measured assembly power data are directly traceable to signals from the incore detector system.

11.2.2 Measurement System Description

The incore detector systems at McGuire and Sequoyah consist of 6 movable miniature fission chamber neutron detectors. The detectors are inserted into the bottom of the reactor vessel and driven up through the core to the top. They are then slowly withdrawn through the core. Incore flux maps are obtained by taking voltage signal readings from the detectors as they are withdrawn through the core. This data is then stored on the plant computer.

The detectors travel inside thimbles that are located in the Instrument Guide Tube of the fuel assemblies. There are 58 instrumented assemblies out of a total of 193 fuel assemblies. There are 61 voltage signals recorded axially along each of instrumented fuel assemblies. The instrumented fuel assemblies are shown on Figure 11-1.

The detectors are inter-calibrated by inserting each detector into one reference (calibration) fuel assembly. After each flux map the detector signals are processed by Shanstrom Nuclear Associates Code for Operating

Reactor Evaluation (SNA-CORE)[26]. SNA-CORE uses the 58 x 61 array of signals to calculate peaking factors, (radial powers and assembly peak axial powers) for each of the 193 assemblies. The 193 radial powers and assembly peak axial powers are then averaged into eighth core or quarter core, depending on the cycle. These peaking factors then make up the measured data base. All power measurements were taken at approximately equilibrium xenon conditions. Tables 11-1, 11-2, and 11-3 show the selected reactor state points.

11.3. EPRI-NODE-P Power Distribution Comparisons

11.3.1 *EPRI-NODE-P Model*

As noted previously, EPRI-NODE-P was used to calculate the three dimensional power distribution data presented in this section. This code can be used for all maneuvering analyses, core follow, and physics test data where three-dimensional core power distributions are required. In this section, comparisons of measured and EPRI-NODE-P calculated values will be shown for both radial powers and assembly peak axial powers. Comparisons were performed on a total of 37 reactor state points covering McGuire Unit 1, Cycle 1 and part of 1A, and Sequoyah Unit 1, Cycle 1.

McGuire Unit 1, Cycle 1 and Sequoyah Unit 1, Cycle 1 were modeled using eighth core symmetry. McGuire Unit 1, Cycle 1A was modeled using quarter core symmetry. Each fuel assembly was modeled with one radial and 12 equidistant axial nodes. The active stack height was set at 144 inches. Control rods could be positioned continuously in this model. Simulations of the McGuire and Sequoyah cores were performed using methods described in Section 3.5 and 5.2.

11.3.2 Fuel Cycle Simulations

Using the EPRI-NODE-P model described in section 11.3.1, McGuire Unit 1, Cycles 1 and part of 1A, and TVA's Sequoyah Unit 1, Cycle 1 were depleted

using thermal and hydraulic feedbacks. The depletions were performed in a core follow mode, utilizing critical boron searches at each exposure step.

McGuire Unit 1, Cycle 1 operated until 191.5 EFPD. Control and shutdown bank locations are shown on Figure 11-2. The core loading pattern is shown on Figure 11-3. During this time the unit was operated mostly at the 50% and 75% power plateaus because of power limitations imposed by steam generator flow impingement problems.

The EPRI-NODE-P radial powers were normalized to PDQ07 depletion at 25 EFPD for McGuire Unit 1, Cycle 1. There were 25 state points for this cycle. These are shown on Table 11-1. Figures 11-6 to 11-30 show comparisons of calculated and measured radial powers. Figure 11-31 to 11-55 show comparisons of calculated and measured assembly peak axial powers.

The data used for McGuire Unit 1, Cycle 1A was through 250 EFPD. Control and shutdown bank locations are the same as those for McGuire Unit 1, Cycle 1. The core loading pattern for cycle 1A was the same as the loading pattern for Cycle 1 except all but 2 burnable poison rods were removed. The two that remained were in core locations H-3 and H-13. The unit was operated mostly at 100% power during this time after the steam generator flow impingement problem was corrected.

The EPRI-NODE-P radial powers were normalized to PDQ07 depletion at 257 EFPD for McGuire Unit 1, Cycle 1A. There were 5 state points for the part of this cycle that was used. These are shown on Table 11-2. Figures 11-56 to 11-60 show comparison of calculated and measured radial powers. Figures 11-61 to 11-65 show comparisons of calculated and measured assembly peak axial powers.

TVA's Sequoyah Unit 1, Cycle 1 operated until the end of cycle which lasted 390 EFPD. Control and shutdown bank locations are shown on Figure 11-4. The core loading pattern is shown on Figure 11-5.

The EPRI-NODE-P radial powers were normalized to PDQ07 depletion at 25 EFPD

for Sequoyah Unit 1, Cycle 1. There were 7 state points for this cycle. These are shown on Table 11-3. Figures 11-66 to 11-72 show comparison of calculated and measured radial powers. Figures 11-73 to 11-79 show comparison of calculated and measured assembly peak axial powers.

11.3.3 Radial Power Methodology

The radial powers are radial peaking factors. Therefore, the radial peaking factors from SNA-CORE are compared directly to the normalized radial powers (P(I,J)) from EPRI-NODE-P.

11.3.4 Assembly Peak Axial Power Methodology

The assembly peak axial powers are peaking factors. There are 61 assembly axial powers for each fuel assembly calculated by SNA-CORE. Of these 61 assembly axial powers, the maximum is chosen for the "measured" assembly peak axial power. The EPRI-NODE-P model calculated 12 nodal axial powers per assembly. The assembly peak axial power could not be compared directly to the maximum nodal power.

Therefore, the nodal axial powers were curve fit using the following equation:

$$P(z) = \sum_{n=1}^{3} A_n \mathrm{Sin}(n\pi z) + B_n \mathrm{Cos}(n\pi z)$$

Where: A_n, B_n = Fourier series coefficients

z = normalized vertical axis variable

n = Fourier sequence number

The 12 level node powers were fit, yielding 61 assembly axial powers for each assembly at each state point. The assembly peak axial power was then selected from the 61 calculated assembly axial powers and the 12 nodal powers.

11.3.5 Conclusions

EPRI-NODE-P yielded consistently good power distributions when compared to measured power distributions. This conclusion applies for both radial and assembly peak axial power comparisons. Although the conclusions in this section are qualitative, quantitative statistical results of these comparisons will be shown in Section 11.5.

11.4 PDQ07 - Power Distribution Comparisons

Radial power distributions from the PDQ07 depletions of McGuire Unit 1, Cycle 1, Cycle 1A, and Sequoyah Unit 1, Cycle 1 were compared to measured radial power distributions from SNA-CORE at various burnups. The PDQ07 model employed a 2-dimensional geometry with two neutron energy groups. (For additional information concerning the use of this code, refer to Section 3.4). All power distributions from PDQ07 were performed at hot full power all rods out. Table 11-4 compares the state points of the measured data to that of PDQ07. Figures 11-80 to 11-86 show the comparisons of the radial powers.

11.5. Statistical Analysis

11.5.1 Observed Nuclear Reliability Factor Derivation

This section will address quantitatively statistics arising from Section 11.3. Normal distribution theory will be used in deriving calculational uncertainties.

In deriving the calculational uncertainty for EPRI-NODE-P, the algebraic difference between a calculated and a measured value forms a normally distributed (refer to Section 11.5.2) random variable.

The difference variable is defined:

$$D_i = C_i - M_i \tag{11-1}$$

where: D is the ith difference; $1 \leq i \leq N$

C is the ith calculated value (radial or assembly peak axial power)

M is the ith measured value (radial or assembly peak axial power)

The mean of the difference as defined in equation 11-2 is:

$$\bar{D} = \bar{C} - \bar{M} \tag{11-2}$$

where: $$\bar{C} = (\sum_{i=1}^{n} C_i) \div n \tag{11-2a}$$

$$\bar{M} = (\sum_{i=1}^{n} M_i) \div n \tag{11-2b}$$

$$\bar{D} = (\sum_{i=1}^{n} D_i) \div n \tag{11-2c}$$

n = number of observations in sample

Now a one sided upper bound factor is derived by employing One Sided Upper Tolerance Limit (OSUTL) methodology. For a normal random variable X with a sample mean X and standard deviation S, the OSUTL of X is defined by:

$$OSUTL(X) = \bar{X} + K \times S \tag{11-3}$$

where: $$\bar{X} = (\sum_{i=1}^{n} X_i) \div n \tag{11-4}$$

$$S = [(\sum_{i=1}^{n} (X_i - \bar{X})^2) \div (n-1)]^{1/2} \tag{11-5}$$

In equation 11-3, K is the one-sided tolerance factor. Equation 11-3 is

formulated such that a predetermined proportion of the population (P) is below the OSUTL with a confidence factor (α)[25]. K is explicitly dependent on n, P, and α. Following industry practice, P = 95% and α = 95%.

The OSUTL is given for D by:

$$OSUTL(D) = \bar{D} + K \times S(D) \tag{11-6}$$

C is a deterministic variable and does not have an OSUTL per se, but a reasonable upper limit to C can be defined by:

$$UL(C) = \bar{M} + OSUTL(D) \tag{11-7}$$

$$UL(C) = \bar{M} + \bar{D} + K \times S(D) \tag{11-7a}$$

If one substitutes equations 11-2 into equation 11-7 you obtain the following:

$$UL(C) = \bar{M} + \bar{C} - \bar{M} + K \times S(D) \tag{11-8}$$

or $$UL(C) = \bar{C} + K \times S(D) \tag{11-8a}$$

From equation (11-8a), it is more obvious that the upper limit is a function of the calculated parameter. Also, it is obvious that the standard deviation being associated with the calculated limit is that of the difference distribution. This means that any error in the measurement of the radial or assembly peak axial power as well as any calculational error will be included in the UL(C) parameter. While equation 11-7a and 11-8a are valid, the definition of $\bar{D} = \bar{C} - \bar{M}$ (equation 11-2) leads to UL(C) being smaller if the measured parameter is underpredicted. The conservative solution to this is to subtract $\bar{D}$ in equation 11-7a instead of adding it. This would yield the following equation:

$$UL(C) = \bar{M} + \bar{D} + K \times S(D) \tag{11-9}$$

Finally, the Observed Nuclear Reliability Factor (ONRF) is defined as the quotient of UL(C) from equation 11-9 and the mean of the measurements:

$$ONRF = \frac{UL(C)}{\bar{M}} \tag{11-10}$$

or

$$ONRF = \frac{\bar{M} - \bar{D} + K \times S(D)}{\bar{M}} \tag{11-11}$$

The ONRF from equation 11-11 will be used as a multiplicative factor applied to EPRI-NODE-P calculated powers such that:

$$ONRF \times C \geq M \tag{11-12}$$

for 95% of the population and with a confidence factor of 95%. Separate ONRF's are derived for radial and assembly peak axial powers.

This procedure was employed in Reference 3 to statistically evaluate ORNFs for EPRI-NODE-P as part of the Oconee Reload Design Methodology.

11.5.2 Normality Test Results

In analyzing the normality of the difference distributions, C, M data were grouped into the following categories:

1) reactor cycle: McGuire 1, Cycle 1; McGuire 1, Cycle 1A; Sequoyah 1, Cycle 1
2) grouped cycles: All reactor cycles combined
3) type: radial powers or assembly peak axial powers

The difference distributions were analyzed for normality using the D' test from ANSI N15.15 - 1974.[27] Using the engineering judgment that only peaking factors greater than the core average are the area of concern, pairs of C,M where both are $\geq$ 1.0 will be treated. Table 11-5 displays the normality test results. The level of significance was chosen to be .05. Therefore, the D' statistic must be between the .025 and .975 percentage point D' values for normality. Here, 3 out of 4 assembly radial power distributions were normal and 4 assembly peak axial power distributions were normal. The remainder of

the difference distributions yielded D' statistics that were close to the critical values and were therefore classified as nearly normal.

11.5.3 Observed Nuclear Reliability Factors (ONRF) for EPRI-NODE-P

In this subsection the statistical treatment developed in Section 11.5.1 will be utilized to develop ONRF's ($F^R_{\Delta H}$, F^R_Q, and F^R_Z) for McGuire Unit 1, Cycle 1 and part of Cycle 1A, and TVA's Sequoyah Unit 1, Cycle 1, combined.

All pairs of C, M $\geq$ 1.0 from all 37 state points of McGuire Unit 1, Cycle 1 and part of Cycle 1A, and Sequoyah Unit 1, Cycle 1, were obtained. The procedure was applied to radial powers, assembly peak axial powers, and assembly normalized axial powers. The variables shown in equation 11-11 were then derived and the ONRF's calculated.

As an example, for radial ORNF ($F^R_{\Delta H}$)

$\bar{M}$ = 1.131

$\bar{D}$ = 0.002

S(D) = 0.020

N = 846

K = 1.7343 (N = 846, 95%/95%)

Therefore, the ONRF would be:

$$ONRF = \frac{1.131 - 0.002 + (1.7343 \times 0.020)}{1.131} \qquad (11\text{-}13)$$

$$ONRF = 1.029 \qquad (11\text{-}13a)$$

Table 11-6 shows the calculated ORNF's and the data used to calculate them.

11.5.4 Quantitative Comparisons of EPRI-NODE-P to Measurement

By analyzing the variable D as defined in equation 11-1, the accuracy of EPRI-NODE-P can be assessed. Four important statistical properties of D are discussed.

$\bar{D}$ is the mean of the differences between EPRI-NODE-P and measured assembly powers. For McGuire Unit 1, Cycle 1 and part of 1A, and Sequoyah Unit 1, Cycle 1 $\bar{D}$ is 0.002 for radial powers and -0.031 for assembly peak axial powers. The above means were derived from all pairs of C, M $\geq$ 1.0 from all 37 state points. Subsequent statistics are also derived from this consideration.

S(D), the standard deviation of the differences, indicates the spread of the values of D about $\bar{D}$. For the above cycles, S(D) for radial powers is 0.020. S(D) for assembly peak axial powers is 0.028.

The mean of the absolute differences $\overline{ABS(D)}$ and its standard deviation can be combined to give limits on this variable. 95% confidence limits on the means were given by:

$$\overline{ABS(D)}_{U,L} = ABS(D) \pm \frac{t(.05,n) \times S(ABS(D))}{\sqrt{n}} \qquad (11\text{-}14)$$

Equation 11-14 yields

$$\overline{ABS(D)}_{U,L} = 0.018 \pm 0.001$$

for radial powers for C, M pairs $\geq$ 1.0 for all 37 state points and:

$$\overline{ABS(D)}_{U,L} = 0.036 + 0.001$$

for assembly peak axial powers for all C, M pairs $\geq$ 1.0 for all 37 state points.

Tables 11-7 and 11-8 present summary D statistics for radial and assembly peak axial powers, respectively, where C, M $\geq$ 1.0 for all pairs considered.

11.5.5 Relative Percent Differences

The relative percent difference between EPRI-NODE-P calculated values and measured values will be defined:

$$\% \text{ Diff} = \frac{C-M}{M} \times 100 \qquad (11\text{-}15)$$

This section will address relative percent differences derived from:

a) the sample mean
b) the mean of the absolute value

Since negative percent differences represent calculational nonconservatisms, the minimum values will be more important. Relative percent differences for all C, M $\geq$ 1.0 will be discussed.

Combining data for McGuire Unit 1, Cycle 1, and part of Cycle 1A, and Sequoyah Unit 1, Cycle 1, the following results were obtained.

The average percent difference was 0.167 and the absolute 1.555 for radial powers. Also, the average percent difference was -2.195 and the absolute 2.392 for assembly peak axial powers.

Table 11-9 shows summary data for percent differences derived from calculated and measured radial powers. Values are presented by cycle and for all cycles combined. Table 11-10 is similar to Table 11-9 and provides data for assembly peak axial power percent differences.

11.5.6 Conclusions

A statistical analysis of EPRI-NODE-P calculated and plant measured power distributions has been performed. The resulting ONRF's for all C, M pairs $\geq$ 1.0 for all 37 state points are:

($F^R_{\Delta H}$) Assembly Radial ONRF	(F^R_Q) Assembly Peak Axial ONRF	(F^R_Z) Assembly Normalized Axial Power ONRF
1.03	1.06	1.05

These values while based upon calculations and measurements performed on McGuire Unit 1, Cycles 1 and part of 1A, and Sequoyah Unit 1, Cycle 1 are applicable to all McGuire and Catawba units for the following reasons:

1. McGuire, Catawba, and Sequoyah have identical incore detector systems.

2. All units are manufactured by the same vendor and use similar fuel.

3. Calculations for all units were performed using the same calculational methods and procedures. Similarly, all calculations performed for McGuire and Catawba will use the same calculational methods and procedures.

As an additional verification of the conservatism in the 1.03 radial and 1.06 assembly peak axial ONRF's, all calculated maximum radial powers were multiplied by 1.03 and compared to measured. Similarly all calculated assembly peak axial powers were multiplied by 1.06 and compared to measured. 29 out of 843 (3.4%) radial powers exceeded the 1.03 x maximum calculated radial power. 43 out of 1038 (4.1%) assembly peak axial powers exceeded the 1.06 x maximum calculated assembly peak axial power. Therefore, the 1.03 radial factor was satisfactory for the entire population. The 1.06 assembly peak axial factor was also satisfactory for the entire population.

For pin power distributions, the uncertainty in the assembly power distribution is statistically combined with the uncertainty in the radial local factor (2% see Section 8.5) and the uncertainty for manufacturing tolerance (3%).

The pin total peak uncertainty factor (F_Q^{SCUF}) is calculated below.

$$F_Q^{SCUF} = 1+\frac{0.031}{1.375}+\sqrt{(0.03)^2+(0.035)^2+(0.02)^2} = 1.073$$

Similary, the pin radial peak uncertainty factor ($F_{\Delta H}^{SCUF}$) is calculated below, not including the bias term.

$$F_{\Delta H}^{SCUF} = 1+\sqrt{(0.03)^2+(0.03)^2+(0.02)^2} = 1.047$$

Finally, the assembly normalized axial peak uncertainty factor (F_Z^{SCUF}) is calculated below.

$$F_Z^{SCUF} = 1+\frac{0.032}{1.251}+\sqrt{(0.022)^2} = 1.048$$

TABLE 11-1

McGuire Unit 1 Cycle 1 State Points

Point #	EFPD	Power (%)	Control Bank D Position (Steps)	Axial Offset (Meas/Calc) (%)
1	1.28	30	213	-4.67 / -4.78
2	5.27	30	170	-10.68 / -9.20
3	7.70	48	200	-7.59 / -6.83
4	11.42	48	164	-11.90 /-11.07
5	37.10	50	186	-8.76 / -7.70
6	41.59	50	201	-5.56 / -6.30
7	48.75	50	201	-6.27 / -6.01
8	59.37	50	201	-5.06 / -5.83
9	75.38	50	198	-6.10 / -5.86
10	80.46	75	213	-8.57 / -6.94
11	91.54	75	213	-7.41 / -6.75
12	104.47	50	215	-4.07 / -3.58
13	112.05	50	215	-1.57 / -3.43
14	115.69	75	217	-5.61 / -6.52
15	118.71	50	180	-8.60 / -7.50
16	122.15	75	215	-5.58 / -6.36
17	130.59	75	215	-7.58 / -6.17
18	135.44	75	215	-5.77 / -5.99
19	139.82	50	180	-8.43 / -6.82
20	151.42	50	215	-0.54 / -2.52
21	146.01	75	215	-4.80 / -5.86
22	150.19	50	215	-0.70 / -2.32
23	162.76	50	215	-4.80 / -2.33
24	173.34	50	215	-0.29 / -2.27
25	185.58	50	215	-0.45 / -2.24

TABLE 11-2

McGuire Unit 1 Cycle 1A State Points

Point #	EFPD	Power (%)	Control Bank D Position (Steps)	Axial Offset (Meas/Calc) (%)
1	198.66	90	217	0.73 / -0.93
2	217.53	100	209	1.35 / -5.05
3	223.35	100	211	-3.51 / -4.92
4	236.23	100	211	-3.44 / -4.89
5	249.75	100	221	-2.51 / -3.77

TABLE 11-3

Sequoyah Unit 1 Cycle 1 State Points

Point #	EFPD	Power (%)	Control Bank D Position (Steps)	Axial Offset (Meas/Calc) (%)
1	71.82	100	200	-7.31 / -9.01
2	101.62	100	218	-4.36 / -6.19
3	133.29	100	216	-3.95 / -5.60
4	166.04	100	210	-2.68 / -5.51
5	231.70	100	216	-1.36 / -3.77
6	290.04	100	216	-1.51 / -3.40
7	378.92	100	222	-1.43 / -2.86

TABLE 11-4

McGuire Unit 1 Cycles 1 and 1A and Sequoyah Unit 1 Cycle 1
State Points for PDQ07 Calculated and Measured Data

			PDQ07 Calculated			Measured		
Pt #	Unit	Cycle	EFPD	Control Bank D Position (Steps)	Power (%)	EFPD	Control Bank D Position (Steps)	Power (%)
1	M1	1	52.2	228	100	48.8	200	50
2	M1	1	104.4	228	100	104.5	218	50
3	M1	1	156.7	228	100	150.2	216	50
4	M1	1A	208.9	228	100	198.7	210	90
5	S1	1	103.6	228	100	101.6	216	100
6	S1	1	155.5	228	100	133.3	216	100
7	S1	1	362.7	228	100	378.9	222	100

TABLE 11-5

Difference Distribution Normality Tests
for C, M ≥ 1.0 - 5% Level of Significance

Assembly Radial Powers

Unit/Cycle	N	D' (P=.025)	D'	D' (P=.975)	Remarks
M1/C1	510	3215.0	3274.7	3275.0	Normal
M1/C1A	190	725.9	746.0	748.1	Normal
S1/C1	146	487.6	491.9	504.6	Normal
All Combined	846	6886.7	7000.9	6986.2	Nearly Normal

Assembly Peak Axial Powers

Unit/Cycle	N	D' (P=.025)	D'	D' (P=.975)	Remarks
M1/C1	642	4546.4	4586.3	4621.7	Normal
M1/C1A	220	904.9	922.9	930.5	Normal
S1/C1	176	646.4	646.4	666.9	Normal
All Combined	1038	9345.5	9379.5	9489.8	Normal

TABLE 11-6

Calculated ONRFs and Associated Data

Assembly Radial Power ONRF ($F^R_{\Delta H}$)

$\bar{M}$	= 1.131
$\bar{D}$	= 0.002
S(D)	= 0.020
N	= 846
K	= 1.7343 (N = 846, 95%/95%)
ONRF($F^R_{\Delta H}$)	= 1.029

Assembly Peak Axial Power ONRF (F^R_Q)

$\bar{M}$	= 1.375
$\bar{D}$	= -0.031
S(D)	= 0.028
N	= 1038
K	= 1.7259 (N = 1038, 95%/95%)
ONRF(F^R_Q)	= 1.058

Assembly Axial Power ONRF (F^R_Z)

$\bar{M}$	= mean value of $(F_Q/F_{\Delta H})$meas.	= 1.251
$\bar{D}$	= mean value of $[(F_Q/F_{\Delta H})$meas. - $(F_Q/F_{\Delta H})$calc.]	= 0.032
S(D)	= 0.016	
N	= 846	
K	= 1.7343 (N = 846, 95%/95%)	
ONRF(F^R_Z)	= 1.048	

TABLE 11-7

Difference, Means, and Standard Deviations for Assembly Radial Powers (C, M ≥1.0)

Unit/Cycle	N	$\bar{D}$	S(D)	$\overline{ABS}$	S(ABS(D))
M1/C1	510	-0.001	0.019	0.017	0.008
M1/C1A	190	-0.001	0.025	0.023	0.010
S1/C1	146	0.013	0.012	0.014	0.010
All Combined	846	0.002	0.020	0.018	0.010

TABLE 11-8

Difference, Means, and Standard Deviations for Assembly Peak Axial Powers (C, M ≥1.0)

Unit/Cycle	N	$\bar{D}$	S(D)	$\overline{ABS}$	S(ABS(D))
M1/C1	642	-0.029	0.027	0.032	0.023
M1/C1A	220	-0.039	0.033	0.041	0.029
S1/C1	176	-0.028	0.026	0.031	0.023
All Combined	1038	-0.031	0.028	0.036	0.025

TABLE 11-9

Percent Difference Means

(C, M ≥1.0) - Assembly Radial Powers

Unit/Cycle	Mean % Difference	Mean Absolute % Difference
M1/C1	-0.058	1.452
M1/C1A	0.007	2.043
S1/C1	1.163	1.281
All Combined	0.167	1.555

TABLE 11-10

Percent Difference Means

(C, M ≥1.0) - Assembly Peak Axial Powers

Unit/Cycle	Mean % Difference	Mean Absolute % Difference
M1/C1	-2.001	2.196
M1/C1A	-2.838	3.031
S1/C1	-2.099	2.310
All Combined	-2.195	2.392

FIGURE 11-1

Instrumented Fuel Assemblies

McGuire and Sequoyah

								X								
1							X			X						
2				X			X		X							
3									X		X		X		X	
4			X	X					X							
5						X				X		X		X		
6		X		X			X		X						X	
7					X			X			X			X		
8	W	X		X		X		X			X		X	X	X		Y
9			X							X		X				X
10						X		X					X			
11		X				X			X			X				X
12							X			X			X			
13				X		X			X						X	
14				X				X			X		X			
15						X			X							
								Z								
		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

FIGURE 11-2

Control and Shutdown Bank Locations

McGuire 1 Cycle 1

								X								
1																
2					SA		B		C		B		SA			
3						SD		SB		SB		SC				
4			SA		D				SE				D		SA	
5				SC										SD		
6			B				C		A		C				B	
7				SB										SB		
8	W		C		SE		A		D		A		SE		C		Y
9				SB										SB		
10			B				C		A		C				B	
11				SD										SC		
12			SA		D				SE				D		SA	
13						SC		SB		SB		SD				
14					SA		B		C		B		SA			
15																
								Z								
		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

FIGURE 11-3

Core Loading Pattern

McGuire 1 Cycle 1

X

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C	10 C	C	10 C	C	10 C	C				
2			9 C	C	12 C	A	20 C	A	19 C	A	12 C	C	9 C		
3		9 C	C	20 B	A	16 B	A	16 B	A	16 B	A	20 B	C	9 C	
4		C	20 B	B	20 B	A	16 B	A	16 B	A	20 B	B	20 B	C	
5	C	12 C	A	20 B	A	16 B	A	16 B	A	16 B	A	20 B	A	12 C	C
6	10 C	A	16 B	A	16 B	A	20 B	A	20 B	A	16 B	A	16 B	A	10 C
7	C	20 C	A	16 B	A	20 B	A	20 B	A	20 B	A	16 B	A	20 C	C
8 W	10 C	A	16 B	A	16 B	A	20 B	A	20 B	A	16 B	A	16 B	A	10 C Y
9	C	20 C	A	16 B	A	20 B	A	20 B	A	20 B	A	16 B	A	20 C	C
10	10 C	A	16 B	A	16 B	A	20 B	A	20 B	A	16 B	A	16 B	A	10 C
11	C	12 C	A	20 B	A	16 B	A	16 B	A	16 B	A	20 B	A	12 C	C
12		C	20 B	B	20 B	A	16 B	A	16 B	A	20 B	B	20 B	C	
13		9 C	C	20 B	A	16 B	A	16 B	A	16 B	A	20 B	C	9 C	
14			9 C	C	12 C	A	20 C	A	19 C	A	12 C	C	9 C		
15					C	10 C	C	10 C	C	10 C	C				

Z

A — Region 1 (2.1 w/o)

B — Region 2 (2.6 w/o)

C — Region 3 (3.1 w/o)

Number indicates number of burnable poison rods

FIGURE 11-4

Control and Shutdown Bank Locations
Sequoyah 1 Cycle 1

								X								
1																
2					SA		B		C		B		SA			
3						SD		SB		SB		SC				
4			SA		D				D				D		SA	
5				SC										SD		
6			B				C		A		C				B	
7				SB										SB		
8	W		C		D		A		D		A		D		C		Y
9				SB										SB		
10			B				C		A		C				B	
11				SD										SC		
12			SA		D				D				D		SA	
13						SC		SB		SB		SD				
14					SA		B		C		B		SA			
15																
								Z								
		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

FIGURE 11-5

Core Loading Pattern

Sequoyah 1 Cycle 1

X

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C	10 C	C	10 C	C	10 C	C				
2			9 C	C	9 B	A	12 B	A	12 B	A	9 B	C	9 C		
3		9 C	C	16 B	A	16 B	A	20 B	A	16 B	A	16 B	C	9 C	
4		C	16 B	A	16 B	A	16 B	A	16 B	A	16 B	A	16 B	C	
5	C	9 B	A	16 B	A	20 B	A	20 B	A	20 B	A	16 B	A	9 B	C
6	10 C	A	16 B	A	20 B	A	20 B	A	20 B	A	20 B	A	16 B	A	10 C
7	C	12 B	A	16 B	A	20 B	A	12 B	A	20 B	A	16 B	A	12 B	C
8	10 C	A	20 B	A	20 B	A	12 B	A	12 B	A	20 B	A	20 B	A	10 C
9	C	12 B	A	16 B	A	20 B	A	12 B	A	20 B	A	16 B	A	12 B	C
10	10 C	A	16 B	A	20 B	A	20 B	A	20 B	A	20 B	A	16 B	A	10 C
11	C	9 B	A	16 B	A	20 B	A	20 B	A	20 B	A	16 B	A	9 B	C
12		C	16 B	A	16 B	A	16 B	A	16 B	A	16 B	A	16 B	C	
13		9 C	C	16 B	A	16 B	A	20 B	A	16 B	A	16 B	C	9 C	
14			9 C	C	9 B	A	12 B	A	12 B	A	9 B	C	9 C		
15					C	10 C	C	10 C	C	10 C	C				

W Y

Z

A	Region 1 (2.1 w/o)
B	Region 2 (2.6 w/o)
C	Region 3 (3.1 w/o)

Number indicates number of burnable poison rods

FIGURE 11-6

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

1.28 EFPD 30%FP Control Bank D at 213 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.03	.94	1.11	1.10	1.21	1.08	1.06	.73
8	1.01	.96	1.10	1.12	1.20	1.09	1.05	.73
		1.07	.99	1.18	1.11	1.17	1.01	.80
9		1.05	1.01	1.16	1.14	1.15	1.03	.79
			1.15	1.10	1.19	1.06	1.04	.67
10			1.14	1.13	1.17	1.10	1.01	.68
				1.18	1.04	1.12	1.02	.57
11				1.18	1.06	1.12	1.01	.58
					1.27	.95	.87	
12					1.24	.95	.87	
						1.06	.50	Calculated
13						1.02	.49	Measured

FIGURE 11-7

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

5.27 EFPD 30%FP Control Bank D at 170 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	.97 .97	.93 .97	1.12 1.12	1.12 1.15	1.23 1.22	1.10 1.12	1.07 1.07	.74 .74
9		1.08 1.06	1.01 1.03	1.20 1.18	1.13 1.16	1.19 1.17	1.02 1.05	.81 .80
10			1.17 1.15	1.11 1.14	1.19 1.17	1.07 1.10	1.04 1.01	.67 .68
11				1.18 1.17	1.02 1.04	1.11 1.09	1.01 .99	.57 .57
12					1.17 1.15	.92 .91	.85 .85	
13						1.03 .98	.49 .48	Calculated Measured

FIGURE 11-8

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

7.70 EFPD 48%FP Control Bank D at 200 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.05	.97	1.14	1.12	1.22	1.09	1.05	.72	
8	1.05	1.00	1.14	1.15	1.22	1.10	1.03	.72	
		1.10	1.02	1.21	1.13	1.18	1.00	.79	
	9	1.09	1.05	1.19	1.15	1.15	1.02	.78	
			1.18	1.12	1.19	1.06	1.03	.66	
		10	1.15	1.14	1.17	1.09	1.00	.67	
				1.19	1.04	1.11	1.00	.56	
			11	1.17	1.05	1.10	.99	.57	
					1.23	.93	.84		
				12	1.22	.93	.85		
						1.02	.48		Calculated
					13	1.00	.48		Measured

FIGURE 11-9

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

11.42 EFPD 48%FP Control Bank D at 164 Steps Withdrawn

	H	G	F	E	D	C	B	A
	.99	.97	1.15	1.14	1.24	1.11	1.07	.73
8	.99	1.00	1.15	1.18	1.24	1.13	1.05	.73
		1.11	1.04	1.22	1.15	1.19	1.02	.80
9		1.09	1.07	1.21	1.18	1.17	1.04	.79
			1.19	1.13	1.20	1.07	1.03	.66
10			1.18	1.17	1.19	1.10	1.00	.67
				1.18	1.02	1.10	.99	.56
11				1.18	1.04	1.07	.98	.56
					1.13	.90	.83	
12					1.12	.88	.82	
						.99	.47	Calculated
13						.94	.46	Measured

FIGURE 11-10

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

37.10 EFPD 50%FP Control Bank D at 186 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.07	1.03	1.17	1.16	1.23	1.10	1.04	.71	
8	1.10	1.07	1.18	1.18	1.23	1.10	1.03	.70	
		1.14	1.08	1.22	1.15	1.17	1.00	.77	
9		1.15	1.11	1.21	1.17	1.14	1.01	.76	
			1.20	1.15	1.19	1.07	1.01	.64	
10			1.21	1.19	1.18	1.09	.98	.65	
				1.19	1.05	1.09	.98	.54	
11				1.20	1.05	1.06	.97	.55	
					1.17	.92	.81		
12					1.15	.90	.81		
						.98	.47		Calculated
13						.94	.46		Measured

FIGURE 11-11

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

41.59 EFPD 50%FP Control Bank D at 201 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.10	1.04	1.18	1.16	1.22	1.10	1.03	.70
8	1.10	1.08	1.18	1.21	1.22	1.11	1.01	.70
		1.15	1.08	1.22	1.15	1.14	1.00	.76
9		1.15	1.12	1.21	1.18	1.14	1.01	.75
			1.20	1.15	1.19	1.07	1.00	.64
10			1.19	1.19	1.18	1.09	.96	.64
				1.19	1.06	1.09	.97	.54
11				1.19	1.08	1.07	.96	.54
					1.20	.93	.81	
12					1.18	.91	.80	
						.98	.47	Calculated
13						.94	.46	Measured

FIGURE 11-12

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

48.75 EFPD 50%FP Control Bank D at 201 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.11	1.05	1.18	1.17	1.22	1.10	1.03	.70	
8	1.11	1.09	1.19	1.21	1.22	1.12	1.01	.70	
		1.15	1.09	1.22	1.15	1.16	1.00	.76	
9		1.15	1.12	1.21	1.19	1.14	1.01	.75	
			1.20	1.16	1.19	1.07	1.00	.64	
10			1.19	1.19	1.18	1.09	.96	.64	
				1.19	1.07	1.09	.97	.54	
11				1.19	1.08	1.07	.95	.54	
					1.20	.93	.81		
12					1.18	.91	.80		
						.98	.47	Calculated	
13						.94	.46	Measured	

FIGURE 11-13

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

59.37 EFPD 50%FP Control Bank D at 201 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.12	1.07	1.18	1.17	1.22	1.10	1.02	.69
8	1.12	1.10	1.19	1.21	1.22	1.12	1.01	.70
		1.16	1.10	1.22	1.16	1.15	1.00	.75
9		1.16	1.13	1.21	1.18	1.14	1.01	.75
			1.20	1.16	1.19	1.07	.99	.63
10			1.19	1.19	1.17	1.09	.96	.64
				1.19	1.07	1.08	.97	.54
11				1.18	1.07	1.06	.95	.54
					1.19	.93	.80	
12					1.17	.92	.80	
						.97	.46	Calculated
13						.95	.46	Measured

FIGURE 11-14

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

75.38 EFPD 50%FP Control Bank D at 198 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.12	1.09	1.19	1.18	1.21	1.11	1.02	.69	
8	1.12	1.11	1.18	1.21	1.20	1.13	1.00	.69	
		1.17	1.12	1.22	1.16	1.15	1.00	.75	
9		1.15	1.14	1.20	1.19	1.13	1.02	.74	
			1.21	1.17	1.18	1.07	.98	.63	
10			1.19	1.20	1.17	1.10	.96	.64	
				1.19	1.08	1.08	.96	.53	
11				1.18	1.09	1.06	.95	.54	
					1.18	.93	.80		
12					1.16	.92	.80		
						.96	.46		Calculated
13						.94	.46		Measured

FIGURE 11-15

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

80.46 EFPD 75%FP Control Bank D at 213 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.17 1.16	1.11 1.14	1.20 1.20	1.19 1.22	1.21 1.20	1.10 1.11	1.01 1.00	.68 .70
9		1.18 1.17	1.13 1.15	1.22 1.21	1.17 1.19	1.14 1.12	.99 1.01	.74 .74
10			1.21 1.19	1.18 1.19	1.18 1.17	1.07 1.09	.98 .95	.63 .64
11				1.19 1.18	1.09 1.09	1.07 1.05	.95 .95	.53 .53
12					1.19 1.17	.93 .92	.79 .80	
13						.95 .94	.46 .46	Calculated Measured

FIGURE 11-16

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

91.54 EFPD 75%FP Control Bank D at 213 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.17	1.12	1.20	1.19	1.20	1.11	1.00	.68	
8	1.15	1.15	1.20	1.22	1.19	1.11	.99	.70	
		1.19	1.14	1.22	1.17	1.14	.99	.73	
	9	1.17	1.16	1.20	1.19	1.11	1.01	.73	
			1.21	1.18	1.18	1.08	.97	.63	
		10	1.19	1.21	1.16	1.10	.94	.63	
				1.19	1.09	1.07	.95	.53	
			11	1.18	1.11	1.05	.95	.53	
					1.18	.94	.79		
				12	1.17	.93	.79		
						.94	.46		Calculated
					13	.93	.46		Measured

FIGURE 11-17

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

104.47 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.16	1.12	1.19	1.18	1.19	1.10	1.00	.68
8	1.15	1.14	1.19	1.22	1.19	1.12	.99	.69
		1.17	1.13	1.20	1.16	1.13	1.00	.74
9		1.17	1.15	1.19	1.19	1.11	1.02	.73
			1.20	1.18	1.17	1.08	.97	.63
10			1.18	1.19	1.16	1.10	.95	.64
				1.18	1.10	1.07	.96	.53
11				1.16	1.10	1.06	.95	.53
					1.20	.95	.80	
12					1.18	.94	.80	
						.96	.46	Calculated
13						.94	.46	Measured

FIGURE 11-18

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

112.05 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.16	1.12	1.19	1.18	1.19	1.10	1.00	.68	
8	1.14	1.14	1.18	1.21	1.19	1.12	.99	.70	
		1.17	1.14	1.20	1.16	1.13	1.00	.74	
	9	1.16	1.15	1.19	1.19	1.11	1.01	.74	
			1.19	1.18	1.17	1.08	.97	.63	
		10	1.18	1.20	1.15	1.10	.95	.64	
				1.18	1.10	1.07	.97	.53	
			11	1.17	1.10	1.06	.95	.53	
					1.20	.96	.80		
				12	1.18	.94	.80		
						.96	.46		Calculated
					13	.95	.47		Measured

FIGURE 11-19

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

115.69 EFPD 75%FP Control Bank D at 217 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.18 1.16	1.14 1.16	1.20 1.20	1.19 1.22	1.19 1.19	1.11 1.12	1.00 .99	.68 .69
9		1.19 1.18	1.15 1.16	1.21 1.19	1.17 1.19	1.13 1.11	1.00 1.01	.73 .73
10			1.20 1.19	1.18 1.20	1.17 1.15	1.08 1.10	.97 .95	.63 .64
11				1.18 1.17	1.10 1.10	1.07 1.05	.95 .95	.53 .53
12					1.18 1.17	.95 .94	.79 .80	
13						.94 .94	.46 .46	Calculated Measured

FIGURE 11-20

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

118.71 EFPD 50%FP Control Bank D at 180 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.09	1.11	1.19	1.20	1.20	1.12	1.02	.70	
8	1.08	1.13	1.18	1.23	1.19	1.14	1.01	.72	
		1.17	1.15	1.21	1.17	1.14	1.02	.75	
9		1.15	1.16	1.19	1.20	1.12	1.04	.75	
			1.20	1.18	1.17	1.09	.98	.63	
10			1.18	1.20	1.15	1.11	.96	.65	
				1.17	1.08	1.06	.96	.53	
11				1.16	1.09	1.04	.96	.54	
					1.12	.93	.79		
12					1.10	.92	.79		
						.94	.46		Calculated
13						.92	.46		Measured

FIGURE 11-21

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

122.15 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.17 1.15	1.14 1.16	1.20 1.19	1.19 1.22	1.19 1.18	1.11 1.12	1.00 .99	.68 .70
9		1.19 1.17	1.15 1.17	1.20 1.18	1.17 1.19	1.13 1.10	1.00 1.02	.73 .73
10			1.20 1.18	1.18 1.21	1.17 1.15	1.08 1.10	.97 .94	.63 .64
11				1.18 1.17	1.10 1.11	1.07 1.05	.96 .95	.53 .53
12					1.18 1.17	.95 .94	.79 .79	
13						.94 .93	.46 .47	Calculated Measured

FIGURE 11-22

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

130.59 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.17	1.15	1.19	1.19	1.18	1.11	1.00	.68	
8	1.16	1.16	1.18	1.22	1.17	1.13	.99	.70	
		1.19	1.15	1.20	1.17	1.12	1.00	.73	
	9	1.17	1.17	1.18	1.18	1.10	1.01	.73	
			1.20	1.18	1.16	1.08	.97	.63	
		10	1.18	1.20	1.14	1.10	.94	.64	
				1.17	1.11	1.07	.96	.53	
			11	1.17	1.11	1.05	.95	.53	
					1.18	.96	.79		
				12	1.17	.95	.80		
						.94	.46		Calculated
					13	.94	.47		Measured

FIGURE 11-23

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

135.44 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.17	1.15	1.19	1.19	1.18	1.11	1.00	.68
8	1.16	1.17	1.19	1.22	1.17	1.12	.99	.70
		1.18	1.16	1.20	1.17	1.12	1.00	.73
9		1.17	1.17	1.18	1.19	1.10	1.02	.73
			1.19	1.18	1.16	1.08	.97	.63
10			1.18	1.20	1.15	1.10	.94	.64
				1.17	1.11	1.07	.96	.53
11				1.16	1.11	1.05	.95	.53
					1.18	.96	.79	
12					1.17	.95	.80	
						.95	.46	Calculated
13						.94	.47	Measured

FIGURE 11-24

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

139.82 EFPD 50%FP Control Bank D at 180 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.09	1.12	1.18	1.19	1.19	1.12	1.01	.70	
8	1.08	1.14	1.18	1.23	1.19	1.14	1.01	.71	
		1.17	1.15	1.20	1.17	1.13	1.03	.75	
	9	1.15	1.17	1.19	1.20	1.12	1.04	.75	
			1.19	1.18	1.16	1.09	.98	.64	
		10	1.18	1.20	1.14	1.11	.96	.65	
				1.16	1.09	1.06	.97	.54	
			11	1.15	1.09	1.04	.95	.54	
					1.11	.94	.79		
				12	1.11	.93	.79		
						.94	.46		Calculated
					13	.92	.46		Measured

FIGURE 11-25

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

141.52 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.16	1.13	1.18	1.18	1.17	1.11	1.00	.69	
8	1.15	1.16	1.18	1.21	1.17	1.12	.99	.70	
		1.17	1.14	1.19	1.16	1.12	1.01	.74	
9		1.16	1.16	1.17	1.18	1.10	1.02	.74	
			1.18	1.18	1.16	1.08	.97	.63	
10			1.17	1.20	1.14	1.10	.95	.64	
				1.17	1.11	1.07	.97	.53	
11				1.16	1.11	1.06	.96	.53	
					1.19	.97	.80		
12					1.17	.96	.81		
						.96	.47		Calculated
13						.95	.47		Measured

FIGURE 11-26

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

146.01 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.17	1.15	1.19	1.19	1.17	1.11	1.00	.68	
8	1.16	1.18	1.18	1.22	1.17	1.12	.98	.70	
		1.18	1.16	1.19	1.16	1.12	1.01	.74	
	9	1.17	1.18	1.18	1.19	1.09	1.02	.73	
			1.19	1.18	1.16	1.09	.97	.63	
		10	1.17	1.21	1.14	1.10	.94	.64	
				1.17	1.11	1.07	.96	.53	
			11	1.16	1.12	1.04	.95	.53	
					1.18	.96	.79		
				12	1.16	.95	.79		
						.95	.46		Calculated
					13	.93	.47		Measured

FIGURE 11-27

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

150.19 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.16	1.14	1.17	1.18	1.17	1.11	1.00	.69
8	1.14	1.16	1.17	1.21	1.16	1.12	.99	.71
		1.17	1.15	1.18	1.16	1.12	1.01	.74
9		1.15	1.16	1.16	1.19	1.10	1.03	.74
			1.18	1.17	1.15	1.09	.97	.63
10			1.16	1.20	1.14	1.11	.95	.65
				1.16	1.11	1.07	.97	.54
11				1.16	1.12	1.05	.96	.54
					1.19	.98	.80	
12					1.17	.97	.80	
						.97	.47	Calculated
13						.95	.48	Measured

FIGURE 11-28

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

162.76 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.15	1.14	1.17	1.18	1.16	1.11	1.00	.69	
8	1.14	1.17	1.17	1.21	1.16	1.13	.99	.70	
		1.16	1.15	1.17	1.16	1.11	1.02	.74	
9		1.15	1.17	1.16	1.18	1.09	1.02	.74	
			1.17	1.17	1.15	1.09	.97	.64	
10			1.16	1.20	1.13	1.11	.94	.65	
				1.16	1.12	1.07	.97	.54	
11				1.15	1.12	1.05	.96	.54	
					1.19	.98	.80		
12					1.17	.97	.80		
						.97	.47		Calculated
13						.95	.48		Measured

FIGURE 11-29

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

173.34 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.15	1.14	1.16	1.17	1.15	1.11	1.00	.69	
8	1.13	1.17	1.16	1.21	1.14	1.13	.99	.71	
		1.16	1.15	1.17	1.16	1.11	1.02	.74	
9		1.14	1.17	1.15	1.18	1.09	1.03	.74	
			1.17	1.17	1.14	1.09	.97	.64	
10			1.15	1.19	1.12	1.11	.94	.65	
				1.15	1.12	1.07	.98	.54	
11				1.14	1.12	1.05	.96	.54	
					1.18	.99	.81		
12					1.16	.98	.81		
						.97	.47		Calculated
13						.95	.49		Measured

FIGURE 11-30

McGuire-1 Cy-1 Assembly Radial Powers Calculated vs Measured

185.58 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.15	1.14	1.16	1.17	1.15	1.11	1.00	.70	
8	1.13	1.16	1.15	1.20	1.14	1.12	.99	.72	
		1.16	1.15	1.16	1.15	1.11	1.03	.75	
9		1.14	1.16	1.14	1.18	1.09	1.04	.74	
			1.16	1.17	1.14	1.09	.97	.64	
10			1.14	1.19	1.12	1.11	.95	.66	
				1.15	1.12	1.07	.98	.54	
11				1.14	1.13	1.06	.97	.54	
					1.18	.99	.81		
12					1.17	.99	.81		
						.97	.48		Calculated
13						.96	.49		Measured

FIGURE 11-31

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

1.28 EFPD 30%FP Control Bank D at 213 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.41 1.40	1.27 1.32	1.50 1.50	1.48 1.52	1.63 1.62	1.46 1.47	1.43 1.42	.99 1.00
9		1.45 1.43	1.34 1.38	1.60 1.57	1.50 1.54	1.59 1.55	1.36 1.40	1.09 1.08
10			1.56 1.55	1.49 1.54	1.61 1.59	1.43 1.49	1.41 1.37	.91 .92
11				1.60 1.60	1.42 1.45	1.52 1.52	1.38 1.37	.77 .78
12					1.74 1.71	1.29 1.29	1.18 1.18	
13						1.45 1.39	.68 .67	Calculated Measured

FIGURE 11-32

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

5.27 EFPD 30%FP Control Bank D at 170 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.50	1.33	1.56	1.53	1.68	1.50	1.47	1.01
8	1.48	1.40	1.58	1.61	1.71	1.55	1.47	1.03
		1.50	1.39	1.65	1.55	1.63	1.40	1.11
9		1.51	1.45	1.65	1.62	1.62	1.45	1.12
			1.61	1.54	1.65	1.47	1.44	.93
10			1.62	1.61	1.66	1.54	1.42	.95
				1.65	1.46	1.56	1.41	.79
11				1.67	1.50	1.57	1.40	.80
					1.81	1.33	1.20	
12					1.76	1.33	1.21	
						1.47	.69	Calculated
13						1.43	.69	Measured

FIGURE 11-33

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

7.70 EFPD 48%FP Control Bank D at 200 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.46	1.33	1.54	1.52	1.65	1.47	1.42	.97
8	1.47	1.38	1.56	1.58	1.66	1.50	1.40	.99
		1.49	1.38	1.63	1.52	1.58	1.35	1.06
9		1.50	1.43	1.62	1.58	1.57	1.39	1.07
			1.59	1.52	1.62	1.43	1.39	.89
10			1.58	1.57	1.61	1.49	1.36	.91
				1.61	1.42	1.51	1.35	.76
11				1.62	1.45	1.51	1.35	.77
					1.71	1.27	1.14	
12					1.71	1.28	1.16	
						1.39	.66	Calculated
13						1.37	.66	Measured

FIGURE 11-34

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

11.42 EFPD 48%FP Control Bank D at 164 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.54	1.38	1.59	1.57	1.70	1.51	1.45	.99
8	1.54	1.46	1.64	1.66	1.73	1.57	1.46	1.01
		1.55	1.43	1.67	1.57	1.63	1.38	1.09
9		1.57	1.51	1.70	1.65	1.63	1.45	1.10
			1.64	1.56	1.66	1.47	1.42	.91
10			1.67	1.65	1.68	1.55	1.41	.94
				1.66	1.47	1.54	1.38	.77
11				1.69	1.52	1.56	1.39	.80
					1.76	1.30	1.16	
12					1.75	1.32	1.19	
						1.41	.67	Calculated
13						1.40	.67	Measured

FIGURE 11-35

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

37.10 EFPD 50%FP Control Bank D at 186 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.52	1.40	1.57	1.55	1.63	1.46	1.37	.93
8	1.58	1.50	1.64	1.62	1.67	1.50	1.40	.95
		1.53	1.44	1.62	1.53	1.54	1.33	1.01
9		1.61	1.53	1.66	1.60	1.56	1.38	1.03
			1.60	1.54	1.59	1.43	1.33	.85
10			1.67	1.64	1.63	1.50	1.34	.89
				1.59	1.43	1.46	1.30	.72
11				1.66	1.48	1.48	1.33	.75
					1.65	1.25	1.09	
12					1.66	1.26	1.13	
						1.32	.63	Calculated
13						1.32	.64	Measured

FIGURE 11-36

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

41.59 EFPD 50%FP Control Bank D at 201 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.50	1.39	1.55	1.53	1.60	1.44	1.35	.92
8	1.55	1.49	1.61	1.64	1.65	1.51	1.37	.95
		1.52	1.43	1.60	1.52	1.52	1.31	1.00
9		1.57	1.52	1.64	1.60	1.53	1.37	1.01
			1.58	1.52	1.57	1.41	1.31	.84
10			1.62	1.61	1.60	1.48	1.30	.86
				1.57	1.42	1.44	1.28	.71
11				1.62	1.48	1.45	1.29	.73
					1.62	1.24	1.07	
12					1.64	1.25	1.09	
						1.30	.62	Calculated
13						1.29	.62	Measured

FIGURE 11-37

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

48.75 EFPD 50%FP Control Bank D at 201 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.50	1.40	1.54	1.53	1.58	1.44	1.33	.91
8	1.56	1.50	1.62	1.64	1.64	1.51	1.36	.94
		1.51	1.43	1.59	1.51	1.50	1.30	.99
9		1.57	1.52	1.64	1.60	1.53	1.36	1.01
			1.57	1.52	1.55	1.40	1.29	.83
10			1.62	1.61	1.59	1.47	1.30	.86
				1.56	1.41	1.42	1.27	.70
11				1.61	1.47	1.44	1.28	.72
					1.60	1.23	1.06	
12					1.63	1.25	1.09	
						1.28	.61	Calculated
13						1.29	.62	Measured

FIGURE 11-38

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

59.37 EFPD 50%FP Control Bank D at 201 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.49	1.41	1.53	1.52	1.56	1.43	1.31	.89
8	1.54	1.47	1.59	1.60	1.61	1.48	1.33	.92
		1.51	1.44	1.57	1.50	1.48	1.29	.97
	9	1.55	1.50	1.60	1.56	1.50	1.33	.99
			1.55	1.51	1.53	1.39	1.27	.82
		10	1.58	1.57	1.55	1.44	1.27	.84
				1.54	1.41	1.40	1.25	.69
			11	1.57	1.43	1.41	1.26	.71
					1.58	1.22	1.04	
				12	1.60	1.23	1.07	
						1.26	.60	Calculated
					13	1.27	.61	Measured

FIGURE 11-39

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

75.38 EFPD 50%FP Control Bank D at 198 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.48	1.41	1.51	1.51	1.53	1.41	1.28	.87
8	1.51	1.46	1.54	1.57	1.55	1.46	1.29	.90
		1.49	1.43	1.54	1.48	1.45	1.27	.95
9		1.51	1.48	1.56	1.53	1.46	1.31	.96
			1.53	1.50	1.50	1.37	1.24	.80
10			1.54	1.55	1.51	1.42	1.24	.83
				1.51	1.40	1.37	1.23	.68
11				1.53	1.42	1.38	1.24	.69
					1.54	1.21	1.02	
12					1.56	1.22	1.05	
						1.23	.59	Calculated
13						1.24	.61	Measured

FIGURE 11-40

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

80.46 EFPD 75%FP Control Bank D at 213 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.49	1.42	1.51	1.50	1.51	1.39	1.26	.86
8	1.54	1.48	1.55	1.57	1.54	1.44	1.29	.90
		1.49	1.43	1.52	1.47	1.43	1.25	.93
9		1.53	1.48	1.56	1.53	1.45	1.30	.96
			1.52	1.48	1.48	1.36	1.22	.79
10			1.53	1.53	1.50	1.41	1.23	.82
				1.49	1.38	1.35	1.21	.67
11				1.51	1.40	1.36	1.22	.68
					1.52	1.19	1.00	
12					1.53	1.20	1.04	
						1.20	.58	Calculated
13						1.22	.60	Measured

FIGURE 11-41

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

91.54 EFPD 75%FP Control Bank D at 213 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.47	1.41	1.49	1.48	1.48	1.38	1.25	.85
8	1.50	1.47	1.52	1.56	1.50	1.41	1.26	.89
		1.47	1.43	1.50	1.45	1.41	1.24	.92
9		1.50	1.48	1.52	1.51	1.41	1.28	.93
			1.50	1.47	1.46	1.34	1.21	.78
10			1.51	1.53	1.47	1.39	1.20	.80
				1.47	1.37	1.33	1.19	.66
11				1.49	1.40	1.33	1.20	.67
					1.50	1.19	.99	
12					1.50	1.19	1.01	
						1.19	.57	Calculated
13						1.19	.59	Measured

FIGURE 11-42

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

104.47 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.43	1.38	1.44	1.45	1.44	1.35	1.22	.84
8	1.47	1.44	1.49	1.53	1.48	1.40	1.24	.86
		1.43	1.39	1.45	1.42	1.37	1.23	.90
	9	1.46	1.45	1.49	1.48	1.39	1.27	.92
			1.45	1.44	1.41	1.32	1.18	.77
		10	1.46	1.48	1.44	1.38	1.19	.79
				1.43	1.36	1.31	1.18	.65
			11	1.44	1.38	1.33	1.19	.66
					1.47	1.18	.98	
				12	1.49	1.18	1.00	
						1.18	.57	Calculated
					13	1.18	.58	Measured

FIGURE 11-43

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

112.05 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.41	1.38	1.43	1.43	1.42	1.34	1.20	.83
8	1.44	1.42	1.47	1.51	1.46	1.39	1.23	.86
		1.42	1.39	1.43	1.41	1.35	1.22	.90
	9	1.44	1.43	1.47	1.47	1.37	1.26	.91
			1.43	1.42	1.40	1.31	1.17	.77
		10	1.46	1.48	1.43	1.36	1.18	.79
				1.41	1.35	1.29	1.18	.65
			11	1.45	1.37	1.32	1.19	.66
					1.46	1.18	.97	
				12	1.48	1.18	1.00	
						1.17	.57	Calculated
					13	1.19	.59	Measured

FIGURE 11-44

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

115.69 EFPD 75%FP Control Bank D at 217 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.45	1.40	1.45	1.45	1.44	1.35	1.21	.83
8	1.47	1.44	1.48	1.50	1.46	1.38	1.22	.86
		1.44	1.41	1.46	1.42	1.37	1.23	.90
	9	1.46	1.44	1.47	1.46	1.37	1.25	.90
			1.45	1.44	1.42	1.32	1.18	.77
		10	1.46	1.47	1.42	1.35	1.17	.78
				1.43	1.36	1.30	1.17	.65
			11	1.43	1.35	1.30	1.17	.65
					1.46	1.18	.97	
				12	1.47	1.17	.99	
						1.16	.56	Calculated
					13	1.17	.58	Measured

FIGURE 11-45

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

118.71 EFPD 50%FP Control Bank D at 180 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.46	1.41	1.46	1.46	1.45	1.37	1.23	.85	
8	1.48	1.47	1.50	1.54	1.48	1.43	1.26	.89	
		1.45	1.42	1.47	1.44	1.38	1.25	.92	
	9	1.47	1.47	1.49	1.50	1.40	1.30	.94	
			1.46	1.45	1.43	1.34	1.20	.78	
		10	1.48	1.52	1.45	1.40	1.20	.82	
				1.44	1.37	1.32	1.20	.66	
			11	1.47	1.41	1.34	1.21	.68	
					1.49	1.20	.99		
				12	1.51	1.22	1.02		
						1.20	.58		Calculated
					13	1.21	.60		Measured

FIGURE 11-46

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

122.15 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.44	1.40	1.44	1.44	1.42	1.34	1.21	.83
8	1.45	1.44	1.46	1.49	1.43	1.38	1.21	.86
		1.43	1.40	1.44	1.41	1.36	1.22	.89
	9	1.44	1.44	1.45	1.45	1.35	1.24	.90
			1.44	1.43	1.40	1.32	1.17	.76
		10	1.44	1.47	1.40	1.35	1.15	.78
				1.42	1.35	1.29	1.17	.65
			11	1.42	1.36	1.28	1.16	.65
					1.45	1.18	.96	
				12	1.45	1.17	.98	
						1.16	.56	Calculated
					13	1.15	.58	Measured

FIGURE 11-47

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

130.59 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.43	1.39	1.43	1.43	1.41	1.34	1.20	.83
8	1.45	1.43	1.44	1.49	1.42	1.38	1.22	.86
		1.42	1.39	1.43	1.40	1.34	1.22	.89
9		1.44	1.44	1.45	1.43	1.34	1.25	.90
			1.43	1.42	1.39	1.31	1.16	.76
10			1.45	1.47	1.39	1.35	1.16	.79
				1.41	1.35	1.29	1.16	.64
11				1.42	1.36	1.29	1.17	.66
					1.44	1.17	.96	
12					1.46	1.18	.99	
						1.15	.56	Calculated
13						1.17	.58	Measured

FIGURE 11-48

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

135.44 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.42	1.39	1.42	1.42	1.40	1.33	1.19	.82
8	1.44	1.43	1.45	1.48	1.41	1.36	1.20	.85
		1.41	1.39	1.42	1.40	1.34	1.22	.89
	9	1.43	1.42	1.43	1.43	1.34	1.23	.89
			1.42	1.42	1.38	1.31	1.16	.76
		10	1.42	1.45	1.38	1.33	1.14	.77
				1.40	1.34	1.28	1.16	.64
			11	1.40	1.34	1.27	1.15	.64
					1.44	1.17	.96	
				12	1.43	1.16	.97	
						1.15	.56	Calculated
					13	1.15	.57	Measured

FIGURE 11-49

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

139.82 EFPD 50%FP Control Bank D at 180 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.42	1.38	1.42	1.43	1.40	1.34	1.21	.84	
8	1.48	1.46	1.48	1.51	1.45	1.41	1.24	.88	
		1.41	1.39	1.42	1.41	1.35	1.24	.91	
9		1.47	1.46	1.47	1.47	1.38	1.28	.92	
			1.42	1.42	1.39	1.32	1.17	.77	
10			1.47	1.49	1.43	1.38	1.19	.80	
				1.40	1.35	1.29	1.19	.65	
11				1.44	1.41	1.32	1.20	.67	
					1.45	1.19	.98		
12					1.50	1.21	1.01		
						1.18	.57		Calculated
13						1.20	.60		Measured

FIGURE 11-50

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

141.52 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.37	1.35	1.37	1.38	1.36	1.30	1.16	.81
8	1.43	1.42	1.44	1.48	1.42	1.37	1.20	.85
		1.37	1.35	1.38	1.36	1.30	1.20	.87
	9	1.42	1.42	1.44	1.44	1.34	1.24	.90
			1.38	1.38	1.35	1.28	1.13	.75
		10	1.43	1.46	1.39	1.34	1.16	.78
				1.36	1.32	1.25	1.14	.63
			11	1.41	1.35	1.29	1.17	.66
					1.41	1.16	.94	
				12	1.45	1.18	.99	
						1.14	.56	Calculated
					13	1.17	.59	Measured

FIGURE 11-51

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

146.01 EFPD 75%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.41 1.43	1.38 1.44	1.40 1.43	1.41 1.47	1.38 1.40	1.32 1.36	1.19 1.19	.82 .85
9		1.40 1.42	1.38 1.43	1.40 1.42	1.39 1.43	1.32 1.32	1.21 1.23	.88 .89
10			1.40 1.41	1.40 1.45	1.37 1.37	1.30 1.33	1.15 1.14	.76 .78
11				1.38 1.39	1.34 1.35	1.27 1.26	1.16 1.15	.64 .64
12					1.43 1.42	1.17 1.16	.95 .97	
13						1.14 1.14	.56 .58	Calculated Measured

FIGURE 11-52

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

150.19 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.36	1.34	1.36	1.38	1.35	1.29	1.16	.81
8	1.39	1.40	1.40	1.44	1.38	1.35	1.18	.84
		1.36	1.34	1.36	1.35	1.29	1.19	.87
9		1.38	1.39	1.39	1.41	1.30	1.22	.88
			1.36	1.37	1.34	1.27	1.13	.74
10			1.39	1.43	1.35	1.32	1.13	.77
				1.35	1.31	1.24	1.14	.63
11				1.37	1.34	1.25	1.15	.64
					1.40	1.16	.94	
12					1.41	1.16	.97	
						1.13	.55	Calculated
13						1.14	.58	Measured

FIGURE 11-53

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

162.76 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.35	1.33	1.35	1.36	1.33	1.28	1.15	.80
8	1.39	1.39	1.39	1.43	1.36	1.33	1.17	.83
		1.34	1.33	1.35	1.34	1.28	1.18	.86
9		1.37	1.38	1.38	1.39	1.29	1.21	.87
			1.35	1.36	1.32	1.26	1.12	.74
10			1.37	1.42	1.34	1.31	1.12	.77
				1.34	1.30	1.23	1.13	.63
11				1.36	1.32	1.24	1.14	.64
					1.38	1.15	.93	
12					1.40	1.15	.96	
						1.12	.55	Calculated
13						1.14	.58	Measured

FIGURE 11-54

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

173.34 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.35	1.33	1.33	1.35	1.32	1.28	1.14	.80
8	1.36	1.39	1.37	1.42	1.34	1.34	1.17	.84
		1.33	1.33	1.33	1.33	1.27	1.18	.86
	9	1.35	1.38	1.36	1.39	1.28	1.22	.88
			1.34	1.35	1.31	1.26	1.11	.74
		10	1.35	1.41	1.33	1.31	1.12	.77
				1.32	1.30	1.23	1.13	.62
			11	1.34	1.33	1.24	1.14	.64
					1.38	1.15	.93	
				12	1.39	1.17	.96	
						1.12	.55	Calculated
					13	1.14	.58	Measured

FIGURE 11-55

McGuire-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

185.58 EFPD 50%FP Control Bank D at 215 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.34	1.32	1.32	1.34	1.30	1.27	1.13	.80
8	1.35	1.37	1.35	1.40	1.33	1.32	1.16	.84
		1.32	1.32	1.32	1.32	1.26	1.18	.86
	9	1.34	1.36	1.34	1.38	1.27	1.21	.87
			1.32	1.34	1.30	1.25	1.11	.74
		10	1.34	1.39	1.31	1.30	1.11	.77
				1.31	1.30	1.22	1.13	.62
			11	1.33	1.32	1.24	1.14	.64
					1.37	1.15	.93	
				12	1.39	1.17	.96	
						1.12	.55	Calculated
					13	1.13	.58	Measured

FIGURE 11-56

McGuire-1 Cy-1A Assembly Radial Powers Calculated vs Measured

198.66 EFPD 90%FP Control Bank D at 217 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.12 1.08	1.20 1.22	1.11 1.08	1.16 1.18	1.07 1.05	1.13 1.18	1.00 .99	.76 .79
9	1.20 1.21	1.12 1.08	1.19 1.20	1.09 1.05	1.15 1.18	1.07 1.04	1.14 1.17	.76 .76
10	1.10 1.06	1.18 1.19	1.09 1.06	1.16 1.19	1.08 1.06	1.13 1.15	.96 .93	.70 .71
11	1.13 1.16	1.07 1.04	1.15 1.17	1.10 1.07	1.19 1.21	1.06 1.03	1.02 1.03	.54 .54
12	1.01 .99	1.11 1.13	1.06 1.04	1.18 1.20	1.18 1.16	1.12 1.13	.82 .83	
13	.96 .98	1.00 .98	1.09 1.13	1.04 1.02	1.11 1.11	1.01 1.01	.56 .57	
14	.92 .91	1.07 1.11	.93 .90	1.00 1.00	.81 .81	.56 .57		
15	.71 .73	.72 .72	.67 .69	.53 .52	Calculated Measured			

FIGURE 11-57

McGuire-1 Cy-1A Assembly Radial Powers Calculated vs Measured

217.53 EFPD 100%FP Control Bank D at 209 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.10	1.19	1.10	1.15	1.07	1.13	1.00	.76
8	1.06	1.21	1.08	1.18	1.05	1.17	.99	.79
	1.19	1.11	1.18	1.08	1.15	1.07	1.13	.76
9	1.20	1.07	1.19	1.05	1.17	1.04	1.17	.77
	1.09	1.17	1.09	1.16	1.08	1.12	.96	.71
10	1.06	1.18	1.05	1.17	1.05	1.15	.94	.72
	1.13	1.07	1.15	1.09	1.17	1.05	1.02	.55
11	1.16	1.04	1.17	1.07	1.19	1.02	1.03	.55
	1.02	1.11	1.06	1.17	1.15	1.10	.82	
12	1.00	1.13	1.04	1.19	1.13	1.11	.83	
	.98	1.01	1.10	1.04	1.10	1.00	.57	
13	.99	.98	1.13	1.02	1.10	1.00	.58	
	.94	1.09	.94	1.01	.82	.56		
14	.93	1.12	.92	1.02	.82	.57		
	.73	.74	.69	.54	Calculated			
15	.76	.74	.71	.54	Measured			

FIGURE 11-58

McGuire-1 Cy-1A Assembly Radial Powers Calculated vs Measured

223.35 EFPD 100%FP Control Bank D at 211 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.09 1.06	1.18 1.21	1.10 1.08	1.15 1.18	1.07 1.05	1.13 1.18	1.00 .99	.77 .79
9	1.18 1.20	1.11 1.07	1.17 1.19	1.08 1.05	1.14 1.17	1.07 1.05	1.13 1.17	.77 .77
10	1.09 1.07	1.17 1.19	1.09 1.05	1.15 1.17	1.08 1.05	1.12 1.14	.96 .93	.71 .72
11	1.13 1.17	1.07 1.04	1.14 1.17	1.09 1.07	1.17 1.19	1.05 1.02	1.02 1.02	.55 .55
12	1.02 1.01	1.11 1.13	1.06 1.04	1.17 1.19	1.16 1.14	1.10 1.11	.82 .82	
13	.99 .99	1.02 .99	1.10 1.13	1.04 1.01	1.10 1.10	1.00 .99	.57 .58	
14	.95 .93	1.09 1.12	.95 .92	1.01 1.02	.82 .82	.57 .57		
15	.74 .76	.74 .74	.69 .71	.54 .54	Calculated Measured			

FIGURE 11-59

McGuire-1 Cy-1A Assembly Radial Powers Calculated vs Measured

236.23 EFPD 100%FP Control Bank D at 211 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.09 1.06	1.17 1.21	1.09 1.08	1.14 1.17	1.07 1.05	1.13 1.17	1.00 .99	.77 .80
9	1.17 1.20	1.10 1.07	1.16 1.18	1.07 1.05	1.14 1.16	1.07 1.04	1.13 1.17	.77 .77
10	1.08 1.06	1.16 1.18	1.08 1.05	1.15 1.17	1.07 1.05	1.12 1.14	.97 .94	.71 .73
11	1.13 1.16	1.06 1.04	1.14 1.16	1.08 1.06	1.17 1.18	1.05 1.02	1.02 1.02	.55 .56
12	1.03 1.01	1.11 1.13	1.06 1.04	1.16 1.18	1.15 1.13	1.10 1.10	.83 .82	
13	1.00 1.00	1.02 .99	1.10 1.13	1.04 1.01	1.10 1.10	1.00 .99	.57 .58	
14	.96 .94	1.10 1.12	.95 .93	1.01 1.02	.82 .82	.57 .58		
15	.75 .77	.75 .75	.70 .72	.55 .55	Calculated Measured			

FIGURE 11-60

McGuire-1 Cy-1A Assembly Radial Powers Calculated vs Measured

249.75 EFPD 100%FP Control Bank D at 221 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.10 1.07	1.17 1.20	1.08 1.06	1.13 1.15	1.06 1.05	1.12 1.16	1.00 .99	.77 .80
9	1.17 1.19	1.09 1.06	1.15 1.17	1.07 1.05	1.13 1.17	1.06 1.04	1.13 1.15	.77 .77
10	1.08 1.06	1.15 1.17	1.07 1.05	1.14 1.17	1.07 1.05	1.12 1.14	.97 .93	.72 .72
11	1.12 1.17	1.06 1.04	1.13 1.15	1.08 1.05	1.17 1.18	1.05 1.03	1.02 1.03	.56 .55
12	1.03 1.03	1.11 1.14	1.06 1.02	1.17 1.17	1.17 1.14	1.10 1.11	.83 .83	
13	1.01 1.00	1.03 1.00	1.10 1.14	1.05 1.02	1.10 1.09	1.00 1.00	.58 .59	
14	.96 .95	1.10 1.13	.96 .94	1.02 1.03	.83 .81	.58 .58		
15	.76 .78	.76 .75	.71 .73	.55 .56	Calculated Measured			

FIGURE 11-61

McGuire-1 Cy-1A Assembly Peak Axial Powers Calculated vs Measured

198.66 EFPD 90%FP Control Bank D at 217 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.34	1.41	1.30	1.35	1.25	1.29	1.13	.86
8	1.34	1.51	1.33	1.44	1.27	1.41	1.19	.94
	1.41	1.32	1.38	1.27	1.33	1.23	1.30	.86
9	1.50	1.33	1.47	1.29	1.42	1.25	1.40	.91
	1.29	1.37	1.28	1.35	1.26	1.29	1.09	.79
10	1.31	1.46	1.30	1.45	1.29	1.39	1.11	.86
	1.32	1.25	1.34	1.28	1.37	1.21	1.17	.61
11	1.43	1.27	1.43	1.32	1.47	1.25	1.24	.65
	1.18	1.28	1.23	1.36	1.37	1.28	.94	
12	1.21	1.37	1.26	1.45	1.41	1.36	1.00	
	1.11	1.15	1.25	1.19	1.27	1.15	.64	
13	1.19	1.17	1.36	1.22	1.34	1.21	.69	
	1.04	1.22	1.05	1.14	.92	.64		
14	1.08	1.32	1.08	1.21	.98	.69		
	.81	.82	.76	.60	Calculated			
15	.87	.86	.83	.63	Measured			

FIGURE 11-62

McGuire-1 Cy-1A Assembly Peak Axial Powers Calculated vs Measured

217.53 EFPD 100%FP Control Bank D at 209 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.33 1.31	1.43 1.48	1.30 1.33	1.37 1.42	1.26 1.26	1.34 1.39	1.18 1.18	.90 .94
9	1.42 1.47	1.31 1.31	1.40 1.44	1.27 1.27	1.36 1.40	1.26 1.24	1.36 1.38	.90 .91
10	1.29 1.30	1.39 1.44	1.28 1.27	1.37 1.42	1.27 1.26	1.33 1.36	1.13 1.11	.83 .86
11	1.34 1.41	1.25 1.26	1.36 1.41	1.28 1.29	1.41 1.44	1.24 1.21	1.21 1.22	.64 .65
12	1.19 1.21	1.31 1.36	1.25 1.25	1.40 1.43	1.41 1.36	1.32 1.31	.97 .98	
13	1.14 1.19	1.18 1.17	1.30 1.34	1.23 1.21	1.32 1.31	1.19 1.18	.67 .68	
14	1.10 1.10	1.29 1.32	1.10 1.09	1.19 1.21	.97 .97	.67 .68		
15	.86 .90	.87 .87	.81 .84	.63 .64	Calculated Measured			

FIGURE 11-63

McGuire-1 Cy-1A Assembly Peak Axial Powers Calculated vs Measured

223.35 EFPD 100%FP Control Bank D at 211 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.32 1.33	1.41 1.49	1.29 1.32	1.36 1.43	1.25 1.28	1.33 1.45	1.17 1.21	.90 .97
9	1.41 1.48	1.30 1.31	1.39 1.45	1.26 1.28	1.35 1.43	1.25 1.28	1.35 1.42	.90 .93
10	1.28 1.30	1.38 1.45	1.27 1.28	1.36 1.43	1.26 1.28	1.33 1.40	1.13 1.13	.83 .87
11	1.33 1.42	1.25 1.27	1.35 1.42	1.28 1.30	1.40 1.47	1.24 1.25	1.21 1.23	.64 .66
12	1.19 1.21	1.31 1.36	1.24 1.26	1.39 1.45	1.40 1.43	1.32 1.36	.97 .99	
13	1.14 1.19	1.18 1.19	1.29 1.37	1.22 1.24	1.31 1.35	1.19 1.21	.67 .69	
14	1.10 1.12	1.29 1.35	1.10 1.11	1.19 1.23	.96 .98	.66 .69		
15	.87 .91	.87 .88	.81 .85	.63 .65	Calculated Measured			

FIGURE 11-64

McGuire-1 Cy-1A Assembly Peak Axial Powers Calculated vs Measured

236.23 EFPD 100%FP Control Bank D at 211 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.29 1.32	1.39 1.47	1.27 1.31	1.34 1.42	1.24 1.27	1.32 1.43	1.17 1.20	.90 .96
9	1.39 1.45	1.28 1.30	1.37 1.43	1.25 1.27	1.33 1.41	1.24 1.27	1.34 1.41	.90 .93
10	1.26 1.28	1.36 1.42	1.26 1.27	1.34 1.41	1.25 1.27	1.32 1.38	1.12 1.12	.83 .87
11	1.32 1.41	1.23 1.26	1.34 1.40	1.26 1.29	1.38 1.45	1.23 1.24	1.20 1.22	.64 .66
12	1.18 1.21	1.30 1.36	1.24 1.25	1.38 1.44	1.38 1.41	1.31 1.34	.97 .98	
13	1.15 1.19	1.18 1.19	1.29 1.36	1.22 1.23	1.30 1.34	1.18 1.20	.67 .69	
14	1.11 1.12	1.30 1.34	1.10 1.11	1.19 1.22	.96 .98	.67 .69		
15	.88 .91	.88 .89	.82 .86	.64 .65	Calculated Measured			

FIGURE 11-65

McGuire-1 Cy-1A Assembly Peak Axial Powers Calculated vs Measured

249.75 EFPD 100%FP Control Bank D at 221 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.26 1.26	1.35 1.40	1.24 1.24	1.31 1.36	1.21 1.25	1.29 1.38	1.15 1.18	.89 .95
9	1.35 1.41	1.25 1.26	1.33 1.39	1.22 1.24	1.30 1.39	1.22 1.23	1.32 1.37	.89 .91
10	1.23 1.25	1.33 1.39	1.23 1.25	1.31 1.38	1.22 1.25	1.29 1.35	1.11 1.09	.83 .85
11	1.29 1.39	1.21 1.24	1.31 1.36	1.23 1.24	1.35 1.41	1.20 1.22	1.18 1.20	.64 .65
12	1.17 1.22	1.28 1.34	1.21 1.20	1.35 1.38	1.35 1.37	1.28 1.32	.95 .97	
13	1.15 1.18	1.17 1.18	1.27 1.35	1.20 1.22	1.28 1.30	1.16 1.17	.66 .68	
14	1.10 1.12	1.29 1.34	1.09 1.12	1.18 1.22	.95 .96	.66 .67		
15	.88 .91	.88 .88	.82 .86	.64 .66	Calculated Measured			

FIGURE 11-66

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

71.82 EFPD 100%FP Control Bank D at 200 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.14	1.09	1.19	1.12	1.16	1.05	1.01	.69	
8	1.12	1.05	1.17	1.11	1.15	1.05	1.01	.71	
		1.18	1.12	1.20	1.16	1.14	1.00	.75	
9		1.16	1.11	1.19	1.16	1.13	1.01	.77	
			1.20	1.13	1.19	1.09	.98	.64	
10			1.18	1.12	1.18	1.09	.98	.66	
				1.19	1.13	1.10	.93	.53	
11				1.18	1.13	1.08	.92	.56	
					1.10	.98	.82		
12					1.09	.99	.86		
						.98	.48		Calculated
13						1.02	.51		Measured

FIGURE 11-67

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

101.62 EFPD 100%FP Control Bank D at 218 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.17 1.14	1.11 1.06	1.20 1.16	1.15 1.13	1.18 1.17	1.06 1.06	1.00 1.00	.69 .71
9		1.19 1.16	1.15 1.12	1.21 1.18	1.17 1.16	1.13 1.12	1.00 1.01	.74 .76
10			1.20 1.18	1.15 1.13	1.18 1.17	1.09 1.09	.97 .97	.63 .65
11				1.19 1.17	1.14 1.13	1.08 1.08	.92 .91	.53 .55
12					1.11 1.11	.99 1.00	.81 .85	
13						.97 1.02	.48 .51	Calculated Measured

FIGURE 11-68

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

133.30 EFPD 100%FP Control Bank D at 216 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.17	1.13	1.20	1.17	1.18	1.08	.99	.68
8	1.14	1.08	1.17	1.14	1.16	1.07	.99	.70
		1.19	1.17	1.20	1.18	1.12	1.00	.73
9		1.17	1.14	1.19	1.17	1.12	1.01	.76
			1.21	1.16	1.17	1.09	.96	.63
10			1.18	1.14	1.17	1.09	.96	.65
				1.18	1.14	1.07	.91	.53
11				1.16	1.13	1.06	.91	.55
					1.09	.99	.80	
12					1.09	1.00	.84	
						.96	.47	Calculated
13						1.01	.51	Measured

FIGURE 11-69

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

166.04 EFPD 100%FP Control Bank D at 210 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.15	1.13	1.20	1.18	1.16	1.09	.99	.69
8	1.13	1.09	1.17	1.15	1.14	1.08	.99	.71
		1.19	1.18	1.20	1.18	1.12	1.00	.74
9		1.16	1.15	1.19	1.17	1.11	1.01	.76
			1.20	1.17	1.17	1.10	.96	.63
10			1.18	1.15	1.16	1.09	.96	.66
				1.18	1.14	1.06	.91	.53
11				1.16	1.13	1.06	.91	.55
					1.08	.99	.80	
12					1.08	1.00	.84	
						.96	.48	Calculated
13						1.00	.51	Measured

FIGURE 11-70

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

231.70 EFPD 100%FP Control Bank D at 216 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.12	1.11	1.17	1.18	1.15	1.10	.99	.71	
8	1.10	1.08	1.14	1.16	1.13	1.09	.99	.72	
		1.16	1.18	1.17	1.17	1.11	1.01	.75	
9		1.13	1.16	1.16	1.17	1.09	1.02	.76	
			1.18	1.18	1.15	1.10	.96	.65	
10			1.16	1.16	1.14	1.10	.96	.67	
				1.16	1.14	1.06	.92	.54	
11				1.14	1.14	1.05	.92	.56	
					1.08	1.01	.81		
12					1.07	1.02	.84		
						.96	.49		Calculated
13						1.00	.53		Measured

FIGURE 11-71

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

292.04 EFPD 100%FP Control Bank D at 216 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.08	1.09	1.14	1.17	1.13	1.11	1.00	.73	
8	1.07	1.07	1.12	1.15	1.11	1.10	.99	.74	
		1.13	1.17	1.15	1.16	1.10	1.03	.76	
9		1.11	1.15	1.14	1.16	1.08	1.03	.78	
			1.15	1.17	1.13	1.11	.97	.67	
10			1.14	1.16	1.12	1.11	.97	.69	
				1.14	1.14	1.06	.94	.56	
11				1.12	1.13	1.05	.93	.58	
					1.07	1.03	.82		
12					1.06	1.04	.85		
						.97	.50		Calculated
13						1.00	.55		Measured

FIGURE 11-72

Sequoyah-1 Cy-1 Assembly Radial Powers Calculated vs Measured

378.92 EFPD 100%FP Control Bank D at 222 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.06	1.06	1.10	1.14	1.11	1.12	1.01	.76	
8	1.02	1.04	1.08	1.14	1.09	1.10	1.00	.77	
		1.09	1.14	1.12	1.14	1.09	1.04	.79	
	9	1.07	1.13	1.10	1.14	1.07	1.05	.80	
			1.12	1.14	1.11	1.11	.99	.71	
		10	1.10	1.14	1.09	1.11	.98	.73	
				1.11	1.13	1.06	.96	.59	
			11	1.09	1.13	1.05	.96	.60	
					1.08	1.05	.84		
				12	1.06	1.06	.87		
						.98	.53		Calculated
					13	1.02	.58		Measured

FIGURE 11-73

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

71.82 EFPD 100%FP Control Bank D at 200 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.48 1.55	1.40 1.42	1.51 1.57	1.45 1.50	1.51 1.60	1.35 1.42	1.26 1.33	.87 .93
9		1.50 1.56	1.44 1.48	1.53 1.59	1.48 1.56	1.44 1.51	1.27 1.34	.94 1.02
10			1.52 1.57	1.44 1.49	1.50 1.57	1.38 1.45	1.23 1.29	.80 .86
11				1.51 1.57	1.44 1.51	1.38 1.43	1.17 1.21	.67 .74
12					1.41 1.50	1.26 1.32	1.03 1.13	
13						1.25 1.36	.60 .67	Calculated Measured

FIGURE 11-74

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

101.62 EFPD 100%FP Control Bank D at 218 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.44	1.37	1.46	1.41	1.45	1.30	1.20	.83
8	1.46	1.35	1.48	1.44	1.50	1.36	1.26	.88
		1.45	1.41	1.47	1.43	1.37	1.21	.89
9		1.47	1.42	1.50	1.47	1.42	1.27	.96
			1.47	1.41	1.43	1.32	1.16	.76
10			1.48	1.42	1.48	1.37	1.22	.82
				1.44	1.39	1.31	1.11	.64
11				1.48	1.43	1.35	1.14	.69
					1.35	1.21	.98	
12					1.40	1.25	1.06	
						1.18	.57	Calculated
13						1.27	.64	Measured

FIGURE 11-75

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

133.30 EFPD 100%FP Control Bank D at 216 Steps Withdrawn

	H	G	F	E	D	C	B	A
8	1.40 1.42	1.35 1.33	1.43 1.44	1.40 1.41	1.41 1.45	1.29 1.33	1.17 1.21	.81 .85
9		1.42 1.43	1.40 1.40	1.43 1.46	1.40 1.44	1.32 1.37	1.18 1.23	.87 .92
10			1.43 1.44	1.39 1.39	1.39 1.42	1.30 1.33	1.13 1.17	.74 .79
11				1.40 1.42	1.36 1.38	1.26 1.29	1.08 1.10	.62 .66
12					1.30 1.34	1.18 1.22	.95 1.03	
13						1.14 1.23	.56 .62	Calculated Measured

FIGURE 11-76

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

166.04 EFPD 100%FP Control Bank D at 210 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.37	1.33	1.40	1.39	1.38	1.28	1.15	.81	
8	1.37	1.29	1.39	1.38	1.39	1.29	1.17	.84	
		1.39	1.40	1.40	1.39	1.30	1.17	.86	
9		1.38	1.37	1.42	1.40	1.32	1.20	.90	
			1.40	1.38	1.36	1.28	1.11	.74	
10			1.40	1.36	1.38	1.30	1.13	.78	
				1.37	1.34	1.24	1.06	.62	
11				1.36	1.34	1.25	1.08	.66	
					1.28	1.17	.93		
12					1.30	1.19	1.00		
						1.12	.56		Calculated
13						1.19	.61		Measured

FIGURE 11-77

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

231.70 EFPD 100%FP Control Bank D at 216 Steps Withdrawn

	H	G	F	E	D	C	B	A	
	1.27	1.26	1.32	1.35	1.31	1.26	1.12	.80	
8	1.28	1.24	1.31	1.34	1.31	1.26	1.13	.83	
		1.31	1.35	1.33	1.34	1.25	1.15	.85	
	9	1.30	1.33	1.33	1.35	1.26	1.17	.88	
			1.33	1.34	1.29	1.26	1.08	.74	
		10	1.33	1.33	1.30	1.27	1.10	.77	
				1.30	1.30	1.19	1.05	.62	
			11	1.30	1.30	1.21	1.06	.65	
					1.23	1.16	.92		
				12	1.25	1.18	.98		
						1.10	.56		Calculated
					13	1.15	.62		Measured

FIGURE 11-78

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

292.04 EFPD 100%FP Control Bank D at 216 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.23	1.22	1.28	1.32	1.27	1.26	1.12	.82
8	1.24	1.23	1.28	1.33	1.29	1.26	1.14	.85
		1.26	1.31	1.29	1.31	1.23	1.16	.87
9		1.28	1.32	1.31	1.34	1.24	1.18	.89
			1.29	1.31	1.26	1.25	1.08	.76
10			1.31	1.32	1.28	1.27	1.11	.79
				1.27	1.28	1.18	1.06	.64
11				1.28	1.30	1.20	1.07	.66
					1.21	1.17	.92	
12					1.24	1.19	.97	
						1.10	.57	Calculated
13						1.15	.63	Measured

FIGURE 11-79

Sequoyah-1 Cy-1 Assembly Peak Axial Powers Calculated vs Measured

378.92 EFPD 100%FP Control Bank D at 222 Steps Withdrawn

	H	G	F	E	D	C	B	A
	1.19	1.20	1.24	1.30	1.26	1.27	1.15	.86
8	1.22	1.22	1.27	1.34	1.28	1.29	1.17	.90
		1.23	1.29	1.26	1.30	1.23	1.19	.89
9		1.26	1.33	1.30	1.34	1.25	1.22	.93
			1.26	1.30	1.25	1.26	1.11	.80
10			1.29	1.33	1.28	1.30	1.14	.84
				1.26	1.28	1.20	1.09	.67
11				1.28	1.32	1.23	1.11	.69
					1.22	1.19	.94	
12					1.26	1.24	1.00	
						1.11	.59	Calculated
13						1.19	.67	Measured

FIGURE 11-80

McGuire-1 Cy-1 PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 52.2 EFPD vs Core Meas - 48.8 EFPD

	H	G	F	E	D	C	B	A
8	1.16 1.12	1.10 1.09	1.20 1.19	1.20 1.21	1.23 1.23	1.12 1.12	1.01 1.01	.68 .71
9		1.18 1.15	1.12 1.13	1.23 1.22	1.18 1.19	1.15 1.14	1.00 1.02	.74 .75
10			1.22 1.20	1.19 1.19	1.20 1.18	1.09 1.10	.97 .97	.63 .64
11				1.20 1.19	1.08 1.08	1.08 1.07	.94 .96	.53 .54
12					1.22 1.18	.92 .92	.80 .81	
13						.96 .95	.46 .46	Calculated Measured

FIGURE 11-81

McGuire-1 Cy-1 PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 104.4 EFPD vs Core Meas - 104.5 EFPD

	H	G	F	E	D	C	B	A	
	1.17	1.13	1.19	1.20	1.20	1.12	1.00	.69	
8	1.15	1.15	1.19	1.23	1.19	1.12	1.00	.70	
		1.18	1.14	1.21	1.18	1.13	1.01	.74	
9		1.17	1.16	1.20	1.19	1.12	1.02	.74	
			1.20	1.19	1.18	1.09	.96	.64	
10			1.18	1.20	1.16	1.11	.96	.64	
				1.19	1.10	1.07	.94	.54	
11				1.17	1.11	1.07	.96	.53	
					1.21	.94	.80		
12					1.18	.94	.80		
						.96	.47		Calculated
13						.94	.47		Measured

FIGURE 11-82

McGuire-1 Cy-1 PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 156.7 EFPD vs Core Meas - 150.2 EFPD

	H	G	F	E	D	C	B	A
	1.17	1.15	1.18	1.19	1.17	1.12	1.00	.70
8	1.15	1.17	1.17	1.21	1.17	1.13	1.00	.71
		1.17	1.15	1.18	1.17	1.12	1.02	.75
9		1.15	1.17	1.17	1.19	1.10	1.03	.74
			1.18	1.18	1.16	1.10	.96	.65
10			1.16	1.20	1.14	1.11	.95	.65
				1.17	1.11	1.07	.95	.55
11				1.16	1.12	1.06	.96	.54
					1.20	.96	.81	
12					1.17	.97	.81	
						.96	.49	Calculated
13						.95	.48	Measured

FIGURE 11-83

McGuire-1 Cy-1A PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 208.9 EFPD vs Core Meas - 198.7 EFPD

	H	G	F	E	D	C	B	A
8	1.10 1.08	1.22 1.23	1.08 1.09	1.17 1.19	1.05 1.06	1.16 1.18	.99 .99	.78 .79
9	1.22 1.22	1.09 1.08	1.20 1.20	1.06 1.06	1.17 1.18	1.05 1.05	1.18 1.17	.76 .76
10	1.07 1.07	1.19 1.19	1.06 1.06	1.18 1.19	1.06 1.06	1.16 1.16	.95 .93	.72 .72
11	1.14 1.17	1.04 1.04	1.17 1.18	1.08 1.08	1.22 1.22	1.04 1.04	1.03 1.03	.54 .54
12	.98 .99	1.12 1.13	1.04 1.04	1.21 1.21	1.20 1.17	1.15 1.13	.83 .83	
13	.96 .99	.98 .98	1.12 1.14	1.03 1.02	1.14 1.12	1.02 1.01	.58 .58	
14	.91 .91	1.11 1.11	.91 .91	1.01 1.01	.82 .82	.58 .57		
15	.73 .74	.72 .72	.69 .70	.53 .53	Calculated Measured			

FIGURE 11-84

Sequoyah-1 Cy-1 PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 103.6 EFPD vs Core Meas - 101.6 EFPD

	H	G	F	E	D	C	B	A
	1.16	1.08	1.17	1.13	1.18	1.08	1.01	.71
8	1.14	1.07	1.17	1.13	1.17	1.07	1.00	.71
		1.17	1.12	1.19	1.17	1.14	1.02	.77
9		1.16	1.12	1.19	1.17	1.13	1.01	.77
			1.18	1.13	1.18	1.10	.97	.65
10			1.18	1.13	1.18	1.10	.98	.66
				1.18	1.15	1.09	.91	.55
11				1.18	1.14	1.08	.92	.56
					1.13	1.00	.84	
12					1.11	1.00	.85	
						1.01	.50	Calculated
13						1.02	.51	Measured

FIGURE 11-85

Sequoyah-1 Cy-1 PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 155.5 EFPD vs Core Meas - 133.3 EFPD

	H	G	F	E	D	C	B	A	
	1.17	1.11	1.18	1.16	1.17	1.09	1.00	.71	
8	1.13	1.09	1.18	1.16	1.15	1.08	.99	.71	
		1.18	1.16	1.19	1.18	1.12	1.02	.76	
9		1.17	1.16	1.19	1.18	1.12	1.01	.77	
			1.19	1.16	1.17	1.10	.96	.65	
10			1.19	1.16	1.17	1.10	.96	.66	
				1.17	1.15	1.07	.90	.55	
11				1.16	1.14	1.06	.92	.56	
					1.11	1.00	.82		
12					1.08	1.00	.84		
						.99	.50		Calculated
13						1.00	.52		Measured

FIGURE 11-86

Sequoyah-1 Cy-1 PDQ07 Assembly Radial Powers Calculated vs Measured

PDQ07 - 362.7 EFPD vs Core Meas - 378.9 EFPD

	H	G	F	E	D	C	B	A
	1.04	1.04	1.09	1.15	1.11	1.13	1.01	.77
8	1.03	1.04	1.09	1.14	1.09	1.10	1.01	.78
		1.07	1.14	1.11	1.16	1.09	1.06	.80
9		1.07	1.14	1.11	1.15	1.08	1.05	.81
			1.11	1.15	1.11	1.12	.98	.72
10			1.11	1.14	1.10	1.12	.99	.73
				1.11	1.15	1.06	.95	.60
11				1.10	1.13	1.05	.96	.61
					1.08	1.05	.85	
12					1.07	1.07	.87	
						1.01	.57	Calculated
13						1.03	.59	Measured

12. REFERENCES

1. Nuclear Associates International Corp., "Advanced Recycle Methodology Program System Documentation", CCM-3, (EPRI Confidential), September, 1977.

2. Studsvik Energiteknik AB, "CASMO-2 A Fuel Assembly Burnup Program," Studsvik/NR-81/3, 1981.

3. Duke Power Company, "Oconee Nuclear Station Reload Design Methodology," NFS-1001, Rev. 4, June 1981.

4. Bettis Atomic Power Laboratory, C. J. Pfeiffer, "PDQ-7 Reference Manual II," WAPD-TM-947(L), February 1971.

5. Rothleder, B. M., Fisher, J. R., "EPRI-NODE-P," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 14, September 1977.

6. Not Used.

7. Not Used.

8. Cobb, W. R., Eich, W. J., Tivel, D. E., "EPRI-CELL Code Description," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 5, October 1978.

9. Edenius, M., Ekberg, K., Haggblom, H., "CASMO - THE DATA LIBRARY," Studsvik/K2-81/491, 1981.

10. Cobb, W. R., Tivel, D. E., "EPRI-CELL: GAM-THERMOS Library Descriptions," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 2, April 1976.

11. Rothleder, B. M., Poetschat, G. R., "NUPUNCHER Code Description," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 8, October 1975.

12. Duke Power Company, "MULTIFIT User Documentation," (Proprietary), February 1983.

13. Hebert, M. J., et. al., "PROGRAM C-HA-R-T CASMO to HARMONY Tableset Conversion Processor," YAEC-1313P, May 1982.

14. Rothleder, B. M. et. al., "PWR Core Modeling Procedures for Advanced Recycle Methodology Program," RP-976-1, August 1979.

15. Rothleder, B. M., Poetschat, G. R., "EPRI-FIT Code Description," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 10, October 14 1975.

16. Rothleder, B. M., Poetschat, G. R., "SUPERLINK-P Code Description," EPRI-ARMP System Documentation, CCM-3, Part II, Chapter 12, October 22 1975.

17. Smith, M. L., "PDQ7V2P7," (Proprietary), Virginia Electric and Power Company, December 1977.

18. McGuire Nuclear Station, Units 1 and 2, Updated Final Safety Analysis Report, Docket Nos. 50-369, 370.

19. Catawba Nuclear Station, Units 1 and 2, Updated Final Safety Analysis Report, Docket Nos. 50-413, 414.

20. Letter, W. O. Parker to H. R. Denton, "Oconee Reload Design Methodology Topical Report," Question 3, Docket Nos. 50-269,-270,-287, November 13 1980.

21. Duke Power Company, Quality Assurance Program Topical Report, Revision 26, September 13, 2000.

22. Not Used.

23. Not Used.

24. Not Used.

25. D. B. Owen, "Factors For One-Sided Tolerance Limits And For Variables Sampling Plans," SCR-607, Sandia Corporation Monograph, March 1963.

26. Shanstrom, R. T., et al, "CORE Codes for Operating Reactor Evaluation", SNA1617 (Proprietary), Shanstrom Nuclear Associates, April 1982.

27. American National Standards Institute, Inc., "Assessment of the Assumption of Normality (Employing Individual Observed Values)", ANSI N15.15-1974, 1974.

28. "Duke Power Company, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Rev. 1, SER dated April 26, 1996.

29. "Duke Power Company, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011PA, March 1990.

30. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology", DPC-NE-3001PA, November 1991.

31. "Duke Power Company, Westinghouse Fuel Transition Report", DPC-NE-2009-P-A, SER dated September 22, 1999.

32. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Nuclear Physics Methodology for Reload Design", DPC-NF-2010-PA, NRC Approved SER, March 13, 1985.

APPENDIX A

Code Summary

CASMO-2

CASMO-2 is a multi-group two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies. This code has been developed by Studsvik Energiteknik AB and supported by EPRI.

CHART

CHART prepares cross section tables in HARMONY format from cross section data produced by CASMO-2. CHART reduces significantly the tedious task of hand transferring values from CASMO-2 printout to macroscopic and microscopic tables in card image HARMONY format. Two, three, and four group cross section data may be obtained with one-dimensional HARMONY interpolating tables.

CORE

CORE (Codes for Operating Reactor Evaluation), is a package of computer routines for the off-line evaluation of reactor performance. CORE uses as input: detailed reactor physics data, isotopics, and thermal-hydraulics data. Calculated values are: F_Q, $F^N_{\Delta H}$, assembly burnups, isotopics, reactivity, and core thermal-hydraulics information.

DELAY

DELAY calculates core averaged delayed neutron fractions for six energy groups, core averaged decay constants for six energy groups, core averaged delayed neutron fraction with and without importance factor, estimated prompt neutron lifetime, and reactivity versus period. Input consists primarily of isotopic fission fractions versus burnup and enrichment from PDQ07 calculations.

EPRI-CELL

EPRI-CELL computes the space, energy and burnup dependence of the neutron spectrum within cylindrical cells of Light Water Reactor fuel rods. Its primary output consists of broad group, microscopic, exposure dependent cross sections for subsequent use in multidimensional diffusion theory depletion analysis. EPRI-CELL utilizes three industry accepted subcodes; GAM-1, THERMOS, and CINDER.

EPRI-CPM

EPRI-CPM is a multi-group two-dimensional collision probability code for burnup calculations on BWR and PWR assemblies. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with gadolinium, burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, boron steel curtains and cruciform control rods in the regions separating fuel assemblies.

EPRI-FIT

EPRI-FIT is a program which processes the PDQ07 integral file and calculates and edits values needed by the EPRI-NODE code. EPRI-FIT greatly reduces the hand calculation time needed to extract these values from the PDQ07 printout and improves the quality assurance. A data file under the local name of COLOR is written which contains the EPRI-FIT edited data and is used as input to the SUPERLINK program.

EPRI-NODE

EPRI-NODE is a multi-dimensional nodal code derived from FLARE. The EPRI-NODE program computes the core effective multiplication factor, the three-dimensional core power distribution, core coolant flow and temperature distribution, and fuel exposure distribution. The program includes the effects of partially inserted full-length control rods, part-length rods, and up to 13 different fuel assembly types with different enrichments and burnable absorber shim loadings. EPRI-NODE has a capacity to represent the core with 32 axial nodes for each fuel assembly and 30x30 nodes in the XY plane.

The program iterates to account for the interaction between power distribution and core nuclear properties which depend on coolant flow and coolant temperature distributions, fuel temperature distribution and xenon distribution. The program computes the time dependence of xenon following changes in power level and/or changes in power distribution. The program permits fuel shuffling from one location to another and fresh fuel insertion for burnup cycle calculations. Individual steps can by stacked for either xenon transient or fuel cycle burnup calculations. See Reference 5.

EPRI-NUPUNCHER

NUPUNCHER prepares cross section tables in HARMONY format from cross section data produced by EPRI-CELL and placed on the ECDATA file. NUPUNCHER reduces significantly the tedious task of hand transferring values from the EPRI-CELL printout to macroscopic and microscopic tables in card image HARMONY format. Two, three and four group cross section data may be obtained with one-dimensional HARMONY interpolating tables.

EPRI-PDQ07 MODIFICATIONS

PDQ07 is an industry accepted multi-group one, two, or three-dimensional diffusion depletion code. EPRI-ARMP uses PDQ07/Version II with minor modifications to allow options for improved removal treatment, peak power editing, and re-editing.

EPRI -SHUFFLE

The EPRI-SHUFFLE program will read a PDQ07 concentration file, make certain modifications to this file, and write a new updated concentration file. This procedure is accomplished by defining "assembly regions" in the program input. Assembly regions are square arrays of mesh points containing depletable nuclide concentrations and superimposed on the original PDQ07 geometry. These assembly regions are then used to describe the movement of existing nuclide concentrations by translation, reflection and/or rotation. In addition, new fuel concentrations can replace spent fuel concentrations in selected assembly regions described in the program's input.

EPRI - SUPERLINK

SUPERLINK accesses data on the files produced by EPRI-FIT and, together with relevant input information for file management and for data processing control, produces polynomial coefficients for use in EPRI-NODE.

MULTIFIT

MULTIFIT reads EPRI-CELL cross section files and generates HARMONY cross sections and g-factors. Both HARMONY masks and function tables can include the effects of up to three independent variables. MULTIFIT can perform almost all of the functions of EPRI-NUPUNCHER.

PDQ07

See EPRI-PDQ07 Modifications and Reference 4.

CASMO-3

CASMO-3 is a multi-group two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies. This code develops cross-section data for use in SIMULATE-3. A full description of this code is contained in Reference 28.

SIMULATE-3P

SIMULATE-3 is a three-dimensional, two-group diffusion theory reactor simulator used for nuclear design calculations. A full description of this code is contained in Reference 28.

APPENDIX B

NRC/DPC Correspondence Regarding NRC Request for Additional Information

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

November 5, 1984

Docket Nos: 50-369, 50-370
and 50-413, 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Request for Additional Information Regarding Topical Report on Physics Methodology for Reloads: McGuire and Catawba Nuclear Station

In response to your letter of July 18, 1984, the NRC staff, with the technical assistance of Brookhaven National Laboratory (BNL), is reviewing Duke Power Company topical report DPC-NF-2010 which describes the nuclear physics methodology for reload design at the McGuire and Catawba Nuclear Stations. We find that additional information identified in the enclosure is needed to complete this review.

A reply at your earliest opportunity and no later than November 30, 1984, is needed for the staff to meet your requested review completion date of January 1985. A copy of your reply should also be forwarded directly to BNL at the address below.

Should you have questions or need to meet with the staff regarding the enclosure, contact Darl S. Hood at (301) 492-8408.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosure:
As stated

cc: Dr. John Carew
Building 475 B
Brookhaven National Laboratory
Upton, Long Island, N.Y. 11973

See next page

CATAWBA

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

cc: William L. Porter, Esq.
Duke Power Company
P.O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

North Carolina MPA-1
P.O. Box 95162
Raleigh, North Carolina 27625

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, Pennsylvania 15230

NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

Mr. Jesse L. Riley, President
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28208

Richard P. Wilson, Esq.
Assistant Attorney General
S.C. Attorney General's Office
P.O. Box 11549
Columbia, South Carolina 29211

North Carolina Electric Membership Corp.
3333 North Boulevard
P.O. Box 27306
Raleigh, North Carolina 27611

Saluda River Electric Cooperative, Inc.
P.O. Box 929
Laurens, South Carolina 29360

Senior Resident Inspector
Route 2, Box 179N
York, South Carolina 29745

James P. O'Reilly, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Robert Guild, Esq.
P.O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135 ½ Devine Street
Columbia, South Carolina 29205

Karen E. Long
Assistant Attorney General
N.C. Department of Justice
P.O. Box 629
Raleigh, North Carolina 27602

cc: Spence Perry, Esquire
Associate General Counsel
Federal Emergency Management Agency
Room 840
500 C Street, S.W.
Washington, D. C. 20472

Mark S. Calvert, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

Mr. Michael Hirsch
Federal Emergency Management Agency
Office of the General Counsel
Room 840
500 C Street, S.W.
Washington, DC 20472

Brian P. Cassidy, Regional Counsel
Federal Emergency Management Agency,
Region I
J. W. McCormach POCH
Boston, Massachusetts 02109

McGuire

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

cc: Mr. A. Carr
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. Robert Gill
Duke Power Company
Nuclear Production Department
P. O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

Mr. Wm. Orders
Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
Route 4, Box 529
Hunterville, North Carolina 28078

James P. O'Reilly, Regional Administrator
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

R. S. Howard
Operating Plants Projects
Regional Manager
Westinghouse Electric Corporation - R&D 701
P. O. Box 2728
Pittsburgh, Pennsylvania 15230

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION ON DUKE POWER COMPANY
TOPICAL REPORT DPC-NF-2010

---

1. Please provide additional information regarding the NUC-MARGINS code and its use in the Dropped Rod Analysis. Provide short descriptions of the input, output, calculational models used, benchmark calculations performed and the conservatisms assumed in the analysis.

2. Identify the nominal and various off-nominal cross-section sets that are generated in order to evaluate the different reactivity coefficients and defects.

3. Provide a short description of the PDQEDIT code and describe the verification program that was undertaken to test data generated with PDQEDIT for use in SNA-CORE.

4. Comment on the reasons for the 3.1% non-conservative bias in the calculated peak axial powers (Section 11.5.4). Describe the model refinements, if any, that have been undertaken to reduce this bias.

5. Duke Power Company's contention that no uncertainty in calculated pin powers needs to be accounted for has not been adequately established. One possible way to establish the uncertainty is to perform a standard problem. A standard problem recently developed at Brookhaven National Laboratory for a licensee to assess its ability to calculate typical PWR fuel assemblies, is attached. A solution of this problem or other justification for the assumed uncertainty should be provided.

6. Please provide the updates to DPC-NF-2010, if any, that will make it consistent with the methodologies currently being used by Duke Power.

FUEL ASSEMBLY STANDARD PROBLEM

The standard problem is to be calculated in two dimensions in an iterated-source mode using reflecting boundary conditions in the horizontal plane neglecting axial leakage. The following series of assembly depletion and reactivity defect calculations are to be calculated.

I. DEPLETION CALCULATIONS

Provide the following edited quantities for an assembly with and without burnable poison rods at BOL, 500, 5000, 10000, 20000, 30000 and 40000 Mwd/MT*:

1. Relative pin powers

2. Assembly volume averaged fuel pellet isotopics; U^{235}, U^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242} and calculated fission product densities [atom/barn-cm]

3. Assembly total reaction rates (A-absorption, F-fission)

   a. Fuel

U^{235} (A)	Pu^{240} (A)
U^{235} (F)	Pu^{240} (F)
U^{238} (A)	Pu^{241} (A)
U^{238} (F)	Pu^{241} (F)
Pu^{239} (A)	Pu^{242} (A)
Pu^{239} (F)	Pu^{242} (F)

   b. Clad (A)
   c. Burnable Poison (A)
   d. Water (A)
   e. Control Rod (A)

4. Assembly Characteristics

   a. k_∞ - Infinite Multiplication Factor
   b. M^2 - Migration Area [cm^2]
   c. B_M^2 - Material Buckling [cm^{-2}]
   d. β - Delayed Neutron Fraction
   e. Two-Group Inverse Neutron Velocity† [cm/sec]

5. Two-Group Collapsed Assembly Averaged Cross Sections†

   D [cm], Σ_a[cm^{-1}], Σ_r[cm^{-1}],

   $\nu\Sigma_f$[cm^{-1}], $\kappa\Sigma_f$[watt/cm], Σ_f[cm^{-1}]

* These are editing points and do not necessarily correspond to the depletion steps.

† Thermal breakpoint assumed at 0.625 [eV]

FUEL ASSEMBLY STANDARD PROBLEM

II. REACTIVITY DEFECT CALCULATIONS

Provide the following reactivity defects (%Δ k/k) for an assembly with and without burnable poison rods at BOL and EOL (30,000 Mwd/MT):

REACTIVITY DEFECT (%Δ k/k)*	UNPERTURBED CASE†	PERTURBED CASE
1. Fuel Temperature (T_{fuel})	T^{base}_{fuel}	$T^{base}_{moderator}$
2. Moderator Temperature ($T_{moderator}$)	$T^{base}_{moderator}$	$T^{base}_{moderator}$ -25°K
3. Moderator & Fuel Temperature†† ($T_{Moderator}$ & T_{Fuel})	$T^{base}_{moderator}$	68°F
	T^{base}_{fuel}	68°F
4. Moderator & Fuel Temperature†† ($T_{Moderator}$ & T_{Fuel})	$T^{base}_{moderator}$	300°F
	base fuel	300°F
5. Boron Concentration (N_{boron})	N^{base}_{boron}	0 ppm
6. Xenon Concentration (N_{xenon})	Equilibrium	0
7. Control Rod #	Unrodded	Rodded

* It is recommended that a full flux solution be carried out for each state-point.

† Unperturbed parameters are at their base values indicated in the Standard Problem definition.

\# In the case of the W (17x17) assembly only the unpoisoned assembly is required.

†† Pressure is to be maintained at base value.

DATA FOR FUEL ASSEMBLY STANDARD PROBLEM

17 x 17 W Type Fuel Assembly

1. General Characteristics

Power density (W/Gm-U)	38.4
Average fuel temperature (°K)	968
Average clad temperature (°K)	600
Moderator temperature (°K)	560
Soluble boron concentration (ppm)	400
Average core pressure (psia)	2250
Xenon concentration	Equilibrium
Samarium concentration	Equilibrium

2. Configuration (1/8 assembly)

```
4
1 1
1 1 1
2 1 1 3
1 1 1 1 1
1 1 1 1 1 2
3 1 1 2 1 1 1
1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1
```

1 - Fuel Rod
2 - Burnable Poison Rod (BPR)
3 - Guide Thimble
4 - Instrument Thimble

Note: 1. For an unrodded or unpoisoned case replace all BPRs (2) with guide thimbles (3).

2. For a rodded case replace all BPRs (2) with control rods inserted in guide thimbles (3).

3. Fuel Assembly Data

Rod array	17 x 17
Fuel rods per assembly	264
Rod pitch (in)#	0.496
Assembly pitch (in)**	8.466 x 8.466
Assembly length (in)	151.0
Active fuel length (in)	144.0
Number of spacer grids†	8
Compositon of spacer grid	Inconel 718
Weight of spacer grids (lb)	12
Number of guide thimbles	24
Number of instrument thimbles	1

All dimensions are given at cold (68°F) conditions.
† Seven in active length.
** Center to center assembly pitch.

4. Fuel Rod Data

Clad O.D. (in)	0.374
Clad thickness (in)	0.0225
Diametral gap (in)	0.0065
Clad material	Zircaloy-4

5. Fuel Pellet Data

Material	UO_2 - Undished
Density (% of theoretical)	95
Enrichment (w/o)	2.6
Diameter (in)	0.3225

6. Burnable Poison Rod Data (See Figure 1)

Number per assembly	16
Material	Borosilicate Glass
Density (Borosilicate glass) (gm/cm^3)	2.28
Outside clad O.D. (in)	0.381
Outside Clad I.D. (in)	0.348
Absorber O.D. (in)	0.344
Absorber I.D. (in)	0.185
Inner-tube O.D. (in)	0.1805
Inner-tube I.D. (in)	0.170
Clad material	Stainless Steel
Inner-tube material	Stainless Steel
Boron loading (w/o B_2O_3 in glass rod)	12.5
Weight of Boron-10 (lb/ft)	0.000419

7. Guide Thimbles and Instrument Thimble Data

Number of guide thimbles	24
Number of instrument thimbles	1
Composition of thimbles	Zircaloy-4
Guide Thimble O.D. (in)	0.482
Guide Thimble I.D. (in)	0.450
Instrument Thimble O.D. (in)	0.482
Instrument Thimble I.D. (in)	0.450

8. Control Rod Data

Neutron absorber (w/o)	5% Cd, 15% In, 80% Ag
Absorber diameter (in)	0.341
Absorber density (lb/in^3)	0.367
Cladding material	304 Stainless Steel
Clad O.D. (in)	0.381
Clad thickness (in)	0.0185
Number of control rods	24

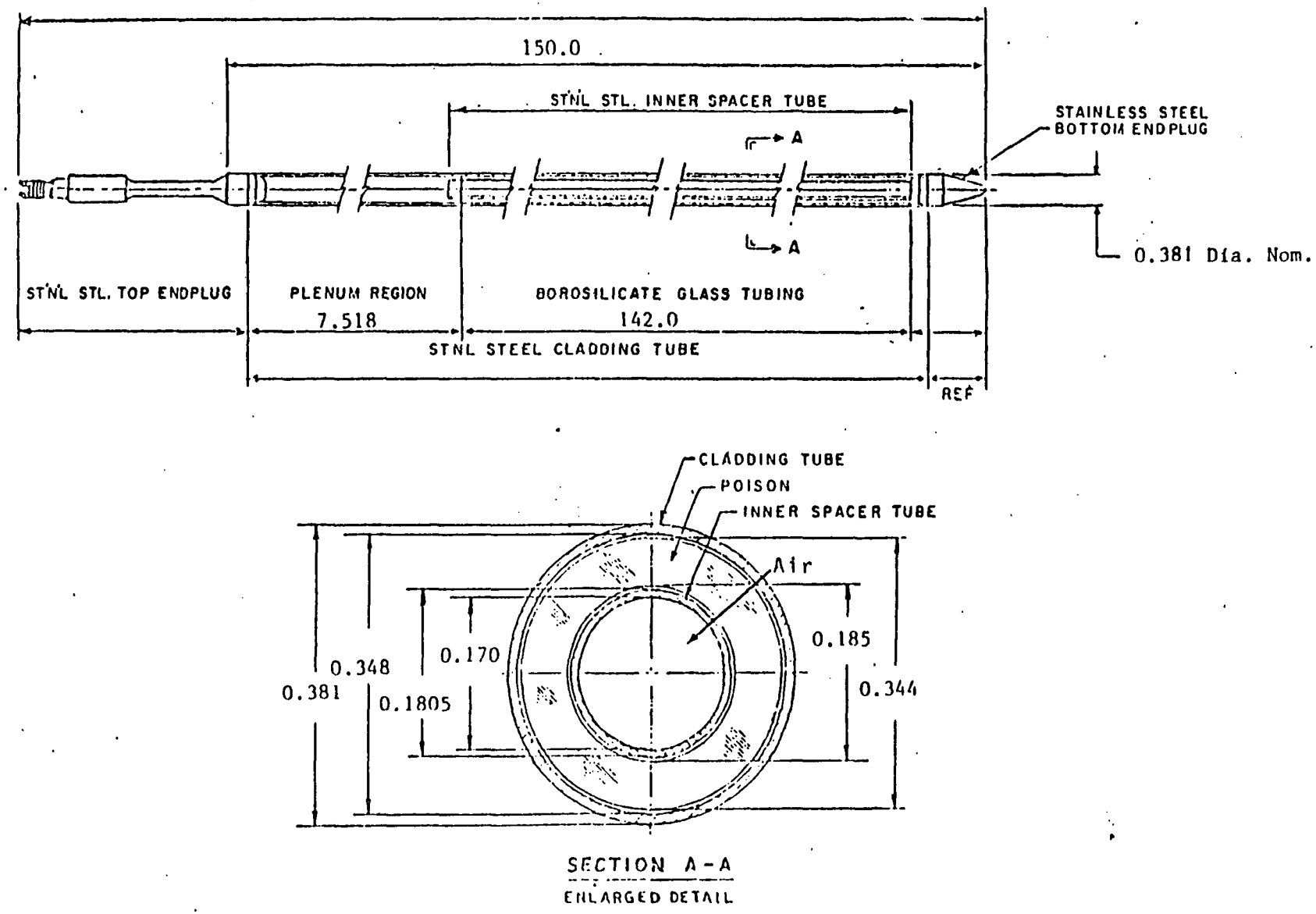


Figure 1. Burnable Poison Rod Configuration

DESCRIPTION OF CALCULATIONS AND METHODS

1. Name of code/code source/version
2. Reference for calculational method
3. Assembly solution method (Diffusion Theory, Collision Probability, Integral Transport, Monte Carlo, etc.)
4. Pin-cell solution method (if distinct from assembly solution method)
5. Spatial mesh assembly/pin-cell (nxm)
6. Neutron cross sections (ENDF/B or other identification)
7. Number of fast/thermal groups in assembly/pin-cell solution
8. Depletion steps

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

December 19, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Subject: McGuire Nuclear Station -- Docket No. 50-369/370
Catawba Nuclear Station - Docket No. 50-413/414
Response Request for Additional Information Regarding
Topical Report DPC-NF-2010, "Nuclear Physics Metodology
for Reload Design"

In response to the request by telephone conference (between NRC, Duke and Brookhaven) on December 17, 1984 for additional information regarding the subject topical report, attached is Duke Power Company's revised answer to question number five, regarding pin power uncertainties.

If any additional information or discussion is desired, please feel free to call Scott Gewehr, Duke Power Licensing at (704) 373-7581.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

SAG/mjf

Attachment

cc: Dr. John Carew
Building 475 B
Brookhaven National Laboratory
Upton, N. Y. 11973

Mr. Jesse L. Riley, President
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28208

James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N. W., Suite 2900
Atlanta, Georgia 30323

8412310017 841219
PDR ADOCK 05000369
P PDR

121984
B97

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

November 30, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Subject: McGuire Nuclear Station - Docket No. 50-369/370
Catawba Nuclear Station - Docket No. 50-413/414
Response Request for Additional Information Regarding
Topical Report DPC-NF-2010, "Nuclear Physics Metodology
for Reload Design"

In response to your request (Reference Letter, E. G. Adensam to H. B. Tucker, November 5, 1984) for additional information regarding the subject topical report, attached are Duke Power Company's answers to the six questions in the request.

If any additional information or discussion is desired, please feel free to call Scott Gewehr, Duke Power Licensing at (704) 373-7581.

Very truly yours,

H. B. Tucker

Hal B. Tucker

SAG/mjf

Attachment

cc: Dr. John Carew
Building 475 B
Brookhaven National Laboratory
Upton, N. Y. 11973

Mr. Jesse L. Riley, President
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28208

James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

cc: Mr. Robert Guild, Esq.
P. O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135 ½ Devine Street
Columbia, South Carolina 29205

Mr. W. T. Orders
Senior Resident Inspector
McGuire Nuclear Station

Senior Resident Inspector
Route 2, Box 179N
York, South Carolina 29745

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P. O. Box 355
Pittsburgh, Pennsylvania 15230

bcc: William L. Porter, Esq.
Duke Power Company
P. O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

North Carolina MPA-1
P. O. Box 95162
Raleigh, North Carolina 27625

North Carolina Electric Membership Corp.
3333 North Boulevard
P. O. Box 27306
Raleigh, North Carolina 27611

Saluda River Electric Cooperative, Inc.
P. O. Box 929
Laurens, South Carolina 29360

N. A. Rutherford
R. L. Gill
K. S. Canady
L. H. Flores
R. H. Clark
H. T. Snead
P. M. Abraham

Q.1 Please provide additional information regarding the NUC-MARGINS code and its use in the Dropped Rod Analysis. Provide short descriptions of the input, output, calculational models used, benchmark calculations performed and the conservatisms assumed in the analysis.

A.1 Under the terms of the current fuel contract with Westinghouse, Duke Power will provide physics data for the rod drop transient to Westinghouse who will then perform the safety evaluation and/or reanalysis. This relationship will exist until Duke submits its thermal-hydraulic and safety analysis methodology reports to the NRC.

The physics methods described in Section 4.2.2.5, 6.2.2.4, and 9.1.3.3 will be further elaborated herein.

A. Initial conditions for analysis:

1. Control Bank D is inserted to the Rod Insertion Limit.
2. Core Power is 102% Full Power (2% calorimetric error included).
3. A full power xenon distribution is used which would produce a DNB limiting axial power profile.

B. Assumptions for system response upon rod drop:

1. No trip occurs.
2. Control bank D is withdrawn to compensate for the dropped rod.
3. A short duration reactor power overshoot will occur with the turbine-reactor control system eventually leveling out the reactor power to the initial power level.

Search cases are performed as described in Section 4.2.2.5 and 6.2.2.4. EPRI-NODE assembly average powers are converted to $F_{\Delta H}^{N}$ using the method described below. This method is employed for all $F_{\Delta H}^{N}$ evaluations. All physics codes employed are static, therefore, "before" and "after" rod drop power distributions are calculated.

The mathetical formulation of $F_{\Delta H}^{C}$ employs the Section 6.2.1.2 definitions as follows:

$$F_{\Delta H,j}^{C} = [\sum_{i=1}^{K} F_{i,j}^{node} \times RL_j^N + FR_j \times F_{K,j}^{node} \times RL_j^N + (1-FR_j) \times F_{K,J}^{node} \times RL_j^R + \sum_{i=N+1}^{M} F_{i,j}^{node} \times RL_j^R] \div M$$

and then:

$$F_{\Delta H}^{C} = \underset{j}{Max} \, (F_{\Delta H,j}^{C})$$

Where:

M = Number of axial nodes.

RL_j^N = Non-rodded radial local factor for assembly j.

RL^R_j = Rodded radial local factor for assembly j.

FR_j = Linear fraction of assembly j which does not contain a control rod.

Radial local factors are edited by PDQ-EDIT using fine mesh PDQØ7 mesh average powers. The PDOØ7 cases are two-dimensional simulations with control bank(s) explicitly represented.

The nodal powers, $F_{i,j}^{node}$, are steady state three-dimensional calculations which explicitly model; control bank insertion, boron and xenon conditions, and other reactor state point variables necessary for a best estimate power distribution calculation.

$F_{\Delta H}^C$ is then evaluated by the NUC-MARGINS code or by hand calculations using the nodal powers from NODE-P and the RL from PDQ07. The NUC-MARGINS code has been independently verified to yield the correct $F_{\Delta H}^N$. $F_{\Delta H}^T$ is the ultimate output as defined by equation 6-2 for DNB analysis.

The system transient response and the transient DNB calculations would be performed by Westinghouse if the physics parameters exceeded the bounds of the previous analyses.

Q.2 Identify the nominal and various off-nominal cross-section sets that are generated in order to evaluate the different reactivity coefficients and defects.

A.2 The various fuel cross-section sets that are generated in order to evaluate different reactivity coefficients and defects are identified in Table 2.1. Nominal cross-sections are generated as a function of burnup at an average moderator temperature of 594°F and an average fuel temperature of 1250°F. The off-nominal cross-sections are generated at various burnups with varying moderator and fuel temperatures.

The cross-section representation in PDQØ7 differs between the quarter-core discrete pin and colorset models. The representation employed in the quarter-core model is discussed first and then the colorset discussion follows. All sets, except the baffle, use combined macroscopic and microscopic cross-sections.

Fuel cross-sections in quarter-core PDQØ7 are calculated according to the following relation:

$$\Sigma(T_M, T_F, Bu) = \Sigma_o(Bu) + \frac{\Delta\Sigma}{\Delta T_M} \times (T_M - T_M^{Ref}) + \frac{\Delta\Sigma}{\Delta\sqrt{T_F}} \times (\sqrt{T_F} - \sqrt{T_F}^{Ref})$$

where $\Sigma(T_M, T_F, Bu)$ = the total macroscopic cross-section as a function of moderator temperature, fuel temperature, and burnup.

$\Sigma_o(Bu)$ = the nominal macroscopic cross section as a function of burnup.

$\frac{\Delta\Sigma}{\Delta T_M}$ = the moderator temperature pseudo-microscopic cross-section which relates the change in macroscopic cross-section to change in moderator temperature.

$\frac{\Delta\Sigma}{\Delta\sqrt{T_F}}$ = the fuel temperature pseudo-microscopic cross-section which relates the change in macroscopic cross-section to a change in fuel temperature.

The macroscopic cross-sections given here may be of any type, e.g. transport, absorption, removal, or fission. The pseudo-microscopic cross-sections (or pseudo-micros) account for the change in the macroscopic cross-section as a result of a change from reference conditions. These pseudo-micros are input to PDQØ7 as a function of burnup. The moderator temperature pseudo-micros are determined from the cross-section sets at moderator temperatures of 630°F and 530°F (fuel temperature held constant at 1250°F).

The fuel temperature pseudo-micros are determined from the cross-section sets at fuel temperatures of 1250°F and 594°F (moderator temperature held constant at 594°F).

Most nonfuel cross-sections employed in quarter core calculations are evaluated as shown in Table 2.4, and are consistent with the core average moderator temperature of interest.

The reflector constants are evaluated at T_{inlet} (usually 557°F) and, at Hot Zero Power, are identical to the water gap constants. Baffle constants are evaluated using the method shown in Chapter 4 of EPRI NP-3642-SR (Few-Group Baffle and/or Reflector Constants for Diffusion Calculation Application, EPRI Special Report, August 1984).

Colorset PDQØ7 calculations are performed which provide sufficient data to characterize operation from Hot Full Power (HFP) to Cold Zero Power (CZP) conditions. A breakpoint is designated at Hot Zero Power (HZP). Two sets of data (B-Constants) are then used in EPRI-NODE-P calculations:

1. Normal Operation - HFP to HZP

2. Low Temperature - HZP to CZP

B-Constants for the Normal Operation and Low Temperature models are generated following the sequence described in Section 3 of DPC-NF-2010.

Tables 2.1 and 2.4 describe conditions for fuel and non-fuel cross-section sets. The Normal Operation cross-sections input to colorset PDQØ7 calculations are shown by the matrices in Table 2.2. Table 2.3 shows matrices of cross section sets for Low Temperature colorset calculations. Nonfuel cross-section sets (Table 2.4) are used which are consistent with the fuel moderator temperature.

Table 2.1

McGuire/Catawba
Fuel Cross-Section Sets

Cross-Section Set Type	T_{mod} (°F)	T_{fuel} (°F)	Power	Burnup Timesteps (GWD/MTU)	Application
P1	594	594	Zero	0.0	HFP → HZP
P2 (Nominal)	594	1250	Full	0.0	"
P3	630	1250	Full	0.0	"
P4	530	1250	Full	0.0	"
P8 (Nominal)	594	1250	Full	0.0, 0.1, 0.5, 1.0, 2.0, 4.0, 6.0, ..., 58.0, 60.0	"
P8B6	594	594	Full	"	"
P8B7	530	1250	Full	"	"
P8B8	630	1250	Full	"	"
P5	200	200	Zero	0.0	HZP → CZP
P9	200	200	Zero	0.0, 0.1, 0.5, 1.0, 2.0, 4.0, 6.0, ..., 58.0, 60.0	"
P6	557	557	Zero	0.0	"
P7	68	68	Zero	0.0	"

Table 2.2

Cross-Section Sets for Normal Operation
PDQØ7 Colorsets

BOL

Effect	Cross-Section Set Type P2(Nominal)	P1	P3	P4
Soluble Boron	X			
K-inf vs. T_{mod}	X		X	X
Migration Area vs. T_{mod}	X		X	X
Doppler	X	X		

Depletion

Reactivity Effect	Cross-Section Set Type P8(Nominal)	P8B6	P8B7	P8B8
Exposure	X			
Soluble Boron	X			
Control Rods	X			
Xenon	X			
Doppler	X	X		
Moderator			X	X

Table 2.3

Cross-Section Sets for Low Temperature
PDQØ7 Colorsets

BOL

Effect	Cross-Section Set Type P5	P6	P7
Soluble Boron	X		
K-inf. vs. T_{mod}	X	X	X
Migration Area vs. T_{mod}	X	X	X

DEPLETION

Reactivity Effect	Cross-Section Set Type P9
Exposure	X
Soluble Boron	X
Control Rods	X

Table 2.4

McGuire/Catawba
Non-fuel Cross-Section Sets

Material	Moderator Temperatures (°F)
Water Gap/Reflector	630, 594, 557, 530, 200, 68
Guide Tube/Inst. Tube	630, 594, 557, 530, 200, 68
Control Rod	594, 557, 200, 68
Burnable Poison Rod	594, 557, 200, 68
Baffle	EPRI NP-3642-SR

Q.3 Provide a short description of the PDQ-EDIT code and describe the verification program that was undertaken to test data generated with PDQ-EDIT for use in SNA-CORE.

A.3 PDQ-EDIT is a utility code written by Duke Power Company that is capable of reading Internal File Management (IFM) files written by PDQ07. This code is primarily used to develop theoretical factors for SNA-CORE, and to edit and process data contained on pointwise flux, power and concentration IFM files. PDQ-EDIT, like all Nuclear Design software used in safety related analysis, is quality assured as required by Duke Power Company's Administrative Policy Manual for Nuclear Stations.

SNA-CORE theoretical factors are generated from PDQ-EDIT in what is commonly known as theoretical factor sets. Each theoretical factor set is valid over a user defined burnup range. Theoretical factor sets consist of assembly average powers, assembly peak pin powers, and detector mesh average two-group fluxes.

Verification of theoretical factor sets is accomplished by the utility code SNAVER. SNAVER compares the symmetric assembly average and peak pin powers on either a 1/4-core or 1/8-core basis, and then calculates a percent difference for each power at a given location with respect to the average at that location. Percent differences greater than 0.1% are flagged by the program. The cognizant engineer must then verify whether these errors are justified. SNAVER also checks for consistancy between detector fluxes at symmetric locations, and for correct data format.

The formal benchmarking of theoretical factors developed from PDQ-EDIT was accomplished by comparing measured powers from Westinghouse's INCORE code, to those calculated from SNA-CORE for Sequoyah Unit 1 Cycle 1. All measured powers were inferred from plant supplied flux traces. Results from these comparisons are shown in Figures 1 thru 7. Good agreement between the two codes was observed. A summary of the average absolute relative error, and the standard deviation associated with these errors are presented in Table 1.

In conclusion, comparisons between measured data from Westinghouse's INCORE code and Duke's SNA-CORE code demonstrate the accuracy of the PDQ07, PDQ-EDIT, SNA-CORE code package. Also, in addition to the software quality assurance program employed at Duke, SNAVER provides an independent means of verifying the correctness of theoretical factor sets before they are used in a production environment.

Table 1

Statistical Summary of INCORE versus SNA-CORE
Measured Powers for Sequoyah 1 Cycle 1

CASE	Burnup EFPD	Average Absolute Relative Error (%)	Standard Derration %
1	71.82	1.34	1.84
2	101.62	1.06	1.43
3	133.30	1.14	1.48
4	166.04	1.28	1.64
5	231.70	1.21	1.48
6	292.04	1.20	1.51
7	378.92	1.05	1.34

Average Absolute Relative Error $(\bar{D}) \equiv \left|[(\text{SNA-CORE} - \text{INCORE})/\text{INCORE}]\right| * 100$

$$\bar{D} \equiv \sum_{i=1}^{N} D_i/N$$

FIGURE 1

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
71.82 EFPD 100(%)FP CONTROL BANK D AT 200 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
8	1.12	1.05	1.17	1.11	1.15	1.05	1.01	.71	
	1.17	1.08	1.17	1.14	1.18	1.07	1.01	.71	
9		1.16	1.11	1.19	1.16	1.13	1.01	.77	
		1.17	1.13	1.19	1.17	1.13	1.03	.77	
10			1.18	1.12	1.18	1.09	.98	.66	
			1.17	1.14	1.18	1.11	.97	.65	
11				1.18	1.13	1.08	.92	.56	
				1.19	1.16	1.08	.92	.55	
12					1.09	.99	.86		
					1.12	.99	.83		
13						1.02	.51		SNA-CORE
						.99	.49		INCORE

FIGURE 2

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
101.62 EFPD 100(%)FP CONTROL BANK D AT 218 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.14	1.06	1.16	1.13	1.17	1.06	1.00	.71	
8	1.16	1.09	1.17	1.15	1.17	1.08	1.00	.71	
		1.16	1.12	1.18	1.16	1.12	1.01	.76	
9		1.17	1.14	1.18	1.18	1.13	1.03	.76	
			1.18	1.13	1.17	1.09	.97	.65	
10			1.17	1.15	1.17	1.11	.96	.65	
				1.17	1.13	1.08	.91	.55	
11				1.18	1.16	1.08	.92	.55	
					1.11	1.00	.85		
12					1.11	1.00	.83		
						1.02	.51		SNA-CORE
13						.99	.50		INCORE

FIGURE 3

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
133.30 EFPD 100(%)FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.14	1.08	1.17	1.14	1.16	1.07	.99	.70	
8	1.16	1.11	1.17	1.17	1.17	1.09	.99	.71	
		1.17	1.14	1.19	1.17	1.12	1.01	.76	
9		1.17	1.16	1.18	1.19	1.12	1.03	.76	
			1.18	1.14	1.17	1.09	.96	.65	
10			1.17	1.16	1.16	1.11	.95	.65	
				1.16	1.13	1.06	.91	.55	
11				1.17	1.16	1.06	.92	.55	
					1.09	1.00	.84		
12					1.10	1.00	.82		
						1.01	.51		SNA-CORE
13						.98	.50		INCORE

FIGURE 4

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
166.04 EFPD 100(%)FP CONTROL BANK D AT 210 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.13	1.09	1.17	1.15	1.14	1.08	.99	.71	
8	1.16	1.11	1.17	1.18	1.15	1.10	.99	.71	
		1.16	1.15	1.19	1.17	1.11	1.01	.76	
9		1.17	1.18	1.18	1.19	1.11	1.03	.76	
			1.18	1.15	1.16	1.09	.96	.66	
10			1.17	1.18	1.15	1.11	.95	.65	
				1.16	1.13	1.06	.91	.55	
11				1.17	1.17	1.06	.92	.55	
					1.08	1.00	.84		
12					1.09	1.00	.82		
						1.00	.51		SNA-CORE
13						.97	.50		INCORE

FIGURE 5

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
231.70 EFPD 100(%)FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.10	1.08	1.14	1.16	1.13	1.09	.99	.72	
8	1.12	1.10	1.14	1.19	1.13	1.12	.99	.73	
		1.13	1.16	1.16	1.17	1.09	1.02	.76	
9		1.14	1.18	1.15	1.19	1.09	1.04	.77	
			1.16	1.16	1.14	1.10	.96	.67	
10			1.15	1.18	1.13	1.12	.95	.68	
				1.14	1.14	1.05	.92	.56	
11				1.14	1.17	1.05	.93	.56	
					1.07	1.02	.84		
12					1.08	1.02	.82		
						1.00	.53		SNA-CORE
13						.98	.52		INCORE

FIGURE 6

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
292.04 EFPD 100(Z)FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.07	1.07	1.12	1.15	1.11	1.10	.99	.74	
8	1.09	1.09	1.12	1.18	1.12	1.13	.99	.75	
		1.11	1.15	1.14	1.16	1.08	1.03	.78	
9		1.11	1.18	1.13	1.18	1.08	1.05	.78	
			1.14	1.16	1.12	1.11	.97	.69	
10			1.13	1.18	1.11	1.13	.96	.69	
				1.12	1.13	1.05	.93	.58	
11				1.12	1.16	1.05	.94	.57	
					1.06	1.04	.85		
12					1.07	1.04	.83		
						1.00	.55		SNA-CORE
13						.98	.54		INCORE

FIGURE 7

SEQUOYAH 1 CYCLE 1 SNA-CORE VS. INCORE MEASURED POWERS
378.92 EFPD 100(%)FP CONTROL BANK D AT 222 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A	
	1.02	1.04	1.08	1.14	1.09	1.10	1.00	.77	
8	1.06	1.06	1.08	1.15	1.09	1.13	1.01	.79	
		1.07	1.13	1.10	1.14	1.07	1.05	.80	
	9	1.08	1.15	1.10	1.15	1.07	1.07	.81	
			1.10	1.14	1.09	1.11	.98	.73	
		10	1.09	1.15	1.09	1.13	.97	.74	
				1.09	1.13	1.05	.96	.60	
			11	1.10	1.15	1.05	.97	.60	
					1.06	1.06	.87		
				12	1.07	1.06	.85		
						1.02	.58		SNA-CORE
					13	1.00	.57		INCORE

Q.4. Comment on the reasons for the 3.1% non-conservative bias in the calculated peak axial powers (Section 11.5.4). Describe the model refinements, if any, that have been undertaken to reduce this bias.

A.4. The reason there is a -0.031 bias on the calculated peak axial powers (Section 11.5.4) is that the models used by Duke at the time of this report underpredicted the peak axial power. This -0.031 bias is the mean difference ($\bar{D}$) and is defined by equation 11-2. This value is a difference and not a percentage difference. The mean percent difference for all cases considered was -2.195% (Table 11-10). Again, it should be pointed out, that this number applies to all peak C, M pairs $\geq$ 1.0.

Although Dukes' models underpredict the peak axial power on an average of -2.195%, the Observed Nuclear Reliability Factor (ONRF) directly reflects this non-conservative prediction. This can be seen by examining equation 11-11. Because $\bar{D}$ is subtracted from $\bar{M}$, this equation is conservative for all cases of $\bar{D}$. (That is, $\bar{D}$ being positive, negative, or 0)

Consider the ONRF calculation of the peak axial power on Table 11-6. In this example if $\bar{D}$ were 0 the ONRF would be 1.035. With a $\bar{D}$ of -0.031 the ONRF is 1.058. This is a 2.2% increase in ONRF. The $\bar{D}$ of -0.031 represents a 2.195% underprediction of measured peak axial power. (Table 11-10). Therefore, it can be seen from this example, that there is a 1% increase in ONRF for each 1% that the model underpredicts the measured peak axial power.

In summary, even though the models used by Duke underpredict the peak axial power, the ONRF reflects this underprediction. As shown in the above example, there is a 1 to 1 correspondence in the percentage of the underprediction to the percentage increase in the ONRF.

The model refinements undertaken to reduce this underprediction are discussed in the answer to question 6 parts one and two. The refinements are; 1) normalization of EPRI-NODE-P to include unrodded M^2 adjustments, and 2) an increase in the number of axial nodes. Attached are the results of some maps compared to predictions using 12 levels and 18 levels of EPRI-NODE-P. Attached are the Difference Means and Standard Deviations for Assembly Peak Axial Powers (C, M $\geq$ 1.0), and Assembly Radial Powers. Also attached are Percent Difference Means (C, M $\geq$ 1.0) for Assembly Peak Axial Powers and Assembly Radial Powers.

Table 4-1

Difference Means and Standard Deviations for Assembly Radial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	N	$\overline{D}$	$S(\overline{D})$	$\overline{ABS(D)}$	S(ABS(D))
M1/C2	12 Level	144	-0.002	0.017	0.014	0.010
M1/C2	18 Level	144	-0.002	0.015	0.012	0.010

Difference Means and Standard Deviations for Assembly Peak Axial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	N	$\overline{D}$	$S(\overline{D})$	$\overline{ABS(D)}$	S(ABS(D))
M1/C2	12 Level	232	-0.004	0.031	0.025	0.018
M1/C2	18 Level	246	0.030	0.035	0.036	0.029

Percent Difference Means for Assembly Radial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	Mean % Difference	Mean Absolute % Difference
M1/C2	12 Level	-0.170	1.35
M1/C2	18 Level	-0.142	1.17

Percent Difference Means for Assembly Peak Axial Powers
(C, M $\geq$ 1.0)

Unit/Cycle	EPRI-NODE-P Model	Mean % Difference	Mean Absolute % Difference
M1/C2	12 Level	-0.407	2.039
M1/C2	18 Level	2.382	2.890

FIGURE 4.1

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

18 EFPD 100%FP CONTROL BANK D AT 207 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.95	1.08	1.24	.97	.93	.80	1.09	1.28
8	.93	1.06	1.27	.98	1.00	.[illegible]5	1.19	1.27
	1.10	1.27	1.25	1.03	.98	.93	1.50	1.30
9	1.09	1.27	1.25	1.03	1.02	.95	1.53	1.28
	1.24	1.25	1.25	1.28	1.00	.96	1.13	1.19
10	1.28	1.27	1.27	1.32	1.03	1.00	1.19	1.16
	.98	1.04	1.28	1.25	1.27	1.14	1.52	.92
11	1.00	1.04	1.32	1.28	1.29	1.15	1.48	.91
	.94	.99	1.01	1.27	1.43	1.43	1.29	
12	1.02	1.04	1.02	1.30	1.40	1.41	1.26	
	.81	.93	.97	1.14	1.43	.99	.79	
13	.88	.98	1.05	1.17	1.44	.98	.77	
	1.10	1.51	1.14	1.52	1.30	.80		
14	1.12	1.46	1.14	1.44	1.26	.79		
	1.28	1.31	1.19	.93	CALC			
15	1.27	1.26	1.15	.90	MEAS			

FIGURE 4.2

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

30 EFPD 100%FP CONTROL BANK D AT 194 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.90	1.04	1.21	.95	.92	.82	1.12	1.30
8	.92	1.06	1.26	.98	1.02	.89	1.20	1.30
	1.05	1.24	1.22	1.00	.98	.95	1.53	1.33
9	1.08	1.26	1.25	1.03	1.04	1.00	1.53	1.29
	1.22	1.23	1.24	1.30	1.01	.98	1.16	1.21
10	1.27	1.26	1.26	1.32	1.04	1.04	1.21	1.19
	.95	1.01	1.30	1.26	1.28	1.15	1.54	.94
11	1.00	1.04	1.33	1.29	1.31	1.17	1.52	.94
	.92	.98	1.02	1.28	1.43	1.43	1.31	
12	1.03	1.05	1.04	1.31	1.42	1.44	1.29	
	.83	.96	.98	1.16	1.44	1.00	.80	
13	.90	1.01	1.08	1.19	1.46	1.01	.79	
	1.12	1.54	1.16	1.55	1.31	.81		
14	1.15	1.50	1.16	1.45	1.28	.81		
	1.31	1.33	1.21	.94	CALC			
15	1.30	1.30	1.16	.90	MEAS			

FIGURE 4.3

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.94 .92	1.08 1.04	1.22 1.23	.98 .97	.95 1.00	.81 .86	1.07 1.16	1.24 1.25
9	1.10 1.07	1.25 1.24	1.23 1.22	1.04 1.01	1.00 1.01	.92 .97	1.46 1.48	1.27 1.24
10	1.23 1.25	1.24 1.23	1.24 1.22	1.26 1.27	1.01 1.00	.94 1.01	1.11 1.17	1.15 1.14
11	.99 .99	1.04 1.03	1.26 1.26	1.22 1.22	1.22 1.25	1.10 1.12	1.47 1.45	.90 .90
12	.96 1.00	1.01 1.02	1.01 1.00	1.22 1.25	1.35 1.34	1.36 1.37	1.24 1.23	
13	.82 .87	.92 .97	.94 1.04	1.10 1.14	1.36 1.39	.95 .96	.77 .76	
14	1.08 1.11	1.47 1.45	1.11 1.12	1.47 1.39	1.24 1.23	.78 .78		
15	1.24 1.26	1.27 1.25	1.16 1.12	.90 .87	CALC MEAS			

FIGURE 4.4

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

61 EFPD 100ZFP CONTROL BANK D AT 220 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.92	1.05	1.19	.97	.94	.81	1.08	1.25
8	.91	1.03	1.23	.96	1.00	.86	1.15	1.24
	1.07	1.22	1.20	1.02	.99	.93	1.47	1.27
9	1.06	1.23	1.21	1.00	1.00	.96	1.47	1.24
	1.20	1.21	1.21	1.24	1.00	.95	1.12	1.16
10	1.24	1.22	1.21	1.26	1.00	1.01	1.17	1.14
	.97	1.02	1.24	1.20	1.22	1.11	1.47	.91
11	.99	1.03	1.26	1.22	1.24	1.12	1.45	.90
	.95	1.00	1.00	1.22	1.35	1.36	1.24	
12	1.01	1.02	1.00	1.24	1.33	1.35	1.22	
	.82	.93	.95	1.11	1.36	.95	.77	
13	.88	.98	1.04	1.14	1.38	.96	.75	
	1.09	1.48	1.12	1.48	1.24	.78		
14	1.10	1.44	1.12	1.38	1.22	.77		
	1.25	1.27	1.16	.91	CALC			
15	1.24	1.23	1.11	.87	MEAS			

FIGURE 4.5

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

101 EFPD 100%FP CONTROL BANK D AT 223 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.91 .90	1.03 1.03	1.16 1.23	.96 .97	.95 1.01	.83 .88	1.08 1.14	1.22 1.21
9	1.04 1.04	1.19 1.21	1.17 1.19	1.00 1.00	.99 1.01	.94 .97	1.46 1.45	1.25 1.21
10	1.17 1.22	1.17 1.20	1.17 1.19	1.23 1.25	.99 1.00	.96 1.01	1.11 1.15	1.14 1.13
11	.96 .99	1.01 1.02	1.23 1.25	1.19 1.21	1.20 1.23	1.09 1.10	1.45 1.41	.90 .90
12	.95 1.01	1.00 1.02	.99 1.00	1.20 1.23	1.31 1.30	1.32 1.32	1.21 1.20	
13	.84 .89	.94 .98	.96 1.03	1.09 1.12	1.32 1.34	.94 .95	.77 .76	
14	1.09 1.10	1.46 1.43	1.11 1.11	1.45 1.37	1.21 1.20	.77 .77		
15	1.22 1.21	1.25 1.20	1.14 1.10	.90 .87	CALC MEAS			

FIGURE 4.6

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (12 LEVEL) VS. MEAS

130 EFPD 100%FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.90	1.02	1.18	.95	.95	.85	1.09	1.21
8	.93	1.04	1.23	.98	1.01	.89	1.14	1.20
	1.04	1.20	1.18	1.00	.99	.96	1.46	1.24
9	1.07	1.23	1.20	1.01	1.02	.98	1.45	1.20
	1.18	1.18	1.18	1.24	1.00	.97	1.12	1.14
10	1.23	1.22	1.20	1.26	1.01	1.02	1.14	1.11
	.95	1.00	1.24	1.19	1.20	1.10	1.44	.90
11	.99	1.03	1.26	1.22	1.24	1.11	1.41	.90
	.95	1.00	1.00	1.20	1.30	1.31	1.20	
12	1.03	1.04	1.02	1.24	1.31	1.32	1.19	
	.85	.96	.97	1.10	1.31	.95	.77	
13	.90	1.00	1.05	1.13	1.34	.95	.76	
	1.09	1.46	1.12	1.44	1.20	.78		
14	1.10	1.43	1.11	1.37	1.19	.77		
	1.21	1.24	1.14	.90	CALC			
15	1.19	1.20	1.10	.87	MEAS			

FIGURE 4.7

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

18 EFPD 100%FP CONTROL BANK D AT 207 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.96	1.11	1.27	1.00	.96	.82	1.13	1.32
8	.93	1.06	1.27	.98	1.00	.85	1.19	1.27
	1.13	1.30	1.28	1.06	1.01	.95	1.55	1.35
9	1.09	1.27	1.25	1.03	1.02	.95	1.53	1.28
	1.27	1.28	1.29	1.32	1.03	.99	1.17	1.23
10	1.28	1.27	1.27	1.32	1.03	1.00	1.19	1.16
	1.00	1.06	1.32	1.29	1.31	1.17	1.57	.95
11	1.00	1.04	1.32	1.28	1.29	1.15	1.48	.91
	.97	1.02	1.03	1.31	1.47	1.47	1.33	
12	1.02	1.04	1.02	1.30	1.40	1.41	1.26	
	.83	.96	.99	1.17	1.47	1.01	.82	
13	.88	.98	1.05	1.17	1.44	.98	.77	
	1.13	1.56	1.17	1.57	1.34	.82		
14	1.12	1.46	1.14	1.44	1.26	.79		
	1.33	1.35	1.23	.96	CALC			
15	1.27	1.26	1.15	.90	MEAS			

FIGURE 4.8

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (1B LEVEL) VS. MEAS

30 EFPD 100%FP CONTROL BANK D AT 194 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.92	1.06	1.24	.97	.94	.84	1.15	1.34
8	.92	1.06	1.26	.98	1.02	.89	1.20	1.30
	1.08	1.27	1.26	1.03	1.01	.98	1.58	1.37
9	1.08	1.26	1.25	1.03	1.04	1.00	1.53	1.29
	1.25	1.26	1.27	1.34	1.04	1.01	1.19	1.25
10	1.27	1.26	1.26	1.32	1.04	1.04	1.21	1.19
	.98	1.04	1.34	1.30	1.32	1.19	1.59	.97
11	1.00	1.04	1.33	1.29	1.31	1.17	1.52	.94
	.95	1.01	1.04	1.32	1.47	1.48	1.35	
12	1.03	1.05	1.04	1.31	1.42	1.44	1.29	
	.85	.98	1.01	1.19	1.48	1.03	.83	
13	.90	1.01	1.08	1.19	1.46	1.01	.79	
	1.16	1.59	1.20	1.59	1.35	.84		
14	1.15	1.50	1.16	1.45	1.28	.81		
	1.35	1.38	1.25	.97	CALC			
15	1.30	1.30	1.16	.90	MEAS			

FIGURE 4.9

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.96	1.10	1.25	1.00	.98	.84	1.11	1.28
8	.92	1.04	1.23	.97	1.00	.86	1.16	1.25
	1.12	1.28	1.26	1.06	1.03	.95	1.51	1.31
9	1.07	1.24	1.22	1.01	1.01	.97	1.48	1.24
	1.26	1.27	1.27	1.30	1.04	.97	1.14	1.19
10	1.25	1.23	1.22	1.27	1.00	1.01	1.17	1.14
	1.01	1.07	1.30	1.26	1.26	1.14	1.52	.93
11	.99	1.03	1.26	1.22	1.25	1.12	1.45	.90
	.98	1.03	1.04	1.26	1.39	1.40	1.28	
12	1.00	1.02	1.00	1.25	1.34	1.37	1.23	
	.84	.95	.97	1.14	1.40	.98	.79	
13	.87	.97	1.04	1.14	1.39	.96	.76	
	1.11	1.52	1.15	1.52	1.28	.80		
14	1.11	1.45	1.12	1.39	1.23	.78		
	1.29	1.31	1.20	.93	CALC			
15	1.26	1.25	1.12	.87	MEAS			

FIGURE 4.10

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

61 EFPD 100%FP CONTROL BANK D AT 220 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.95	1.08	1.23	.99	.97	.84	1.12	1.29
8	.91	1.03	1.23	.96	1.00	.86	1.15	1.24
	1.10	1.25	1.24	1.05	1.02	.96	1.52	1.31
9	1.06	1.23	1.21	1.00	1.00	.96	1.47	1.24
	1.23	1.24	1.25	1.28	1.03	.98	1.15	1.20
10	1.24	1.22	1.21	1.26	1.00	1.01	1.17	1.14
	1.00	1.05	1.28	1.24	1.26	1.14	1.52	.93
11	.99	1.03	1.26	1.22	1.24	1.12	1.45	.90
	.98	1.02	1.03	1.26	1.39	1.40	1.28	
12	1.01	1.02	1.00	1.24	1.33	1.35	1.22	
	.85	.96	.98	1.14	1.40	.98	.80	
13	.88	.98	1.04	1.14	1.38	.96	.75	
	1.12	1.53	1.16	1.52	1.28	.80		
14	1.10	1.44	1.12	1.38	1.22	.77		
	1.29	1.31	1.20	.94	CALC			
15	1.24	1.23	1.11	.87	MEAS			

FIGURE 4.11

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

101 EFPD 100%FP CONTROL BANK D AT 223 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.93	1.05	1.20	.98	.97	.85	1.12	1.26
8	.90	1.03	1.23	.97	1.01	.88	1.14	1.21
	1.07	1.23	1.21	1.03	1.02	.97	1.51	1.29
9	1.04	1.21	1.19	1.00	1.01	.97	1.45	1.21
	1.21	1.21	1.21	1.27	1.02	.99	1.15	1.18
10	1.22	1.20	1.19	1.25	1.00	1.01	1.15	1.13
	.98	1.03	1.27	1.23	1.24	1.13	1.49	.93
11	.99	1.02	1.25	1.21	1.23	1.10	1.41	.90
	.98	1.02	1.02	1.24	1.35	1.36	1.25	
12	1.01	1.02	1.00	1.23	1.30	1.32	1.20	
	.86	.98	.99	1.13	1.36	.97	.79	
13	.89	.98	1.03	1.12	1.34	.95	.76	
	1.12	1.51	1.15	1.49	1.25	.80		
14	1.10	1.43	1.11	1.37	1.20	.77		
	1.26	1.29	1.18	.93	CALC			
15	1.21	1.20	1.10	.87	MEAS			

FIGURE 4.12

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

130 EFPD 100%FP CONTROL BANK D AT 216 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.93	1.05	1.21	.98	.97	.88	1.13	1.25
8	.93	1.04	1.23	.98	1.01	.89	1.14	1.20
	1.07	1.24	1.22	1.03	1.02	.99	1.50	1.27
9	1.07	1.23	1.20	1.01	1.02	.98	1.45	1.20
	1.22	1.22	1.22	1.27	1.03	1.00	1.15	1.17
10	1.23	1.22	1.20	1.26	1.01	1.02	1.14	1.11
	.98	1.03	1.27	1.23	1.24	1.13	1.48	.93
11	.99	1.03	1.26	1.22	1.24	1.11	1.41	.90
	.98	1.03	1.03	1.24	1.34	1.35	1.24	
12	1.03	1.04	1.02	1.24	1.31	1.32	1.19	
	.88	.99	1.00	1.13	1.35	.98	.80	
13	.90	1.00	1.05	1.13	1.34	.95	.76	
	1.13	1.50	1.15	1.48	1.24	.80		
14	1.10	1.43	1.11	1.37	1.19	.77		
	1.25	1.27	1.17	.93	CALC			
15	1.19	1.20	1.10	.87	MEAS			

Q.5 Duke Power Company's contention that no uncertainty in calculated pin powers needs to be accounted for has not been adequately established. One of a set of standard problems, recently developed at Brookhaven National Laboratory for a licensee to assess its ability to calculate typical PWR fuel assemblies, is attached. The licensee's solution using PDQ07 will be an important means of determining the uncertainty in the calculated pin peaking factors.

Re: A.5 Based upon the Duke solution to the BNL benchmark assembly problem, BNL has identified an underprediction of the peak pin power after about 15,000 MWD/MTU which increases to about 1% at 40,000 MWD/MTU. As a result of a conference call held December 11, 1984 between BNL, NRC and Duke, it was determined that a two percent radial local uncertainty was conservative and would be applied in a statistical combination with the reliability factors and engineering hot channel factor.

The three factors to be statistically combined to determine the $F_{\Delta H}^{SCUF}$ factor to multiply the calculated $F_{\Delta H}$ are:

1. F_Q^E, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux for manufacturing tolerances. This factor allows for local variations in enrichment, pellet density, and diameter. It's numeric value is 1.03.

2. $F_{\Delta H}^R$, the Observed Nuclear Reliability Factor for $F_{\Delta H}$. This factor is developed in Section 11.5 and is 1.03. It represents the ability of EPRI-NODE-P to calculate assembly average power.

3. RL^R, Radial Local Uncertainty or pin power uncertainty. It represents the ability of EPRI-CELL/PDQØ7 to calculate the pin power in an assembly. Determined to be 2%.

These factors are statistically combined as follows:

$$F_{\Delta H}^{SCUF} \equiv 1+\sqrt{(.03)^2 + (.03)^2 + (.02)^2} = 1.047.$$

Where SCUF is the statistically combined uncertainty factor.

These factors are statistically independent because they are calculated using different codes and represent different phenomena. The NRC has previously reviewed and approved the statical combination of the radial local uncertainty factor and the $F_{\Delta H}^R$ factor in Northern States Power's report "Qualification of Reactor Physics Methods for Application to Prairie Island Units NSPNAD-8101NP", December 1981. In addition, the NRC has previously reviewed and approved the statistical combination of all three factors in Westinghouse's "Improved Thermal Design Procedure", WCAP-8576, July 1975.

The three factors to be statistically combined to determine the F_Q^{SCUF} factor to multiply the calculated F_Q by are:

1. F_Q^E, Engineering Heat Flux Hot Channel Factor, 1.03.

2. F_Q^R, Assembly Peak Axial Observed Nuclear Reliability Factor. This factor is developed in Section 11.5 and consists of a bias of $(\frac{0.031}{1.375})$ and a $K\sigma$ of 0.048.

3. RL^R, Radial Local Uncertainty or pin power uncertainty, 2%.

The factors are combined to determine the F_Q^{SCUF} factor, where SCUF is the statistically combined uncertainty factor, as follows:

$$F_Q^{SCUF} \equiv 1 + \frac{.031}{1.375} + \sqrt{(.03)^2 + (.048)^2 + (.02)^2} = 1.083$$

$F_{\Delta H}^{SCUF}$ will replace $F_{\Delta H}^R$ in equation 6-2 and F_Q^{SCUF} will replace $F_Q^R \times F_Q^E$ in equation 6-3.

Table 5.1

Benchmark Problem

EPRI-CELL/PDQØ7 Analysis

Maximum Rod Power Summary

Exposure (MWD/MT)	Non-BP Assembly	16-BP Assembly
0	1.060	1.107
500	1.059	1.104
5000	1.054	1.073
10000	1.046	1.041
20000	1.028	1.021
30000	1.014	1.016
40000	1.008	1.010

Table 5.2

Benchmark Problem

Reactivity Defect Calculations

No BP's

		0 MWD/MTU		30000 MWD/MTU	
Case	Description	K-Infinity	% Δρ	K-Infinity	% Δρ
0	Base	1.183699	--	0.896243	-
1	Doppler	1.194852	-0.789	0.907013	-1.325
2	MTC	1.186067	-0.169	0.897301	-0.132
3	68°F	1.211947	-1.969	0.898143	-0.236
4	300°F	1.204695	-1.472	0.904724	-1.046
5	SOLB	1.241994	-3.965	0.937659	-4.928
6	Xe	1.223867	-2.773	0.921068	-3.007
7	Rods	0.789700	42.149	0.605476	53.583

16 BP's

		0 MWD/MTU		30000 MWD/MTU	
Case	Description	K-Infinity	% Δρ	K-Infinity	% Δρ
0	Base	1.020581	--	0.901031	--
1	Doppler	1.030387	-0.932	0.912429	-1.386
2	MTC	1.025619	-0.481	0.903525	-0.306
3	68°F	1.069628	-4.493	0.912266	-1.367
4	300°F	1.053687	-3.079	0.916026	-1.817
5	SOLB	1.060567	-3.694	0.938213	-4.398
6	Xe	1.049333	-2.685	0.926059	-3.000

Table 5.2
(Continued)

Additional Xenon Defect Data

No BP's

	0 MWD/MTU	30,000 MWD/MTU
Xenon Defect (% $\Delta\rho$)	-2.773	-3.007
Xenon Concentration (Atoms/cm^3)[1]	2.1337×10^{15}	1.8623×10^{15}
Xenon Defect (% $\Delta\rho$/atoms/cm^3)[2]	-1.300×10^{-15}	-1.615×10^{-15}

16 BP's

	0 MWD/MTU	30,000 MWD/MTU
Xenon Defect (%	-2.685	-3.000
Xenon Concentration (atoms/cm^3)[1]	2.1334×10^{15}	2.0056×10^{15}
Xenon Defect (% /atoms/cm^3)[2]	-1.259×10^{-15}	1.496×10^{-15}

1. Value averaged over entire assembly volume.
   Fuel to Assembly volume ratio = .90459.

2. Defect per unit volume evaluated over entire assembly.

Table 5.3

1. Name of Codes - PDQØ7; EPRI-CELL[1]
   Code Sources EPRI; EPRI[1]
   Version 2; RAM112[1]

2. Reference for Calculational Method - DPC-NF-2010

3. Assembly Solution Method - Two Group Diffusion Theory

4. Pin-Cell Solution Method - Transport Theory [1]

5. Spatial Mesh Assy/Pin-Cell

   Assembly - One mesh interval per pin

   Pin-Cell[1] - Four Mesh intervals in fuel pin
   One mesh interval in clad
   Five mesh intervals in moderator
   Two mesh intervals in extra region

6. Neutron Cross Section Library - ENDF/B4[1]

7. Number of Fast/Thermal Groups

	No. Fast Groups	No. Thermal Groups
Assembly	1	1
Pin Cell[1]	62	35

8. Depletion Steps -

Assembly (hrs) - 0, 150, 500, 1000, 2000, 3000, 4000, 5000, 6000, 8000, 10000, 12000, 14000, 16000, 18000, 20000, 22000, 24000, 26000, 28000, 30000, 32000, 34000, 36000, 38000, 40000

Pin/Cell(MWD/MTU)[1] 0, 0.001, 100, 500, 1000, 2000, 4000, 6000, 8000, 10000, 12000, 14000, 16000, 18000, 20000, 22000, 24000, 26000, 28000, 30000, 32000, 34000, 36000, 38000, 40000

1 _ All cross-section sets for benchmark problem except CRA and BP were calculated with EPRI-CELL.

Table 5.3 (Continued)

1. Name of Codes - CASMO2E[2]
   Code Sources STUDSVIK
   Version 5

2. Reference for Calculational Method - DPC-NF-2010

3. Assembly Solution Method - Two Group Diffusion Theory

4. Pin-Cell Solution Method - Transport Theory[2]

5. Spatial Mesh Assy/Pin-Cell

   Assembly - One mesh interval per pin[1]

   Pin-Cell - One mesh interval per pin[2]

6. Neutron Cross Section Library - ENDF/B3[2]

7. Number of Fast/Thermal Groups

	No. Fast Groups	No. Thermal Groups
Assembly	4	3
Pin-Cell	9	16

8. Depletion Steps

   Assembly - See Table 5.3 page 1

   Pin-Cell (MWD/MTU)[2] - 0, 150, 500, 1000, 2000, 3000, 5000, 7500, 10000, 12500, 15000, 20000, 25000, 30000, 35000, 40000

[2] - Refers to Burnable Poison and Control Rod Data

Figure A-3. Comparisons of Measured and Predicted Normalized Relative Power Densities for Core 1

INCORE DETECTOR	1.018 1.038 .020	1.011 .997 -.014	.987 .979 -.008	.981 .975 -.006	.997 .978 -.019	.966 .958 -.008	.945 .936 -.009
	1.019 1.035 .016	1.067 1.069 .002	1.012 1.015 .003	1.009 1.012 .003	1.058 1.054 -.004	.999 .988 -.011	.945 .941 -.004
		WATER	1.081 1.087 .006	1.090 1.089 -.001	WATER	1.032 1.045 .013	.953 .947 -.006
			1.054 1.070 .016	1.104* 1.117* .013	1.086 1.100 .014	.989 .994 .005	.945 .939 -.006
				WATER	1.059 1.062 .003	.965 .957 -.008	.934 .928 -.006
					.988 .986 -.002	.938 .937 -.001	.923 .919 -.004
				Measured RPD Calculated RPD ΔRPD		.925 .921 -.004	.914 .911 -.003
							.903 .903 .000

RMS(ΔRPD) = 0.008
Max (ABS(ΔRPD)) = 0.020

*Maximum power fuel rod predicted or measured.

FIGURE 5.1

Figure A-4. Comparisons of Measured and Predicted Normalized Relative Power Densities for Core 5

INCORE DETECTOR	1.005 1.026 .021	.913 .886 -.027	.170 .196 .026	.932 .903 -.029	1.036 1.045 .009	1.063 1.077 .014	1.072 1.090 .018
	.999 1.021 .022	1.017 1.012 -.005	.931 .901 -.030	1.007 .997 -.010	1.125 1.135 .010	1.094 1.112 .018	1.069 1.096 .007
		WATER	.988 .962 -.026	1.087 1.073 -.014	WATER	1.158* 1.174* .016	1.100 1.102 .002
			.181 .203 .022	1.050 1.035 -.015	1.131 1.158 .027	1.038 1.105 .017	1.086 1.090 .004
				WATER	1.048 1.018 -.030	1.035 1.018 -.017	1.070 1.070 .000
					.187 .211 .024	.963 .939 -.024	1.054 1.058 .004
				Measured RPD Calculated RPD ΔRPD		1.018 1.009 -.009	1.060 1.069 .009
							1.070 1.083 .013

UO_2-Gd_2O_3

RMS(ΔRPD) = 0.018

Max (ABS(ΔRPD)) = 0.030

*Maximum power fuel rod predicted or measured.

FIGURE 5.2

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.18377
BURNUP	0
*MAX. ROD POWER	1.060

0.								
1.035	1.013							
1.038	1.013	1.015						
0.	1.029	1.032	0.					
1.037	1.014	1.018	1.044	1.051				
1.035	1.011	1.015	1.043	1.060 *	0.	PDQØ7		
0.	1.019	1.023	0.	1.042	1.014	0.975		
1.012	0.991	0.993	1.006	0.989	0.961	0.942	0.932	
0.975	0.971	0.970	0.972	0.964	0.951	0.942	0.939	0.949

FIGURE 5.3

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.17560
BURNUP	500
*MAX. ROD POWER	1.059

0.								
1.035	1.013							
1.037	1.013	1.014						
0.	1.028	1.032	0.					
1.037	1.013	1.017	1.043	1.051				
1.034	1.011	1.015	1.043	1.059 *	0.	PDQØ7		
0.	1.018	1.022	0.	1.041	1.014	.975		
1.012	.992	.993	1.006	.989	.962	.943	.933	
.976	.971	.971	.972	.964	.952	.942	.940	.950

FIGURE 5.4

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.12604
BURNUP	5000
*MAX. ROD POWER	1.054

0.								
1.032	1.012							
1.034	1.012	1.014						
0.	1.026	1.029	0.					
1.034	1.013	1.016	1.040	1.047				
1.031	1.010	1.014	1.039	1.054 *	0.	PDQØ7		
0.	1.017	1.021	0.	1.038	1.013	.977		
1.011	.992	.993	1.006	.990	.965	.947	.938	
.978	.974	.973	.974	.967	.955	.947	.944	.954

FIGURE 5.5

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	1.06962
BURNUP	10.000
*MAX. ROD POWER	1.046

0.								
1.028	1.012							
1.030	1.012	1.013						
0.	1.023	1.026	0.					
1.029	1.012	1.015	1.034	1.040				
1.027	1.010	1.013	1.034	1.046 *	0.	PDQØ7		
0.	1.015	1.018	0.	1.032	1.010	.980		
1.009	.994	.995	1.005	.991	.969	.954	.945	
.981	.978	.977	.978	.971	.961	.953	.950	.958

FIGURE 5.6

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	0.97482
BURNUP	20,000
*MAX. ROD POWER	1.028

0.								
1.019	1.010							
1.019	1.010	1.011						
0.	1.016	1.018	0.					
1.019	1.010	1.012	1.023	1.026				
1.017	1.008	1.010	1.022	1.028 *	0.	PDQØ7		
0.	1.010	1.012	0.	1.019	1.006	.986		
1.005	.997	.997	1.002	.994	.980	.969	.962	
.988	.986	.986	.986	.981	.973	.967	.965	.969

FIGURE 5.7

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	0.89624
BURNUP	30.000
*MAX. ROD POWER	1.014

0.								
1.010	1.007							
1.011	1.007	1.007						
0.	1.009	1.010	0.					
1.011	1.007	1.008	1.013	1.014				
1.010	1.006	1.007	1.012	1.014 *	0.	PDQØ7		
0.	1.005	1.006	0.	1.009	1.002	.992		
1.002	.999	.999	1.001	.997	.989	.982	.978	
.994	.993	.993	992	.989	.985	.981	.979	.98

FIGURE 5.8

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	0
K-INFINITY	0.83305
BURNUP	40.000
*MAX. ROD POWER	1.008

0.								
1.005	1.003							
1.005	1.003	1.004						
0.	1.005	1.005	0.					
1.005	1.004	1.004	1.007	1.008 *				
1.005	1.003	1.004	1.006	1.008 *	0.	PDQØ7		
0.	1.003	1.003	0.	1.005	1.001	.996		
1.001	.999	.999	1.000	.998	.994	.991	.988	
.997	.996	.996	.996	.995	.992	.990	.988	.989

FIGURE 5.9

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.02062
BURNUP	0
*MAX. ROD POWER	1.107

0.								
1.054	1.027							
0.954	0.986	1.029						
0.	0.964	1.046	0.					
0.957	0.986	1.021	1.022	0.959				
1.064	1.030	0.975	0.925	0.883	0.	PDQØ7		
0.	1.060	0.957	0.	0.882	0.906	0.954		
1.107 *	1.062	0.989	0.942	0.950	0.980	1.008	1.038	
1.083	1.066	1.035	1.013	1.013	1.027	1.046	1.067	1.092

FIGURE 5.10

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.01969
BURNUP	500
*MAX. ROD POWER	1.104

0.								
1.053	1.027							
0.957	0.987	1.028						
0.	0.966	1.046	0.					
0.959	0.987	1.021	1.023	0.962				
1.063	1.030	0.976	0.929	0.888	0.	PDQØ7		
0.	1.059	0.959	0.	0.887	0.910	0.955		
1.104 *	1.059	0.989	0.944	0.952	0.980	1.006	1.034	
1.080	1.063	1.033	1.012	1.012	1.025	1.042	1.062	1.087

FIGURE 5.11

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.02749
BURNUP	5.000
*MAX. ROD POWER	1.073

0.								
1.046	1.023							
0.983	0.997	1.024						
0.	0.987	1.040	0.					
0.984	0.996	1.019	1.027	0.987				
1.052	1.024	0.989	0.963	0.938	0.	PDQØ7		
0.	1.045	0.979	0.	0.933	0.941	0.961		
1.073 *	1.038	0.991	0.964	0.964	0.975	0.988	1.005	
1.047	1.035	1.014	1.000	0.998	1.003	1.012	1.025	1.044

FIGURE 5.12

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	1.02278
BURNUP	10,000
*MAX. ROD POWER	1.041

0.								
1.036	1.018							
1.007	1.005	1.018						
0.	1.006	1.031	0.					
1.007	1.004	1.016	1.027	1.008				
1.038	1.016	1.000	0.995	0.986	0.	PDQØ7		
0.	1.029	0.998	0.	0.978	0.972	0.970		
1.041 *	1.017	0.994	0.984	0.977	0.973	0.974	0.979	
1.017	1.009	0.998	0.990	0.986	0.984	0.986	0.992	1.006

FIGURE 5.13

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	0.97150
BURNUP	20,000
*MAX. ROD POWER	1.021

0.								
1.020	1.010							
1.018	1.009	1.010						
0.	1.015	0.017	0.					
1.018	1.008	1.010	1.020	1.020				
1.019	1.008	1.007	1.017	1.021 *	0.	PDQØ7		
0.	1.012	1.010	0.	1.012	1.000	0.984		
1.010	1.000	0.997	1.000	0.991	0.978	0.971	0.968	
0.994	0.991	0.988	0.987	0.982	0.976	0.973	0.973	0.97[illegible]

FIGURE 5.14

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	0.90103
BURNUP	30,000
*MAX. ROD POWER	1.016

0.								
1.011	1.006							
1.013	1.007	1.006						
0.	1.011	1.010	0.					
1.012	1.007	1.007	1.012	1.014				
1.010	1.005	1.006	1.013	1.016 *	0.	PDQØ7		
0.	1.005	1.007	0.	1.010	1.002	0.991		
1.003	1.000	1.000	1.001	1.000	0.987	0.981	0.978	
0.995	0.993	0.992	0.991	0.988	0.984	0.981	0.980	0.983

FIGURE 5.15

PDQØ7 CALCULATED
ROD POWERS

	PDQ-7
PPMB	400
NUMBER BA	16
K-INFINITY	0.84163
BURNUP	40.000
*MAX. ROD POWER	1.010

0.								
1.005	1.003							
1.007	1.004	1.004						
0.	1.006	1.005	0.					
1.007	1.004	1.004	1.007	1.008				
1.005	1.003	1.004	1.008	1.010 *	0.	PDQØ7		
0.	1.003	1.004	0.	1.007	1.002	0.995		
1.001	0.999	0.999	1.001	0.998	0.993	0.989	0.987	
0.996	0.996	0.995	0.995	0.993	0.990	0.988	0.988	0.989

FIGURE 5.16

Q.6 Please provide the updates to DPC-NF-2010, if any, that will make it consistent with the methodologies being used by Duke Power.

A.6 The following sections address updates to the methods described in DPC-NF-2010.

1. EPRI-NODE-P Normalization:

In addition to adjusting radial albedoes, small M^2 adjustments are made for various fuel types (usually only fresh fuel) to attain better agreement with PDQØ7 radial power calculations. Fugures 6.1 and 6.2 show the improvement for assembly radial powers with respect to measurement. Figures 6.3 and 6.4 address assembly peak power improvements. The data in figures 6.1 through 6.4 represent McGuire Unit 1 Cycle 2.

2. Axial Nodal Modeling:

Section 11 of DPC-NF-2010 presents a benchmark analysis which employed twelve axial nodes per assembly. Core-specific axial modeling would conform to the physics requirements of the core. Answer 4 addressed the calculated-to-measured improvement shown by employing eighteen axial nodes per assembly. Should future fuel assemblies become non-uniform, i.e., axial blankets or part length burnable absorbers, the Duke Power version of EPRI-NODE-P can adequately model the core.

Since the upgrades described in parts 1 and 2 have significantly improved calculated-to-measured agreement, the ONRF values for F_Q and $F_{\Delta H}^N$ in DPC-NF-2010 are considered conservative. Therefore, even though the upgraded methods have demonstrated improved agreement, Duke Power will still employ previously derived ONRFs.

3. EPRI-NODE-P Enhancements:

EPRI-NODE-P has received several major enhancements which are discussed below. This enhanced version was used throughout the analyses shown in DPC-NF-2010. These enhancements are:

a. Partial reactivity formulations due to xenon, moderator temperature, and doppler temperature have been revised to include third order burnup dependent multipliers.

b. Fuel assemblies can be axially modeled as containing up to three different fuel types.

c. Rodded M^2 is linearly adjusted according to the fraction of node length occupied by a control rod.

d. The full power volumetric average fuel temperature has been revised to a burnup dependent fourth order polynomial.

f. The nodal source convergence routine has been modified to use the Gauss-Seidel iterative method with the inclusion of an optional acceleration parameter.

g. Minor enhancements have also been made which allow more user-friendly input and output features.

Likewise, Duke Power's fitting code EPRI-SUPERLINK has been modified to provide compatibility with EPRI-NODE-P. All codes are rigorously tested and certified before production usage in conformance with Duke Power's Q/A procedures.

MCGUIRE-1 CYCLE-2 ASSEMBLY RADIAL POWERS - CALC (NO MSQUARE ADJ) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.84	.97	1.15	.88	.88	.73	.91	1.03
8	.83	.94	1.11	.87	.90	.77	.98	1.06
	.99	1.18	1.16	.93	.90	.83	1.25	1.04
9	.97	1.12	1.10	.91	.91	.85	1.26	1.05
	1.16	1.16	1.16	1.20	.92	.86	.93	.96
10	1.13	1.11	1.11	1.16	.91	.89	.99	.97
	.88	.94	1.20	1.16	1.15	.97	1.23	.76
11	.89	.93	1.16	1.11	1.12	.98	1.23	.77
	.89	.91	.92	1.15	1.23	1.16	1.05	
12	.91	.92	.91	1.13	1.19	1.19	1.06	
	.74	.84	.86	.97	1.16	.79	.65	
13	.79	.87	.91	.99	1.20	.83	.66	
	.91	1.25	.93	1.24	1.05	.65		
14	.98	1.26	.97	1.20	1.05	.67		
	1.03	1.05	.96	.76	CALC			
15	1.06	1.05	.95	.75	MEAS			

FIGURE 6.1

MCGUIRE-1 CYCLE-2 ASSEMBLY RADIAL POWERS - CALCULATED VS. MEASURED

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
	.83	.96	1.10	.88	.86	.75	.95	1.03
8	.83	.94	1.11	.87	.90	.77	.98	1.06
	.98	1.13	1.12	.93	.91	.85	1.24	1.04
9	.97	1.12	1.10	.91	.91	.85	1.26	1.05
	1.11	1.12	1.13	1.16	.93	.88	.97	.96
10	1.13	1.11	1.11	1.16	.91	.89	.99	.97
	.88	.94	1.16	1.13	1.13	1.00	1.23	.75
11	.89	.93	1.16	1.11	1.12	.98	1.23	.77
	.86	.91	.93	1.13	1.22	1.19	1.04	
12	.91	.92	.91	1.13	1.19	1.19	1.06	
	.76	.86	.88	1.00	1.19	.85	.66	
13	.79	.87	.91	.99	1.20	.83	.66	
	.96	1.24	.98	1.23	1.05	.66		
14	.98	1.26	.97	1.20	1.05	.67		
	1.03	1.05	.96	.75	CALCULATED			
15	1.06	1.05	.95	.75	MEASURED			

FIGURE 6.2

MCGUIRE-1 CYCLE-2 ASSEMBLY RADIAL POWERS - CALCULATED VS. MEASURED

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.83 .83	.96 .94	1.10 1.11	.88 .87	.86 .90	.75 .77	.95 .98	1.03 1.06
9	.98 .97	1.13 1.12	1.12 1.10	.93 .91	.91 .91	.85 .85	1.24 1.26	1.04 1.05
10	1.11 1.13	1.12 1.11	1.13 1.11	1.16 1.16	.93 .91	.88 .89	.97 .99	.96 .97
11	.88 .89	.94 .93	1.16 1.16	1.13 1.11	1.13 1.12	1.00 .98	1.23 1.23	.75 .77
12	.86 .91	.91 .92	.93 .91	1.13 1.13	1.22 1.19	1.19 1.19	1.04 1.06	
13	.76 .79	.86 .87	.88 .91	1.00 .99	1.19 1.20	.85 .83	.66 .66	
14	.96 .98	1.24 1.26	.98 .97	1.23 1.20	1.05 1.05	.66 .67		
15	1.03 1.06	1.05 1.05	.96 .95	.75 .75	CALCULATED MEASURED			

FIGURE 6.2

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (NO MSQUARE ADJ) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.97 .92	1.11 1.04	1.29 1.23	1.00 .97	1.00 1.00	.81 .86	1.06 1.16	1.30 1.25
9	1.13 1.07	1.32 1.24	1.30 1.22	1.06 1.01	1.02 1.01	.93 .97	1.54 1.48	1.32 1.24
10	1.30 1.25	1.31 1.23	1.31 1.22	1.33 1.27	1.03 1.00	.95 1.01	1.09 1.17	1.20 1.14
11	1.01 .99	1.07 1.03	1.33 1.26	1.28 1.22	1.28 1.25	1.10 1.12	1.53 1.45	.94 .90
12	1.01 1.00	1.02 1.02	1.03 1.00	1.28 1.25	1.41 1.34	1.37 1.37	1.30 1.23	
13	.82 .87	.93 .97	.95 1.04	1.10 1.14	1.37 1.39	.92 .96	.78 .76	
14	1.06 1.11	1.55 1.45	1.10 1.12	1.54 1.39	1.30 1.23	.79 .78		
15	1.30 1.26	1.32 1.25	1.21 1.12	.95 .87	CALCULATED MEASURED			

FIGURE 6.3

MCGUIRE-1 CYCLE-2 ASSEMBLY PEAK AXIAL POWERS - CALC (18 LEVEL) VS. MEAS

48 EFPD 100%FP CONTROL BANK D AT 228 STEPS WITHDRAWN

	H	G	F	E	D	C	B	A
8	.96 .92	1.10 1.04	1.25 1.23	1.00 .97	.98 1.00	.84 .86	1.11 1.16	1.28 1.25
9	1.12 1.07	1.28 1.24	1.26 1.22	1.06 1.01	1.03 1.01	.95 .97	1.51 1.48	1.31 1.24
10	1.26 1.25	1.27 1.23	1.27 1.22	1.30 1.27	1.04 1.00	.97 1.01	1.14 1.17	1.19 1.14
11	1.01 .99	1.07 1.03	1.30 1.26	1.26 1.22	1.26 1.25	1.14 1.12	1.52 1.45	.93 .90
12	.98 1.00	1.03 1.02	1.04 1.00	1.26 1.25	1.39 1.34	1.40 1.37	1.28 1.23	
13	.84 .87	.95 .97	.97 1.04	1.14 1.14	1.40 1.39	.98 .96	.79 .76	
14	1.11 1.11	1.52 1.45	1.15 1.12	1.52 1.39	1.28 1.23	.80 .78		
15	1.29 1.26	1.31 1.25	1.20 1.12	.93 .87	CALC MEAS			

FIGURE 6.4

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

June 26, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church St
Charlottte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: REQUEST FOR ADDITIONAL INFORMATION - APPLICATION FOR CHANGES TO TECHNICAL SPECIFICATIONS (TAC NOS. MB3343, MB3344, MB3222 AND MB3223)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission is reviewing your application dated October 7, 2001, entitled "License Amendment Request applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003" and has identified a need for additional information as identified in the Enclosure. These issues were discussed with your staff on June 6, 2002. Please provide a response to this request within forty-five (45) days of receipt of this letter so that we may complete our review.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

cc:
Ms. Lisa F. Vaughn
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

County Manager of
Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Dr. John M. Barry
Mecklenburg County
Department of Environmental
Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinshouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health and Natural
Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

TECHNICAL SPECIFICATION 5.6.5, CORE OPERATING LIMITS REPORT,

REVISIONS TO BASES 3.2.1 and 3.2.3

REVISIONS TO TOPICAL REPORTS DPC-NE-2009-P,

DPC-NF-2010, DPC-NE-2011-P, AND DPC-NE-1003

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

Topical Reports Numbered DPC-NE-2009-P Duke Power Company Westinghouse Fuel Transition Report and DPC-NF-2010-A, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design

1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

2. To expedite the review process, please provide a qualitative and quantitative technical basis for each of the changes in these topical reports.

3. Please provide validation data that bench-marks the results of comparisons between the old and the new models (changes).

4. If the changes to these topical reports and methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes.

5. Please provide the basis for why the proposed changes to the above stated topical reports should be found acceptable.

Topical Report Numbered DPC-NF-2010-A, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design

1. In the revision history section on page ii, the licensee provides the staff with the reason for the submittal. Since this is a licensing action, please list those Technical Specification(s), Bases, FSAR sections, conformance to regulatory documents, criteria, generic letters, etc. that are impacted by the request for these changes within the licensing framework.

2. Section 4.2.4.2, second paragraph. Please provide clarification of this change and the technical justification for it. Please provide a comparison between the old sentence and the new sentence.

3. In Attachment 7a, "Detailed Listing of the Changes to DPC-NF2010A," it is stated in many places, that "this change is made to avoid difficulties with the literal interpretation of the original description." Please provide clarification of this statement with a supporting example.

4. Section 4.2.4.4, fifth paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

5. Section 8.1, first paragraph. Is the added equation the same as that in the current version of the DPC-NF-2010A topical? If not, please provide technical justification for its use.

6. Section 9.1.5, first paragraph. Please provide clarification of this change and the technical justification for it. Please provide a comparison between the old sentence and the new sentence.

Topical Report Numbered DPC-NE-2011-P-A, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors

1. The description of the transient conditions was changed in Tables 1 and 2, of Section 2.5. It is not clear to the staff exactly what was changed. Please clarify.

2. From section 6.1, please explain what is meant by "updated the equation."

3. From section 6.1, please provide further clarification of this statement.

4. Section 6.2, were is UMR listed in section 6.2? Please provide original definition and new definition for comparison.

Topical Report Numbered DPC-NE-1003, Revision 1 McGuire Nuclear Station and Catawba Nuclear Station Rod Swop Methodology Report for Startup Physics Testings, Revision 1

1. Appendix A of topical report DPC-NE-1003, Revision 1, contains two versions of Duke Power Company's rod swap measurement procedure PT/O/A/4150/11A: Attachment 3 (dated June 1986) and Attachment 4 (dated April 1984). There are differences in these two versions of the procedure. For example, in the Attachment 3 version, Steps 12.2.2 and 12.2.3, respectively, specify the insertion of bank 1 until the indicated reactivity is approximately -20 pcm, and the withdrawal of reference bank until the indicated reactivity is approximately +20 pcm; whereas in the Attachment 4 version, the insertion and withdrawal of bank 1 and reference bank, respectively, of steps 12.2.1 and 12.2.2 specify reactivity change of -/+ 10 pcm.

a. Since the Attachment 3 version of procedures is more recent, why is the Attachment 4 version referenced in Revision 1 of the topical report (Reference 2)?

b. Which of these two versions of rod swap measurement procedures will be used for McGuire and Catawba Units?

2. In the Attachment 3 version of rod swap measurement procedures PT/O/A/4150/11A, Step 12.1.3 states that: "Repeat steps 12.2.1 and 12.2.2 until the previously inserted bank is fully withdrawn."

   Is there a typographic error in the words "steps 12.2.1 and 12.2.2"? Should correct words be "steps 12.1.1 and 12.1.2"?

3. The equation in Section 3, Measurement Procedure, of the topical report for calculating the inferred rod worth of bank x is different from the equation in Step 12.5.3 of the Attachment 3 procedures. The difference appears to be due to the initial height of the reference bank for performing the rod swap measurement of the measured bank.

   Clarify the exact procedure to be used in the rod swap test, and make all necessary corrections in the topical report and the procedures to be consistent.

4. The third sentence in Section 3 of the topical report is revised to read: "All other banks are then exchanged with the reference bank or other test banks at constant boron conditions until the measured bank is fully inserted." It is stated, in Attachment 9a, "Detailed Listing of Changes to DPC-NE-1003A," that the third sentence in Section 3 is revised to make the report consistent with current procedures. The "Revision History" in the topical report states that this revision [Revision 1] also reflects a refinement in the rod swap to make use of two test banks.

   a. What are the current procedures? What is the date of the current procedures?

   b. Are the current procedures the same or different from the ones in Attachment 3? The Attachment 3 procedures do not include the exchange of a test bank with the other test bank.

   c. If the current procedures are different from those of Attachment 3 or 4, provide a copy of the procedures, and appropriately reference them in the report.

   d. Is the statement in "Revision History" referring to this revision? Please explain what the statement means.

Duke Power®
A Duke Energy Company

Duke Energy Corporation
526 South Church Street
PO Box 1006
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
(704) 382-4360 FAX

Michael S. Tuckman
Executive Vice President
Nuclear Generation

August 7, 2002

U. S. Nuclear Regulatory Commission
Washington D.C. 20555-0001
ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 370

Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 414

Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343, and MB3344) and License Amendment Request Supplement

This purpose of this letter is to provide Duke Energy Corporation's (Duke) response to an NRC request for additional information (RAI) and to supplement a Duke license amendment request (LAR) previously submitted pursuant to 10CFR50.90. Please note that some of the information contained in this submittal package has been determined to be proprietary and is being submitted pursuant to 10CFR2.790. This proprietary information is discussed below.

Duke submitted[1] a LAR applicable to McGuire and Catawba Technical Specifications (TS) 5.6.5.a and 5.6.5.b. Also included in this submittal were proposed revisions to the four Duke Topical Reports listed below.

[1] Reference 1: Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated October 7, 2001, SUBJECT: License Amendment Request Applicable to Technical Specification 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003

- DPC-NE-2009-P, *Duke Power Company Westinghouse Fuel Transition Report*, Revision 1;

- DPC-NF-2010, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*, Revision 1;

- DPC-NE-2011-P, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, Revision 1;

- DPC-NE-1003, McGuire Nuclear Station and Catawba Nuclear Station *Rod Swap Methodology Report for Startup Physics Testing*, Revision 1.

The NRC RAI[2] asked questions on these topical reports. As described below, the Duke responses to these questions are included in the attachments to this letter.

In a subsequent submittal,[3] Duke proposed another LAR for McGuire and Catawba TS 5.6.5, but this LAR was only applicable to TS 5.6.5.b. The information contained herein explains the necessary coordination for changing TS 5.6.5.b for McGuire and Catawba. This LAR implements the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler.[4] The NRC has approved and issued this LAR for both McGuire[5] and Catawba.[6] Implementation of the

---

[2] Reference 2: Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation, Dated June 26, 2002, SUBJECT: Request for Additional Information, Application for Changes to Technical Specifications (TAC Nos. MB3222, MB3223, MB3343, and MB3344

[3] Reference 3, Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated December 20, 2001, SUBJECT: License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report – Oconee, McGuire, and Catawba Technical Specification 5.6.5

[4] TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR"

[5] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 10, 2002, SUBJECT: McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3702 and MB3703)

[6] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 2, 2002, SUBJECT: Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3728 and MB3729)

referenced industry traveler eliminates the need for the changes Duke proposed to McGuire and Catawba TS 5.6.5.b in Reference 1. The LAR supplement transmitted herein deletes the proposed changes to McGuire and Catawba TS 5.6.5.b contained in Reference 1. The attached McGuire and Catawba TS pages (both marked and reprinted versions) update Reference 1 such that it contains the latest approved version of the affected TS pages and only applies to McGuire and Catawba TS 5.6.5.a. The affected TS pages are:

McGuire Units 1 and 2 Pages: 5.6-2, 5.6-3, B3.2.1-11, and B3.2.3-4; and

Catawba Units 1 and 2 Pages: 5.6-3, B3.2.1-11, and B3.2.3-4.

As shown, conforming Bases changes have been made and the necessary Bases pages are also included.

The attachments to this letter are listed and described below.

- Attachment 1 provides the Duke response to the NRC's general questions on Topical Reports DPC-NF-2010 and DPC-NE-2011-P.

- Attachment 2 provides the Duke response to the NRC's specific questions on Topical Report DPC-NF-2010.

- Attachments 3a and 3b provide the Duke responses to the NRC's specific questions on Topical Report DPC-NE-2011-P. Attachment 3a is the proprietary version and Attachment 3b is the non-proprietary version.

- Attachment 4 provides the Duke response to the NRC's specific questions on Topical Report DPC-NE-1003.

- Attachment 5 provides the Duke response to an NRC concern on Topical Report DPC-NE-2009-P. This concern was not included in the NRC's RAI,[2] however it was discussed during an NRC/Duke telephone conference held on July 24, 2002.

- Attachments 6a and 6b provide a marked copy of the existing approved Technical Specifications pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively. These marked copies show the proposed changes.

- Attachments 7a and 7b provide the reprinted Technical Specifications and Bases pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively.

Duke has determined that the revisions contained in this LAR supplement, as shown in Attachments 6a, 6b, 7a, and 7b have no impact on the determination of no significant hazards consideration that was included in Reference 1.

This submittal package contains information that Duke considers proprietary. This information is contained within the proprietary version of the response to the NRC questions on Topical Report DPC-NE-2011-P that is provided as Attachment 3a to this letter. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. An affidavit that attests to the proprietary nature of this information is included with this letter. A non-proprietary version of this response is also provided as Attachment 3b to this letter.

Inquiries on this matter should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

xc w/Attachments:

C. P. Patel (Addressee Only)
NRC Senior Project Manager (CNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

R. E. Martin (Addressee Only)
NRC Senior Project Manager (MNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

L. A. Reyes
U. S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

D. J. Roberts
Senior Resident Inspector (CNS)
U. S. Nuclear Regulatory Commission
Catawba Nuclear Site

S. M. Shaeffer
Senior Resident Inspector (MNS)
U. S. Nuclear Regulatory Commission
McGuire Nuclear Site

M. Frye
Division of Radiation Protection
3825 Barrett Drive
Raleigh, NC 27609-7221

R. Wingard, Director
Division of Radioactive Waste Management
South Carolina Bureau of Land and Waste Management
2600 Bull Street
Columbia, SC 29201

M. S. Tuckman, affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: August 7, 2002
Date

Mary P. Dehus, Notary Public

My commission expires: JAN 22, 2006

SEAL

bxc w/Attachments:

M. T. Cash
C. J. Thomas
G. D. Gilbert
L. E. Nicholson
K. L. Crane
K. E. Nicholson
J. M. Ferguson (2) - CN01SA
L. J. Rudy
G. A. Copp
R. L. Gill
P. M. Abraham
G. G. Pihl
D. R. Koontz
R. C. Harvey
MNS Master File - MG01DM
Catawba Master File - CN04DM
NRIA/ELL

Catawba Owners:
Saluda River Electric Corporation
P. O. Box 929
Laurens, SC 29360-0929

NC Municipal Power Agency No. 1
P. O. Box 29513
Raleigh, NC 27626-0513

T. R. Puryear
NC Electric Membership Corporation
CN03G

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

Attachment 1

Responses to Request for Additional Information

Topical Reports Numbered DPC-NE-2011-P, Revision 1, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors and DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design (TAC NOS. MB3343, MB3344, MB3222, MB3223)

General

Subsequent to receiving the NRC RAI package, a clarification of Questions 1, 2, and 3 was obtained from the NRC during a conference call on Thursday July 18, 2002. Responses to all questions in the NRC RAI are given below, and responses to Questions 1, 2, and 3 take into account the clarification received from the NRC.

Question 1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

Response

Subsequent to the approval of the current version of these reports, there have been various changes in calculation methods and plant operating philosophy. Therefore, sections of these topical reports affected by these changes have been reviewed and updated to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. These revisions do not change approved methods nor introduce new methods. These changes and justifications were identified and described in the October 7, 2001 DEC submittal.

Question 2. To expedite the review process, please provide a qualitative and quantitative technical basis for each of the changes in the above stated topical reports.

Response

Qualitative and quantitative bases for each change to DPC-NF-2010 and DPC-NE-2011-P are provided in Attachments 7a and 8a, respectively in the License Amendment Request package submitted by Duke with a cover letter date of October 7, 2001.

Question 3. Please provide validation data, bench-marking the results of comparisons between the old and the new models (changes).

Response

These revisions do not change approved methods nor introduce new methods; therefore, additional benchmarking is not necessary.

Question 4. If the changes to these topical reports/methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes?

Response

The methodology changes correspond to previously approved methodologies or licensing basis documents, or to administrative non-technical changes. Therefore, these changes do not impact the safe operation of the reactor core.

Question 5. Please provide the basis as to why the proposed changes to the above stated topical reports should be found acceptable.

Response
The purpose for these changes is to maintain the topical reports in a condition that is consistent with other current, NRC approved licensing related documents and to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. The changes do not change previously approved methodologies.

Attachment 2
Responses to Request for Additional Information
Topical Report Numbered DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design
(TAC NOS. MB3343, MB3344, MB3222, MB3223)

Question 1. In the revision history section on page ii, the licensee provides the staff with the reason for the submittal. Since this is a licensing action, please list/Tabulate what Technical Specification(s), Basis, FSAR, conformance to regulatory documents, criteria, generic letters, etc., etc. are impacted by the request for these changes within the licensing framework?

Response
The impact to licensing basis documents by changes made to DPC-NF-2010 is described below.

- Technical Specifications and Bases: TS 5.6.5.b

No Technical Specification or Bases requires a change as a result of these revisions. Even the Licensing Amendment Request to change Technical Specification 5.6.5b for this proposed topical report revision is no longer required (see the License Amendment Request to implement the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler (TSTF 363, "Revise Topical Report References in ITS 5.6.5 COLR")).

- UFSAR Sections: 1.6.3, 4.3, and 15.0
- Topical Reports: DPC-NE-1004, DPC-NE-1003, DPC-NE-2004P, DPC-NE-2007P DPC-NE-2009P, DPC-NE-3001P

These documents contain general references to the methods contained in the proposed topical report. Changes to these documents are expected to be made as part of the normal UFSAR and Topical Report update processes.

Question 2. Section 4.2.4.2, second paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Cases are run with the moderator temperature at 5 °F above and at the reference temperatures."

Proposed Sentence: "Cases are run changing the moderator temperature from the reference temperature."

The original sentence may imply that the calculation of the moderator temperature coefficient will be performed by only changing the moderator temperature +5 °F. Whereas, these calculations may be more appropriately performed using a -5 °F change, using an average of the +5 and -5 °F results, or using a different temperature change depending on actual plant conditions. Therefore, specificity is removed to reflect that calculations are performed to match plant conditions or intended use of the data.

Question 3. In Attachment 7a–Detailed Listing of the Changes to DPC-NF2010A, it is stated in many places, that "this change is made to avoid difficulties with the literal interpretation of the original description". Please provide clarification of this statement with a supporting example.

Response
Changes documented in Attachment 7a which state "this change is made to avoid difficulties with the literal interpretation of the original description" also provide additional information about the reason why the literal interpretation could potentially be misconstrued. Changes with this statement can be categorized into 3 types: (1) descriptions of plant operations, (2) descriptions of calculations, and (3) administrative. An example within each category is provided below.

Descriptions of Plant Operations
Example: Change #3

> Section 1.1, First Paragraph
> Description: Changed the third sentence to give examples of intervals between refueling outages.
> Justification: The original sentence implies a maximum fuel cycle length of 18 months, and possible fuel cycle lengths are not limited to 18 months. This change is made to avoid difficulties with the literal interpretation of the original description.

The current version states: "Refueling occurs at intervals of 6 to 18 months, depending on the utility's operational requirements."

The proposed version states: "Refueling occurs at intervals appropriate for the power production needed, for example 12, 18, or 24 months."

A literal interpretation of the current version may imply that development of a core design is limited to a 6 to 18 month fuel cycle, whereas current core designs may be different from the exact range of 6 to 18 months.

Descriptions of Calculations
Example: Change #32

> Section 4.2.1, Third Paragraph
> Description: Clarified the first sentence.
> Justification: Depletion model statepoints may be specified in MWD/MTU or EFPD and may be different than those listed. This change is made to avoid difficulties with the literal interpretation of the original description.

The current version states: "The cycle is then depleted in steps corresponding to 0, 150, 500, 1000, 2000, 4000 ... MWD/MTU to verify that power peaking versus burnup remains acceptable."

The proposed version states: "The cycle is then depleted to various times in the cycle to verify that power peaking versus burnup remains acceptable."

A literal interpretation of the current version may imply that core depletions would have to be performed at the burnup statepoints listed, using MWD/MTU units, and at specific burnup intervals. Current core depletions may use a different set of burnup statepoints and intervals

depending on fuel and burnable poison depletion effects. Also, burnup statepoints may be specified in units other than MWD/MTU (for example EFPD).

Administrative
Example: Change #104

Section 9.1.2, First Paragraph
Description: Changed the last sentence for clarity.
Justification: This change is made to avoid difficulties with the literal interpretation of the original description. Equilibrium xenon worth data may be shown in plot or table format.

The current version states: "The results are displayed in a format similar to Figure 9-4."

The proposed version states: "Figure 9-4 shows the results of a typical equilibrium xenon worth calculation."

A literal interpretation of the current version may imply that equilibrium xenon worth calculation results would be displayed in a plot format to be used in startup test predictions and core physics parameters. However, it is also acceptable to provide this information in a table or electronic database.

Question 4. Section 4.2.4.4, fifth paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Then a second EPRI-NODE case is run with the core power level reduced 5% while holding everything else constant."

Proposed Sentence: "Then a second case is run with the core power reduced while holding control rods, boron, and xenon constant."

The original sentence may imply that the calculation of the power coefficient will be performed by changing the core power -5%. Whereas, these calculations may be more appropriately performed using a different power reduction or increase depending on actual plant conditions. Therefore, specificity is removed to reflect that calculations are performed to match plant conditions or intended use of the data. By removing the reference to the core simulator, the implication is made that any NRC approved model may be used. Finally, the revised sentence removes the ambiguity of the statement "everything else".

Question 5. Section 8.1, first paragraph. Is the added equation the same as that in the current version of the DPC-NF-2010A topical? If not, please provide technical justification for its use.

Response
The equation is in the current approved version of DPC-NF-2010. This equation is located in Section 6.2.1.2 (Page 6-2) of the current version and is labeled Equation "6-1". Section 6 of the proposed version was rewritten for reasons explained in Attachment 7a of the Licensing Amendment Request Package dated October 7, 2001.

Section 6 was rewritten, because subsequent to the initial NRC approval of this topical report, methods for performing safety related calculations were approved by the NRC in References 1, 2, and 3 (below). The NRC excluded Section 6.3 when the NRC SER of the original version of this report was issued. The rewrite of this section references safety analysis methods approved by the NRC (References 1 and 2, below) and provides a brief outline of the physics parameters and power peaking analyses performed, including the application of uncertainty factors. These changes make the methods consistent with current NRC approved methods.

Reference 1 - "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.
Reference 2 - "Multidimensional Reactor Transient's and Safety Analysis Physics Parameter Methodology", DPC-NE-3001P-A, November 1991.
Reference 3 - "FSAR Chapter 15 System Transient Analysis Methodology", DPC-NE-3002-A, Revision 3, SER Dated February 5, 1999.

Question 6. Section 9.1.5, first paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Calculations using EPRI-NODE are run at these power levels and nominal conditions to provide predicted power distributions for comparison."

Proposed Sentence: "Calculations are performed at these power levels and nominal conditions to provide predicted power distributions for comparison."

Specifically the words "Calculations using EPRI-NODE are run" were changed to "Calculations are performed". This change makes the description in this section valid when other NRC approved design methods are used (for example, SIMULATE).

APPENDIX C

Original Issue NRC SER

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

March 13, 1985

Docket Nos: 50-369, 50-370
and 50-413, 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Topical Report on Physics Methodology for Reloads: McGuire and Catawba Nuclear Station

In response to your letter of July 18, 1984, with its supplemental information provided on November 30, and December 19, 1984, the NRC staff and its contractor, Brookhaven National Laboratory (BNL), have reviewed Duke Power Company Topical Report DPC-NF-2010, entitled "McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," dated April 1984. This topical report is the first of a sequence of topical reports planned in regards to reload design at these stations. It describes the fuel, physics codes, fuel cycle design methods, and derivation of core physics parameters. It also presents statistical benchmarks which quantify reactivity and power distribution uncertainties.

Enclosed is our Safety Evaluation Report (SER) for this review. The SER notes in Section 3 that Section 6.3 and Chapter 7 of the Topical Report were excluded in our evaluation. Section 6.3 discusses the systematic application of safety related physics parameters for reload safety evaluation and, therefore, is outside the scope of the methodology described in the report. Chapter 7 discusses application of the physics methods to power peaking analysis and will be reviewed following a future submittal on three-dimensional power peaking analysis. Apart from these exclusions, we find that the methodology in the report, as modified by Duke's supplemental information, is acceptable for referencing in licensing actions involving nuclear physics calculations for reload design for the McGuire and Catawba Nuclear Stations.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that you publish the accepted version of this report within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, you will be expected to revise and resubmit the report or submit justification for the continued effective applicability of the topical report.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Division of Licensing

Enclosure:
As stated

cc: See next page

McGuire

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

cc: Mr. A. Carr
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. Robert Gill
Duke Power Company
Nuclear Production Department
P. O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

Mr. Wm. Orders
Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
Route 4, Box 529
Hunterville, North Carolina 28078

J. Nelson Grace, Regional Administrator
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

R. S. Howard
Operating Plants Projects
Regional Manager
Westinghouse Electric Corporation - R&D 701
P. O. Box 2728
Pittsburgh, Pennsylvania 15230

CATAWBA

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

cc: William L. Porter, Esq.
Duke Power Company
P.O. Box 33189
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell and Reynolds
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

North Carolina MPA-1
P.O. Box 95162
Raleigh, North Carolina. 27625

Mr. F. J. Twogood
Power Systems Division
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, Pennsylvania 15230

NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

Mr. Jesse L. Riley, President
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28208

Richard P. Wilson, Esq.
Assistant Attorney General
S.C. Attorney General's Office
P.O. Box 11549
Columbia, South Carolina 29211

North Carolina Electric Membership Corp.
3333 North Boulevard
P.O. Box 27306
Raleigh, North Carolina 27611

Saluda River Electric Cooperative, Inc.
P.O. Box 929
Laurens, South Carolina 29360

Senior Resident Inspector
Route 2, Box 179N
York, South Carolina 29745

J. Nelson Grace, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Robert Guild, Esq.
P.O. Box 12097
Charleston, South Carolina 29412

Palmetto Alliance
2135 ½ Devine Street
Columbia, South Carolina 29205

Karen E. Long
Assistant Attorney General
N.C. Department of Justice
P.O. Box 629
Raleigh, North Carolina 27602

cc: Spence Perry, Esquire
Associate General Counsel
Federal Emergency Management Agency
Room 840
500 C Street, S.W.
Washington, D. C. 20472

Mark S. Calvert, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

Mr. Michael Hirsch
Federal Emergency Management Agency
Office of the General Counsel
Room 840
500 C Street, S.W.
Washington, DC 20472

Brian P. Cassidy, Regional Counsel
Federal Emergency Management Agency,
Region I
J. W. McCormach POCH
Boston, Massachusetts 02109

ENCLOSURE

SAFETY EVALUATION REPORT

Report Title:	McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design
Report Number:	DPC-NF-2010
Report Date:	April 1984
Originating Organization:	Duke Power Company
Reviewed By:	Core Performance Branch, BNL, and Core Performance Branch, NRC

1. Introduction

This report describes the methodology adopted by Duke Power Company for the physics analysis of the McGuire and Catawba nuclear reactors. The physics analysis (also referred to as the nuclear design process in the topical report) is intended to determine the values of safety related parameters including those describing the core power distribution, reactivity worths and coefficients, and the reactor kinetics characteristics. These values of the physics parameters are then intended to serve as input to the reload safety analysis.

2. Summary of Report

In this methodology the main computational tools used for the physics analysis are the EPRI-ARMP code system[2] and the CASMO-2 code. The fuel performance codes COMETHE-IIIK[3] and TACO-2[4] are used for fuel performance analyses. CASMO-2, using a processed version of the ENDF/B-3 library in either 69 or 25 groups, and EPRI-CELL, using a 97-group library derived from ENDF/B-4, are used for cross section generation. Strong absorbers are modeled with CASMO-2, and equivalent diffusion theory parameters are generated by matching reaction rates calculated with CASMO-2 and PDQ07.[5] An assembly colorset PDQ07 model is used to generate k_∞ and M^2 data for the EPRI-NODE-P 3-D simulator, while a quarter core PDQ07 model is used for the calculation of x-y power distributions,

control bank worths, boron and xenon worths, and temperature coefficients. The NODE-P model is used for 3-D power distributions, ejected rod worths, differential rod worths, and xenon transient calculations.

The report describes the procedures used to calculate integral and differential control rod worths, shutdown margins, ejected and dropped rod worths, trip reactivity, critical boron concentrations, boron worth, xenon worth, reactivity coefficients, kinetics parameters, radial power peaking, and local power peaking. Measured parameters for the first cycles of McGuire Units 1 and 2, and Sequoyah Unit 1 have been compared with calculated values. Measured and calculated power distributions have been analyzed statistically and 95/95 Observed Nuclear Reliability Factors (ONRF) have been extracted.

3. Summary of Evaluation

The nuclear physics methodology described in Topical Report DPC-NF-2010 is the first part of a reload safety evaluation methodology to be submitted by the licensee, which is expected to also include fuel performance analysis, thermal-hydraulics analysis and transient and accident analysis. The licensee has indicated that this reload methodology will include Reload Safety Analysis Checklist (RSAC) comparisons which will be submitted first in collaboration with the fuel vendor, and later independently by the licensee. The licensee has also indicated that a 3-D Power Peaking Analysis will be submitted separately and, consequently, Sections 7.1, 7.2, 7.3, 7.4 and 7.4.1 will be reviewed after this analysis has been submitted. Although the application of the physics parameters has been briefly discussed in Section 6.3, the systematic application of safety related physics parameters for reload safety evaluation is outside the scope of the methodology described in the topical report and, consequently, has also been excluded from this review. The focus of the present evaluation has been on the adequacy of the methodology for calculating safety related physics parameters for use in reload safety analyses. The reload design methods are discussed in the following sections.

A. Nuclear Code System and Calculational Procedures

The Duke Power nuclear methodology is based on the well known and benchmarked EPRI-ARMP system, CASMO-2 and PDQ07 codes. Additionally, the use of a similar system of nuclear codes has been approved[6] by the NRC for use by Duke Power in the design of reload cores for the Oconee Nuclear Station.[7] The fuel performance codes COMETHE-IIIK and TACO-2, which are used for generating fuel properties related input data for the nuclear codes, are also well known and widely used in the industry. The cross section libraries used with EPRI-CELL and CASMO-2 have been derived from either the ENDF/B-3 or the ENDF/B-4 library, and contain a sufficiently detailed energy structure to enable an accurate determination of safety related physics parameters. EPRI recommended procedures[8] are followed in the use of the nuclear code system. A sufficient number of branch calculations are performed with the PDQ07 colorset model (both at beginning of life (BOL) and at selected burnup points, varying moderator and fuel temperature, soluble boron concentration, control rod insertion and xenon concentration) to allow proper determination of boron, xenon, Doppler and control rod worths and the relevant reactivity coefficients.[11] Sufficiently small steps are taken during the depletion calculations with the quarter core PDQ07 model to properly account for the effects of exposure. Measured values of critical boron concentrations, control rod worths, ejected rod worths, and isothermal temperature coefficients for Cycle 1 of both McGuire Unit 1 and Unit 2 have been compared with predictions. The measured critical boron concentrations are reproduced to within about 60 ppm with a standard deviation of about 15 ppm. Control rod bank worths are reproduced with a standard deviation of less than 8%. The isothermal temperature coefficients are reproduced to within about 5 pcm/°F, with a standard deviation of 1.87 pcm/°F. The quality of agreement between measured and predicted values of these physics parameters is acceptable provided the uncertainties are properly considered in the safety analysis.

B. Safety Related Parameters and Their Application

Calculation and application of the safety related physics parameters are described in chapter 6 of the report. A list of selected reload safety

related physics parameters is given in Table 6-1. It should be noted however, that parameters such as fuel temperature, fuel rod pressure, core DNB limits, fuel census data, maximum critical boron concentration, maximum shutdown boron concentration, which are used in the reload safety analyses of Westinghouse reactors[9], do not appear in Table 6-1. The criteria for evaluating the safety of a reload core design are not specified in sufficient detail. Duke Power should include this information in future topical reports.

C. Kinetics Parameters

Kinetics parameters are calculated using PDQ07 and the DELAY code. The calculated kinetics parameters include the six group delayed neutron fractions and effective yields, the total effective delayed neutron fraction, the prompt neutron generation time, and reactivity versus positive and negative doubling time. PDQ07 is used to obtain spatially averaged isotopic fission rates as a function of burnup, and DELAY is used to calculate kinetics parameters and to relate the reactor period to the inserted reactivity. The kinetics parameters are generated for both beginning of cycle (BOC) hot zero power (HZP) and hot full power (HFP) conditions with all rods out (ARO). A second set of delayed neutron parameters is generated for end of cycle (EOC).

The codes and methodology employed for the determination of these parameters have been previously reviewed and approved[10] by the staff.

D. Radial Local Power Peaking Analysis

A quadrant symmetric EPRI-NODE model is used to calculate nodal power distributions. A full core EPRI-NODE model is used to evaluate non-symmetric power distributions such as those encountered in the dropped rod configuration. The nodal powers are multiplied by the corresponding assembly radial local factor to yield the calculated total peaking factor:

$$F^C_Q = \text{Max}\ \{F_{i\ell}^{Node} \times RL_\ell\}\ , \qquad (1)$$

where RL_ℓ is the radial local factor for assembly ℓ, and $F_{i\ell}^{Node}$ is the nodal

power calculated at the axial location i for the assembly ℓ. The reliability factor for F_Q, F_Q^R, is calculated such that 95% of the calculated powers will be greater than the measured powers at a confidence level of 95%. Applying an additional multiplier, F_Q^E, to account for manufacturing tolerance, the total peaking factor, F_Q^T is defined as

$$F_Q^T = F_Q^R \times F_Q^E \times F_Q^C. \qquad (2)$$

Duke Power Company has presented comparisons between PDQ07 and CASMO-2 predictions of pin powers for 10 fuel assemblies at HFP, BOL, and no xenon conditions. In addition, measured pin powers in cold critical assemblies have been compared to PDQ07 predictions in two cases[7]. None of the measured or calculated lattices had any control rods inserted. On the basis of these results, Duke Power concluded that the PDQ07 prediction of the peak pin power is always conservative with respect to CASMO-2 calculations and to measurement; therefore, no uncertainty in the calculated radial local power is required. In response to a request for additional information, Duke Power has provided (1) results from two cold critical measurements that Duke Power made as prime contractor to DOE (Report DOE/ET/34212-41) and (2) a comprehensive solution to a standard problem recently developed at BNL to evaluate calculations of typical PWR fuel assemblies. The thorough and detailed nature of the solution, supplied in a relatively short period of time, is clearly an indication of the resources available to Duke Power in making physics calculations and their familiarity with the methods and procedures applicable in these analyses.

Comparison of EPRI-CELL/PDQ07 predictions of peak pin powers to measurements for the two criticals in the DOE study show that the EPRI-CELL/PDQ07 predictions of peak pin power are conservative by ~1%. Duke Power believes that the overprediction of pin powers near the water holes is attributable to the use of Mixed Number Density (MND) thermal cross sections. It should be noted, however, that the use of MND cross sections does not necessarily lead to an overprediction of peak pin powers.[12] Comparison of the Duke Power solution to the standard problem with the benchmark solutions shows that at

BOL the Duke Power methods do indeed overpredict the peak pin power by just over 1%. However, the Duke Power methods underpredict the peak pin power by approximately 1% at 40,000 MWD/MTU with the "cross over" occurring smoothly at approximately 15,000 MWD/MTU. The Duke Power predictions are expected to have a similar exposure dependence relative to measurement. Any conservatism that might be present in the methodology used by Duke Power at BOL is not expected to persist at all exposures.

The basic methods used by Duke Power to calculate local radial peaking factors are in wide use, and the uncertainties associated with them have been published.[12,13] A review of the literature indicates that the appropriate uncertainty is a standard deviation of 2% between measured local radial power peaking factors and those calculated with a fine mesh diffusion theory code. In an amendment[14] to DPC-NF-2010 Duke Power has accounted for a 2% uncertainty in the calculation of the local peaking factor. The corresponding revised values of F_Q^R and $F_{\Delta H}^R$ are discussed in Section 3F.

E. Assembly Axial Power Analysis

The EPRI-NODE-P model with 12 axial nodes underpredicts the axial power peaking by an average of 2.2%. This deficiency of the model has been discussed with the licensee, who has noted that the agreement of model prediction to measurement is improved if (1) the number of axial nodes is increased from 12 to 18, and (2) the rodded M^2 is linearly adjusted according to the control fraction in the node. Despite the underprediction of the axial peaking using the EPRI-NODE-P model with 12 axial nodes, the total peaking factor F_Q^T (Equation 2) is not underestimated since the observed nuclear reliability factor (ONRF), F_Q^R, accounts for the bias between measurement and prediction.

While the 12 node model is acceptable, it is recommended that the Duke Power Company use the EPRI-NODE-P model with 18 axial nodes per assembly in all calculations. The enhanced accuracy of the model will improve the representation of non-uniform axial effects in the fuel assemblies.

F. Statistical Analysis

In deriving the calculational uncertainty of the models, the difference between measured and calculated power peaking factors has been assumed to be a normally distributed random variable. The D'Test has been applied to the difference distributions to establish their normality. The one-sided upper tolerance limit (OSUTL) on the difference variable, D, is

$$OSUTL(\overline{D}) = D+KxS(D), \qquad (3)$$

where $\overline{D}$ is the mean value of the difference variable, S(D) is the standard deviation, and K is the (sample size dependent) one-sided tolerance factor for the 95% probability at the 95% confidence level.

Using Equation (3), an upper limit to the calculated parameter can be defined as

$$UL(C) = \overline{M}-\overline{D}+KxS(D), \qquad (4)$$

where $\overline{M}$ is the mean of the measured variable. Finally, the observed nuclear reliability factor (ONRF) is defined

$$ONRF = UL(C)/\overline{M}. \qquad (5)$$

Utilizing 1038 observations (i.e., comparisons between measurements and predictions), the assembly peak axial ONRF (F_Q^R) has been determined by Duke Power to be 1.058, using the values; $\overline{M}$ = 1.375, $\overline{D}$ = 0.031, S (D) = 0.028 and K = 1.7259.

As noted in Section 3D, this value of F^R_Q assumes that there is no uncertainty in the calculation of the local power peaking factor. If, as indicated in reference 14, a fractional uncertainty of .02 is assumed for the local peaking, then by statistically combining the uncertainties for manufacturing tolerance (.03), assembly axial peaking (0.035), and local peaking (.02) the following

reliability factor for the total peaking, F_Q^{SCUF}, is obtained

$$F_Q^{SCUF} = 1 + (.031/1.375) + [(.03)^2 + (.035)^2 + (.02)^2]^{1/2} = 1.073. \qquad (6)$$

The corresponding Duke Power analysis for the radial ONRF ($F_{\Delta H}^R$) using $\bar{M}$ = 1.131, $\bar{D}$ = 0.002, S(D) = 0.02 and K = 1.7343 (846 Observations) results in an ONRF ($F_{\Delta H}^R$) of 1.029. As in the case of the F_Q^{SCUF}, combining the uncertainties due to manufacturing tolerance (.03), the radial assembly peaking (.03) and the radial local peaking (.02) yields

$$F_{\Delta H}^{SCUF} = 1 + [(.03)^2 + (.03)^2 + (.02)^2]^{1/2} = 1.047. \qquad (7)$$

These values for F_Q^{SCUF} and $F_{\Delta H}^{SCUF}$ include a 2% allowance for uncertainty in the calculation of the local peaking factor and are acceptable.

4. CONCULSION

The Duke Power Company Topical Report on Nuclear Physics Methodology for Reload Design (DPC-NR-2010) has been reviewed. As noted in Section 3 above, Sections 6.3, 7.1, 7.2, 7.3, 7.4, and 7.4.1 of the Topical Report were excluded from this evaluation.

Apart from these exclusions the methodology described in DPC-NF-2010 and modified in Reference 14 is found to be acceptable for referencing in licensing documents for the McGuire and Catawba Nuclear Stations.

References

1. "Advanced Recycle Methodology Program System Documentation," Nuclear Associates International, CCM-3, September 1977.

2. "CASMO-2 A Fuel Assembly Burnup Program," Studsvik Energiteknik AB, Studsvik /NP-81/3, 1981.

3. "COMETHE-IIIK-A Computer Code for Predicting Mechanical and Thermal Behavior of a Fuel Pin," Belgonuclear S.A., BN7609.1, March 1977.

4. "TACO2-Fuel Pin Performance Analysis," Babcock & Wilcox, BAW-10411, Lynchburg, Virginia, August 1979.

5. "PDQ-7 Reference Manual II," Bettis Atomic Power Laboratory, WAPD-TM-947(L), February 1971.

6. Letter from P. Wagner, NRC, to W. O. Parker, Jr., providing results of review of Duke Power Company Technical Report NSF-1001 on Oconee Nuclear Station Reload Design Methodology, July 29, 1981.

7. "Duke Power Company/Oconee Nuclear Station/Reload Design Methodology" Technical Report NFS-1001," Duke Power Company, NFS-1001, Rev. 1, May 1980.

8. B. M. Rothleder, et al., "PWR Core Modeling Procedures for Advanced Recycle Methodology Program," RP-976-1, August 1979.

9. "Westinghouse Reload Safety Evaluation Methodology (WCAP-9272)," F. M. Boredelon et al., Westinghouse Electric Corporation, March 1978.

10. Letter, W. O. Parker to H. R. Denton, "Oconee Reload Design Methodology Topical Report," Question 3, Docket Nos. 50-269, 270, 287, November 13, 1980.

11. Letter, H. B. Tucker (Duke Power) to H. R. Denton (NRC), "Response Request for Additional Information Regarding Topical Report DPC-NF-2010," Docket Nos., 50-369/370, 50-413/414, November 30, 1984.

12. R. D. Mosteller and R. S. Borland, "COPHIN Code Description," EPRI NP-1385, Electric Power Research Institute, April 1980.

13. P. L. Langford Jr., and R. J. Nath, "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7398-L, Westinghouse Electric Corporation, April 1969.

14. Letter, H. B. Tucker (Duke Power) to H. R. Denton (NRC), "Response Request for Additional Information Regarding Topical Report DPC-NF-2010, "Docket Nos., 50-369/370, 50-413/414, December 19, 1984.

APPENDIX D

Revision 1 NRC SER

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

October 1, 2002

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: McGUIRE NUCLEAR STATION, UNITS 1AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3222 AND MB3223)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 208 to Facility Operating License NPF-9 and Amendment No. 189 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also addresses these topical reports.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 208 to NPF-9
2. Amendment No. 189 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

cc:
Ms. Lisa F. Vaughn
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

County Manager of
Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Dr. John M. Barry
Mecklenburg County
Department of Environmental
Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health and Natural
Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Power Company, et al. (DPC, the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). A new item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," is also proposed to be added to TS 5.6.5.a.

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.
- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the McGuire TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 10, 2002, the NRC issued amendments numbered 203 and 184 to the McGuire Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 203 and 184 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 203 and 184, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 203 and 184 at the McGuire Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2, specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCOs.

Since many parameter limits, such as core physics parameters, generally change with each reload core, licensees previously needed to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, thus eliminating unnecessary burden on the licensees and the NRC to update these limits in the TS for each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The McGuire TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions to four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions concentrated on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of RFAs in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to its Chapter 6, "UFSAR Accident Analyses."

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39, Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model, that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic for a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010, Revision 1

Topical Report DPC-NF-2010, (Ref. 7), describes DPC's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology that DPC intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010 and the responses to the requests for additional information pertaining to these changes. The staff has concluded

that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes to be acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NF-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as the McGuire and Catawba Nuclear Stations. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control (RAOC) Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system (RPS) trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report, the staff finds the changes to be acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10), describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it to be acceptable. The licensee, in the August 7, 2002, letter also modified the equation in Section 3 of the topical report for the calculation of the inferred rod bank worth from the

measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The values of the limits are located in the COLR. For McGuire Nuclear Station, Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3, and

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

McGuire TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also place surveillance limits for conducting the end of cycle MTC measurement at boron concentrations of 300 ppm and 60 ppm. The LCO limits and the 300 ppm and 60 ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the McGuire TS would add the 60 ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60 ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are

calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as a function of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable, as stated in the staff's Safety Evaluation supporting license Amendment Nos. 188 and 169, respectively, for McGuire Nuclear Station, Units 1 and 2 (Ref. 14).

The proposed changes to the McGuire TS would add Item 5.6.5.a.12 that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

4.0 SUMMARY

The staff has reviewed the revisions to four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to McGuire Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation, described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change recordkeeping, reporting, or administrative procedure requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67FR 54680). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

2. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

5. DPC-NE-2009-P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985..

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9. DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990.

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

14. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2411 and MA2412)," September 22, 1999.

15. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2359 and MA2361)," September 22, 1999.

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

Principal Contributor: Y. Hsii
A. Attard

Date: October 1, 2002

EC050_

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

October 1, 2002

RECEIVED
OCT - 8 2002
REGULATORY COMPLIANCE

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3343 AND MB3344)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 202 to Facility Operating License NPF-35 and Amendment No.195 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also address these topical reports.

A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager, Section 1
Project Directorate II /RA/
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 202 to NPF-35
2. Amendment No. 195 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

Catawba Nuclear Station

cc:

Mr. Gary Gilbert
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Ms. Lisa F. Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

Catawba Nuclear Station

cc:

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). Two new items were also proposed to be added to TS 5.6.5.a. These two items are (1) Item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and (2) Item 13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.

- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the Catawba TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 2, 2002, the NRC issued amendments numbered 199 and 192 to the Catawba Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 199 and 192 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 199 and 192, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 199 and 192 at the Catawba Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2 specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCO.

Since many parameters limits, such as core physics parameters, generally change with each reload core, licensees need to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, and thus eliminates the unnecessary burden on the licensees and the NRC to update these limits in the TS each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The Catawba TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to the GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions of four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions will concentrate on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of the RFA design in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009-A, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to Chapter 6, "UFSAR Accident Analyses:"

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39 a Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, the Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic to a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010A, Revision 1

Topical Report DPC-NF-2010A, (Ref. 7), describes Duke Power Company's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology the licensee intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010A, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the Topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010A and the responses to the requests for additional information pertaining to these changes. The staff has concluded

that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NE-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as McGuire and Catawba Nuclear Station. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information to the staff regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consists mostly of administrative changes and clarifications to the original NRC approved topical report, the staff find the changes acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10) describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it acceptable. The licensee in the August 7, 2002, letter also modified the equation in Section 3

of the topical report for the calculation of the inferred rod bank worth from the measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The valves of the limits are in the COLR. For Catawba Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3,

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and

(3) adding Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

Catawba TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also places surveillance limits for conducting the end of cycle MTC measurement at 300 ppm and 60 ppm boron concentration. The LCO limits and the 300-ppm and 60-ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the Catawba TS would add the 60-ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60-ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector

system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as functions of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable as stated in the staff's safety evaluation supporting license amendment Nos. 180 and 172, respectively for Catawba Units 1 and 2 (Ref. 15).

The proposed changes to the Catawba TS would add Item 5.6.5.a.12, that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

3.2.3 Reactor Makeup Water Pumps Combined Flow Rates Limit

The relocation of the reactor makeup water pumps combined flow rates limit for the boron dilution mitigation system from Catawba TS 3.3.9 and 3.9.2 to the COLR was approved by the NRC as described in a letter dated March 25, 1994 (Ref. 16). The reactor makeup water pumps flow rate limit is included in the Catawba COLR.

The proposed changes to the Catawba TS would add Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specification 3.3.9 and 3.9.2," to TS 5.6.5.a. The addition of this item would make the TS 5.6.5.a list consistent with the core operating limits included in the Catawba COLR and is therefore, acceptable.

4.0 SUMMARY

The staff has reviewed the revisions of four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to Catawba Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding [67 FR 54680]. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

2. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

5. DPC-NE-2009-P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

7. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985.

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9. DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990.

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

14. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2411 and MA2412)," September 22, 1999.

15. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2, Re: Issuance of Amendments (TAC Nos. MA2359 and MA2361)," September 22, 1999.

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

Principal Contributor: Y. Hsii
A. Attard

Date: October 1, 2002

APPENDIX E

Revision 2 NRC Approval Letter

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

June 24, 2003

Mr. M. S Tuckman
Executive Vice President
Duke Energy Corporation
526 South Church St
Charlotte, NC 28201-1006

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, RE: TOPICAL REPORT DPC-NF-2010, NUCLEAR PHYSICS METHODOLOGY FOR RELOAD DESIGN, REVISION 2

Dear Mr. Tuckman:

The Nuclear Regulatory Commission (NRC) staff has reviewed your letter dated November 19, 2002, that proposed a revision to the Duke Power Topical Report, DPC-NF-2010, "Nuclear Physics Methodology for Reload Design," Revision 2. Specifically, you proposed to delete the sentence in Section 3.2 of the Topical Report that states: "A secondary data source for the core simulator are estimates of fuel pellet volume averaged temperatures which are calculated by fuel performance codes as a function of power and burnup." Your letter proposed to delete this sentence to avoid the implication that the fuel temperature data needed for core neutronics calculations is based on specific fuel performance code calculations such as would be obtained from the COMETHE-IIIK or TACO-2 codes. You stated that other fuel temperature data may be available that are better suited for core neutronics calculations.

The NRC staff has reviewed the change to the Topical Report that was proposed in your letter dated November 19, 2002, and finds that it does not impact or change any of the conclusions made in our most recent Safety Evaluation of this Topical Report. The subject Safety Evaluation was issued with Amendment Nos. 208 and 189 to the McGuire Nuclear Station, Unit Nos. 1 and 2 Facility Operating Licenses, respectively, on October 1, 2002. Therefore, the NRC staff determined the proposed change to be acceptable.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

cc: See next page

McGuire Nuclear Station
Catawba Nuclear Station

cc:
Ms. Lisa F. Vaughn
Legal Department (ECIIX)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U. S. Nuclear Regulatory
Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

Dr. John M. Barry
Mecklenburg County
Department of Environmental
Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

NCEM REP Program Manager
4713 Mail Service Center
Raleigh, NC 27699-4713

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

McGuire Nuclear Station
Catawba Nuclear Station

cc:
Mr. Gary Gilbert
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626-0513

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Henry Porter, Assistant Director
Division of Waste Management
Bureau of Solid and Hazardous Waste
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
4830 Concord Road
York, South Carolina 29745

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Mr. Dhiaa Jamil
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, North Carolina 28078