Duke Power Company

Nuclear Design Methodology

Using

CASMO-3/SIMULATE-3P

DPC-NE-1004A
(Revision 1)

December 1997

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Nuclear Design Methodology Using CASMO-3/SIMULATE-3P
DPC-NE-1004A
Revision 1

Abstract

On December 12, 1995, and as supplemented on April 9, 1996, Duke Power submitted revision 1 to DPC-NE-1004A. This submittal requested a review of new reliability and uncertainty factors applicable to Westinghouse reactors based on a 24 axial level SIMULATE-3P model, versus the 12 axial level model used in the original analysis. This re-evaluation of reliability and uncertainty factors was performed to capture the effects of current reload design strategies versus the strategies employed in the original uncertainty factor determination. An SER was issued by the Nuclear Regulatory Commission on April 26, 1996 accepting the revision.

The results of the statistical analysis and the calculation of observed nuclear reliability factors (ONRFs) and uncertainty factors are included in Section D. The NRC's request for additional information, including Duke Power Company's responses, and the NRC SER are also included this Section.

A new Section was added to this report which includes all 10 CFR50.59 evaluations which were performed to modify the licensing basis of this report. These evaluations are contained in Section E.

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Section A

(NRC Acceptance Letter dated November 23, 1992 and NRC Safety Evaluation of Duke Power Nuclear Reactor Methodology, DPC-NE-1004, Revision 0)

NRC Safety Evaluation Report

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

November 23, 1992

H. B. Tucker, Senior Vice President
Duke Power Company
P.O.Box 1007
Charlotte, North Carolina, 28201-1007

Dear Mr. Tucker:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT DPC-NE-1004,"NUCLEAR DESIGN METHODOLOGY USING CASMO-3/SIMULATE-3P"

The staff reviewed Topical Report DPC-NE-1004, which describes the Duke Power Company's (DPC's) alternative methodology using the CASMO-3 fuel assembly depletion code and SIMULATE-3P, a three-dimensional code simulator. The licensee submitted the methodology in this topical report to support steady state core physics calculations for operating its B&W 177-assembly plants and its 193-assembly Westinghouse plants. The topical report summarizes the calculational method, the benchmark comparisons for critical boron concentration, the control rod worths, the isothermal temperature coefficients, and the core power distributions. The report includes evaluations of power distribution uncertainties based on comparisons to measured data from McGuire Unit 2 and from Catawba Units 1 and 2.

The staff finds the application of DPC-NE-1004 to be acceptable for referencing in license applications to the extend specified, and under the limitations stated in DPC-NE-1004 and the associated U. S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for accepting this topical report.

The staff will not repeat its review of the matters found acceptable as described in DPC-NE-1004, when the report appears as a reference in license applications, except to verify that the material presented applies to the specific plant involved. Our acceptance applies only to the matters described in the application of DPC-NE-1004.

In accordance with procedures established in NUREG-0390, the staff requests that the Duke Power Company publish accepted versions of this topical report, proprietary and non proprietary, within 3 months of receiving this letter. The accepted version shall include an "A" (designating accepted) following the report identification symbol.

If the staff's criteria or regulations change so that its conclusions about the acceptability of the report are invalidated, Duke Power Company and/or the applicants referencing the topical report should revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revising their respective documentation.

Sincerely,

Ashok C. Thadani, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DUKE POWER NUCLEAR REACTOR METHODOLOGY, DPC-NE-1004

DUKE POWER COMPANY

1.0   INTRODUCTION

By letter dated January 24, 1990, the Duke Power Company (DPC) submitted topical report DPC-NE-1004 for staff review (References 1 and 2). Additional information was submitted on January 23 and on March 30, 1992 (References 3 and 4). The topical report is using the CASMO-3, a fuel assembly depletion code and SIMULATE-3P a three-dimensional core simulator code for steady state core physics calculations to support operation of DPC's B&W 177-assembly plants and its Westinghouse 193-assembly plants. The subject topical report presents DPC's alternative methodology using the above codes and includes uncertainties to be applied to the analyses when these codes are used for reload design. The topical report summarizes the calculational methods based on CASMO-3, SIMULATE-3P, NODE-P and certain auxiliary programs. DPC included benchmark comparisons in the report and presented the results for critical boron concentrations, control rod worths, isothermal temperature coefficients and core power distributions. DPC presented and discussed evaluations of power distribution uncertainties based on measured data from McGuire Unit 2 and Catawba Units 1 and 2. DPC also included radial, axial, and total peaking

uncertainty factors for the B&W and Westinghouse plants. An overview of the topical report is presented in the next section. The safety evaluation of the report is presented in Section 3 and a summary of the technical position is given in Section 4. The staff was assisted in this review by Brookhaven National Laboratory acting as its contractor under FIN No.A-3868.

2.0   SUMMARY OF THE TOPICAL REPORT

The principal components of the DPC steady-state physics methodology are the CASMO-3, SIMULATE-3P, and NODE-P computer codes and models, and associated benchmarks, which DPC intends to use as alternates to its previously approved methods for evaluating reload designs.

Computer Codes

CASMO-3 is a multigroup two-dimensional transport theory code for fuel assembly burnup calculations. It uses a library of 40 or 70 energy group cross sections based primarily on the ENDF/B-IV data base. Certain data used in the CASMO-3 library, such as the Xe-135 yields and fission spectra data for U-235 and Pu-239, are taken from ENDF/B-V. The DPC version of this code produces two-group cross sections, assembly discontinuity factors, fission product data, detector reaction rates, and pin power data. The data from CASMO-3 is reformatted into two- or three-dimensional tables using a data processing program, TABLES-3 for input to the three-dimensional code SIMULATE-3P. SIMULATE-3P interpolates the data from TABLES-3 for the independent

variables for certain core conditions that SIMULATE-3 models.

SIMULATE-3P is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics model. SIMULATE-3P accounts for the effects of fuel and moderator temperature feedback using its nodal thermal-hydraulics model. The program explicitly models the baffle and reflector region. The program uses data from CASMO-3 for each pin in the fuel assembly and uses inter-assembly and intra-assembly data obtained from the coarse mesh solution to reconstruct the power distribution for each pin. NODE-P is a one-group three-dimensional core simulator code for steady-state neutronic calculations for pressurized water reactor (PWR) cores. NODE-P evaluates the distribution of reactivity throughout the core and for each assembly. It also calculates and edits the radial, axial, and three-dimensional power distributions as needed. NODE-P performs the depletion calculations for each node and assembly. The NORGE-P code then reformats the macroscopic nuclear data obtained from CASMO-3 as a function of their respective independent variables and in a form suitable for input into NODE-P. NORGE-P represents parameters dependent on temperature and exposure as polynomial coefficients, generally referred to as B-constants. To ensure that the fitted data are adequate, NODE-P compares to the input data at the corresponding values of the independent variables. Since the NODE-P code is much simpler than SIMULATE-3P and thus runs faster, DPC intends to use it for analyses requiring repetitive calculations, while using SIMULATE-3P to perform most calculations required for reload design and other higher order functions. DPC normalized NODE-P to SIMULATE-3P.

Benchmark Comparisons

DPC verified the adequacy of the SIMULATE-3P model by benchmarking it against
measured data from zero power physics tests and power operation for five
cycles at McGuire, seven cycles at Catawba, and eight cycles at Oconee. The
benchmarking measurements included critical boron concentrations, control rod
worths, isothermal temperature coefficients, and power distributions.

3.0 EVALUATION

The staff evaluated topical report DPC-NE-1004 by: (1) verifying the results
of the CASMO-3 and SIMULATE-3P calculations against the database of benchmark
power distribution measurements, (2) comparing both hot and cold models
against measured data, and (3) evaluating DPC's responses to the questions
raised during the review (References 3 and 4). The staff raised the following
principal issues during the review.

Cross-Section Libraries

Most of the nuclear data used in the CASMO-3 library are from the ENDF/B-IV
data base. However, DPC derived from the ENDF/B-V data base the Xenon-135
yields and fission spectra data for U-235 and Pu-239. Furthermore, DPC
adjusted the U-238 resonances to agree with the Hellstrand resonance integral
measurements. The nuclear libraries will perform better if DPC uses selected
data from later versions of the ENDF/B data base, and if the results of the

new calculations are adequately benchmarked. Upon reviewing DPC's method of extensive benchmarking, the staff accepts DPC's practice of selectively using nuclear data from ENDF/B-V along with ENDF/B-IV libraries in the CASMO-3 calculations.

Hot-Full-Power Boron Predictions

The staff reviewed the comparisons between measurements and the SIMULATE-3P calculations of the hot-full-power (HFP) boron concentration and found that the SIMULATE-3P model underpredicted this parameter for cycle 1 at Catawba Unit 2, cycles 9 and 10 at Oconee Unit 2 and overpredicted the HFP boron concentration for cycles 2 through 5 at McGuire Unit 2. DPC reported that this difference in trends between the B&W and Westinghouse units results from differences in their design, operating procedures, fuel designs, and specific as-built features of each plant. Since the differences between the calculations and the measurements are small and are consistent with expected differences caused by plant-specific fuel designs and plant operation, the staff accepts DPC's method for determining the hot-full-power critical boron concentration.

Reflector Albedos

DPC used the results of SIMULATE-3P calculations to determine the axial and radial albedos used in NODE-P. DPC selected the axial albedos to match axial offsets and axial peaks obtained from the SIMULATE-3P calculations. Similarly,

DPC selected radial albedos in NODE-P to obtain peripheral assembly powers consistent with those calculated by SIMULATE-3P. While the axial albedos remain the same from cycle to cycle when fuel assembly types and burnable poison designs are not changed, the radial albedos are changed every cycle to match peripheral assembly powers calculated by SIMULATE-3P. The method is acceptable because the axial albedos do not vary significantly from cycle to cycle and because the NODE-P model is normalized to the SIMULATE-3P model.

Pin Power Uncertainty

The staff assessed the capability of the CASMO-3 and SIMULATE-3P codes to accurately calculate the power distribution for each pin by analyzing three critical reactor experiments. DPC evaluated the critical experiments using a three-dimensional SIMULATE-3P model with cross sections and discontinuity factors generated by CASMO-3. The maximum pin power error observed in any core location was less than 2.1 percent. Using all of the data from the cores analyzed, DPC obtained an observed nuclear reliability factor (ONRF) of 1.017 for the pin power, with a 95 percent probability and a 95 percent confidence level. Therefore, the selected 2 percent uncertainty factor for pin power is conservative and acceptable.

Power Distribution Measurements

The staff verified that the SIMULATE-3P power distribution calculation is adequate by comparing it with measured benchmark data. At McGuire and Catawba

the incore detector systems consist of movable incore fission chambers which are intercalibrated for each power distribution measurement. The normalized signals are converted to relative assembly powers using sets of data obtained by comparing the PDQ predicted assembly and peak pin powers with the instrument's predicted responses. Precalculated theoretical factors are used to account for exposure during the cycle and the presence of control rods. The Oconee incore detector system consists of fixed rhodium emitters. After being corrected for rhodium depletion, background current, and core configuration, the signals from these detectors are converted to relative assembly powers using precalculated signal-to-power ratios. This method of converting the instrument's response to relative assembly powers is consistent with the industry's current practice and is an acceptable method for determining measured power distributions for benchmarking methods.

4.0 SUMMARY, CONCLUSIONS, AND LIMITATIONS

The staff reviewed the DPC topical report DPC-NE-1004 in detail and evaluated the supporting information supplied in References 3 and 4. The staff concluded that the DPC static nuclear design methodology using CASMO-3 and SIMULATE-3P is acceptable for performing reload analyses for the DPC B&W 177-assembly and Westinghouse 193-assembly cores, subject to the following limitations:

1. The application of CASMO-3 and SIMULATE-3P to fuel designs that differ significantly from those included in the topical data base should be

supported by additional code validation to ensure that the DPC-NE-1004 methodology and uncertainties apply.

2. The system of codes represented in the topical report must be protected with appropriate quality assurance procedures, subject to auditing by the staff.

The staff recommends that the applicant incorporate the contents of References 2, 3 and 4 into a single report for ease of reference.

5.0 REFERENCES

1. Letter from Hal B. Tucker, Duke power Company, to NRC, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004," January 24, 1990.

2. DPC-NE-1004, "Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," Duke Power Company, January 1990.

3. Letter from Hal B. Tucker, Duke Power Company, to NRC, "Topical Report DPC-NE-1004, Response to Request for Additional Information," January,23, 1992.

3. Letter from Hal B. Tucker, Duke Power Company, to NRC, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004," March 30, 1992.

Section B

(Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Rev. 0)

Topical Report

DPC-NE-1004A

Duke Power Company

Nuclear Design Methodology

Using

CASMO-3/SIMULATE-3P

DPC-NE-1004A

November 1992

ABSTRACT

This report presents an alternative methodology for calculating nuclear physics data by Duke Power Company. This methodology is based on the CASMO-3/SIMULATE-3P code package. A new set of reliability and uncertainty factors for this methodology is provided. Duke Power intends to use this methodology for performing nuclear design calculations for both B&W 177-assembly plants and Westinghouse 193-assembly plants.

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1.0 INTRODUCTION

Duke Power Company currently performs reload design analysis for the
Oconee Nuclear Station with the methodology of DPC-NE-1002A
(Reference 1). This methodology is basically the same as that described
in NFS-1001A (Reference 2) with the exception that CASMO-2 was used in
place of EPRI-CELL and colorset PDQ as the cross section and
assembly-averaged physics data generator. DPC-NE-1002A presented CASMO-2
as an alternative methodology and provided appropriate reliability
factors to be applied in conjunction with this methodology. Duke Power
Company also performs analysis for the McGuire and Catawba Nuclear
Stations with the methodology of DPC-NF-2010A (Reference 3). This
methodology utilizes an EPRI-CELL colorset model supplemented by CASMO-2
for modeling strong absorbers. Application of this methodology to reload
design is described in DPC-NE-1003A, DPC-NE-2011P, and DPC-NE-3001P
(References 6, 8, and 9, respectively). Both the Oconee and
McGuire/Catawba methodologies currently being employed utilize 2-D PDQ07
and 3-D EPRI-NODE-P models as reactor simulators.

As part of Duke Power Company's continuing effort to improve its reload
design methods, the CASMO-3/SIMULATE-3P methodology has been developed.
While CASMO-3 is an improved version of the already approved CASMO-2
program, SIMULATE-3P provides a significantly improved reactor simulator
model over those previously approved for use in reload design
applications. The theoretical basis and validation of CASMO-3 and
SIMULATE-3P have previously been provided to the NRC by Yankee Atomic
Electric Company (YAEC) in References 4 and 5, respectively. In these
reports YAEC provides detailed descriptions of the calculations performed
by the programs and a general methodology for implementing them to

perform reload design analysis. This report demonstrates Duke Power Company's competence in implementing these programs and provides the appropriate uncertainties to be applied when they are utilized for reload design.

This report provides a description of the CASMO-3 based methodology as it will be employed by Duke Power Company (Section 2). Comparisons to measured physics parameters, generation of the appropriate Observed Nuclear Reliability Factors (ONRFs), and a description of the application of this methodology to reload design are provided to demonstrate Duke Power Company's competence with these programs (Sections 3 and 4). The SIMULATE-3P program is presented as either a stand-alone reactor simulator or as supplemented by a CASMO-3 based EPRI-NODE-P model with appropriate ONRFs for either case. This methodology is applicable to either Babcock and Wilcox (B&W) 177-assembly plants or Westinghouse 193-assembly plants, and comparisons are provided for both types of plants (Section 5). Finally, the conclusions of this report are provided in Section 6.

2.0 METHODOLOGY OVERVIEW

As part of the reload design process, reactor physics calculations are performed on a cycle-specific basis to finalize the core nuclear design and ensure safety. First, the number of feed assemblies, their enrichment and burnable poison loading are determined from calculations of cycle lengths and nominal power distributions. Then calculations are performed to verify core safety parameters, generate operational and Reactor Protection System (RPS) limits, and identify the core loading pattern. Finally, calculations are performed to provide operational and startup testing data to the plant. Details of these calculations have previously been described in References 1, 2, 3, 6, 8, and 9.

This section provides a brief description of the programs utilized in the CASMO-3/SIMULATE-3P methodology, their interfaces and the sequence employed when they are used to perform the above calculations. Two calculational sequences are presented. In the first sequence SIMULATE-3P is utilized for all core-wide calculations including cycle length, boron concentrations, reactivity coefficients, control rod worths, power distributions, and 3-D analysis for generation of operational and RPS limits. In the second sequence SIMULATE-3P is used in conjunction with EPRI-NODE-P to obtain local power distribution information for generating operational and RPS limits. The analysis performed to set operating limits (References 2 and 8) requires numerous three-dimensional cases. Due to its calculational efficiency, EPRI-NODE-P may be desirable for this type of analysis. When using this sequence, EPRI-NODE-P is supplemented by peak pin to assembly power ratios extracted from SIMULATE-3P. Detailed descriptions of the

calculations performed by each program have previously been provided in References 1, 4, and 5.

2.1 Code Descriptions

2.1(a) CASMO-3

CASMO-3 is a multigroup, two-dimensional transport theory code for burnup calculations on PWR or BWR fuel assemblies. The program may utilize either a 40 or 70 energy group cross section library which is based on ENDF/B-IV with some data taken from ENDF/B-V. The version used by Duke Power Company is capable of editing two group cross sections, assembly discontinuity factors, fission product data, detector reaction rates, and pin power data. In production mode, CASMO-3 is executed in a single assembly format using the 40 energy group library to generate all cross section data required by SIMULATE-3P. All assembly-averaged physics data required by EPRI-NODE-P is also edited.

2.1(b) TABLES-3

TABLES-3 reformats the data provided by CASMO-3 for input as a binary library to SIMULATE-3P. The data is parameterized, according to card input, into 2-D and/or 3-D tables which may be interpolated to provide data at intermediate conditions. The use of 2-D or 3-D tables allows for variable dependencies not available using 1-D or polynomial fits. The base cross section values are generated at nominal reactor

conditions as a function of assembly exposure. Changes in cross sections from their base values (Deltas) are then obtained by altering one independent variable (e.g., boron concentration, moderator temperature) from its base value at various exposures. This is performed for each instantaneous cross section dependency required. Cumulative effects due to operating at off-nominal conditions over a period of time (e.g. moderator history) are treated in an analogous manner. History Deltas are obtained by comparing branch cases from the nominal depletion to depletions performed at the branch case conditions. SIMULATE-3P will interpolate/extrapolate the tabulated Delta cross sections based on the reactor conditions being modelled and add them to the base values to obtain the appropriate total cross sections.

2.1(c) SIMULATE-3P

SIMULATE-3P is a 3-D, two group diffusion theory reactor simulator program. The program explicitly models the baffle/reflector region, thereby eliminating the need to normalize to higher-order fine mesh calculations. Homogenized cross sections and discontinuity factors are used on a coarse mesh nodal basis to solve the two group diffusion equation using the QPANDA neutronics model. A nodal thermal hydraulics model is incorporated to provide both fuel and moderator temperature feedback effects. Inter- and intra-assembly information from the coarse mesh solution is then utilized along with the pinwise assembly lattice data from CASMO-3 to reconstruct pin-by-pin power distributions on a 2-D or 3-D basis. Finally, the program performs a macroscopic depletion with microscopic depletion of particular fission products (i.e., Iodine,

Xenon, Promethium, and Samarium) to model the depletion of fuel in the core.

2.1(d) NORGE-P

NORGE-P formats the lattice physics data calculated by CASMO-3 for input as B-constants to EPRI-NODE-P. Polynomial coefficients are determined for parameters which vary with temperature and burnup, such as K^{∞}, M^2, xenon reactivity, control rod reactivity, K/ν, Σ_f, γ_I, δ_{xe}, etc. NORGE-P automatically plots the input data and the fitted data as well as compares input to fitted data differences to ensure a good fit.

2.1(e) EPRI-NODE-P

EPRI-NODE-P is a 3-D, coarse mesh, modified one group reactor simulator program. Assembly-averaged data is used on a coarse mesh nodal basis to solve the source kernal equation using a FLARE type neutronics model. The model is normalized using albedos to match an independent higher order solution (either PDQ or SIMULATE-3P). Core and assembly-wise reactivity and power distribution information is edited as requested. A macroscopic depletion may then be performed to simulate the depletion of fuel in the core.

2.2 Code Sequence

2.2(a) SIMULATE-3P Sequence

The SIMULATE-3P sequence is simpler and easier to use than the CASMO-2/PDQ/NODE sequence of Reference 1 which was, likewise, an

improvement over the EPRI-CELL based sequence of References 2 and 3. The CASMO-3 program is used to generate the assembly data which is processed by TABLES-3. The binary library produced by TABLES-3 is read directly by SIMULATE-3P. SIMULATE-3P then produces all reactivity and power distribution data required for reload design analysis, including the generation of operating limits. A flow chart of this sequence is shown in Figure 2-1.

2.2(b) EPRI-NODE-P Sequence

The large number of 3-D simulations which are required to generate operating limits for a particular cycle currently make the use of SIMULATE-3P impractical for this type of analysis. However, it is anticipated that continuing advances in the speed and cost of computers will eventually make it the preferred method. Currently, the EPRI-NODE-P program is sufficiently simpler and faster than SIMULATE-3P such that it is the preferred program for calculationally intensive analyses. In this sequence SIMULATE-3P is used as a core simulator for most calculations required for reload design and is also utilized as the higher order code to which EPRI-NODE-P is normalized. NORGE-P is executed to process the CASMO-3 assembly data and produce B-constant input to EPRI-NODE-P. EPRI-NODE-P is then normalized and used as the core simulator for such calculationally intensive analyses. A flow chart of this sequence is shown in Figure 2-2.

Figure 2-1

SIMULATE-3P CODE SEQUENCE
FLOW CHART

```
┌──────────┐
│ CASMO-3  │
└──────────┘
     │
     ▼
Cross  Sections
     │
     ▼
┌──────────┐
│ TABLES-3 │
└──────────┘
     │
     ▼
Binary  Library
     │
     ▼
┌─────────────┐
│ SIMULATE-3P │
└─────────────┘
     │
     ▼
```

Power Distributions, Reactivity Coefficients,
Control Rod Worths, Cycle Length,
Boron Concentrations,
Input to Operating Limits Analysis, etc.
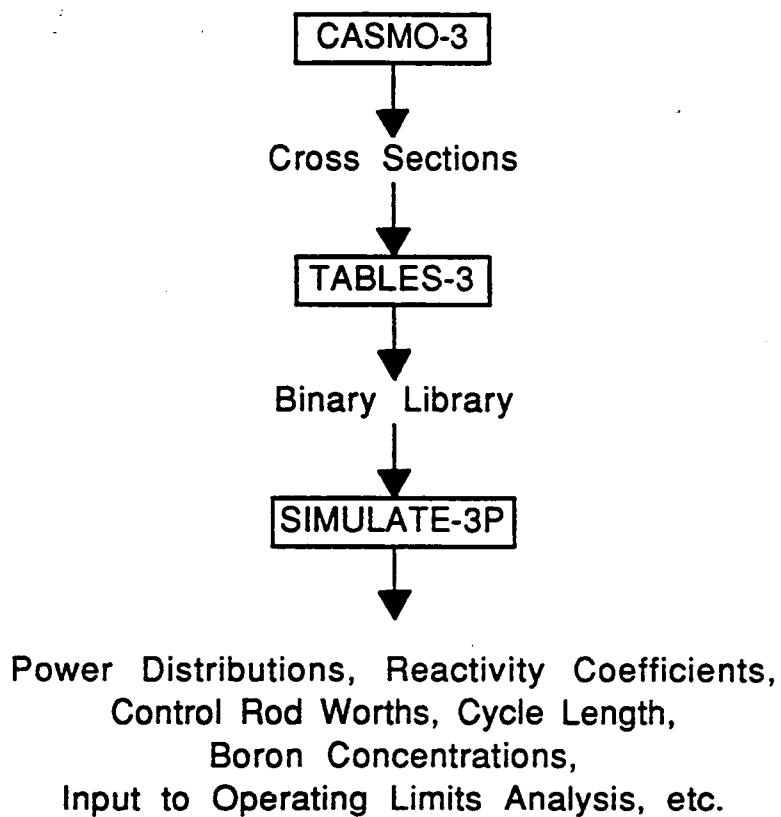
Figure 2-2

EPRI-NODE-P CODE SEQUENCE
FLOW CHART

```
                         ┌──────────┐
                         │ CASMO-3  │
                         └──────────┘
                              │
                              ▼
                        Cross Sections
                       ╱              ╲
                      ▼                ▼
              ┌──────────┐        ┌──────────┐
              │ TABLES-3 │        │ NORGE-P  │
              └──────────┘        └──────────┘
                    │                   │
                    ▼                   ▼
             Binary Library        B-Constants
                    │                   │
                    ▼                   ▼
        ┌─────────────┐  Power Distribution  ┌─────────────┐
        │ SIMULATE-3P ├──────────────────────►│ EPRI-NODE-P │
        └─────────────┘  for Normalization   └─────────────┘
               │                                     │
               ▼                                     ▼
```

Pin/Assembly Factors,   Input to Operating Limits Analysis
Reactivity Coefficients,
Control Rod Worths, Cycle Length,
Boron Concentrations, etc..
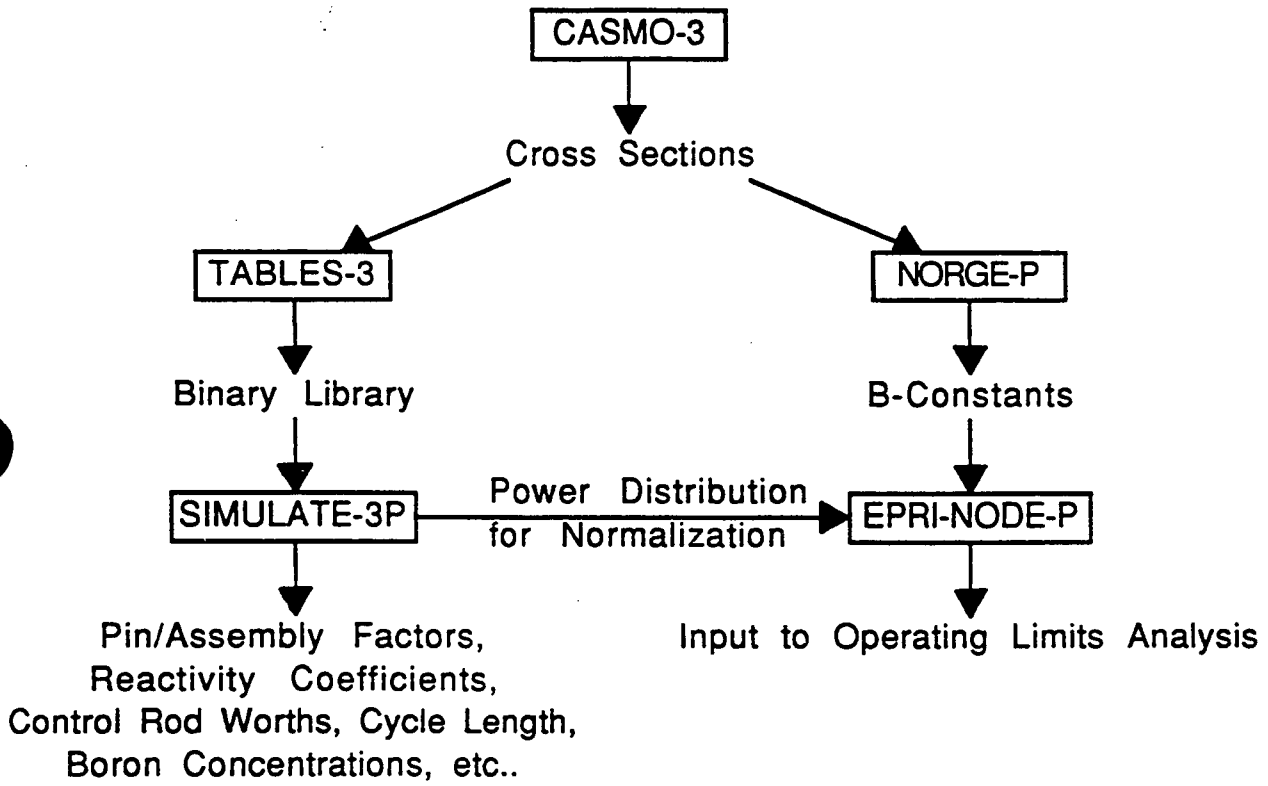
3.0   BENCHMARK COMPARISONS

This section provides comparisons of calculated and measured data
resulting from Duke Power Company's benchmark analysis used to verify the
adequacy of the SIMULATE-3P model. Comparisons are provided to measured
data from zero power physics testing and operating data for McGuire,
Catawba, and Oconee fuel cycles. Five cycles at McGuire, seven at
Catawba, and eight at Oconee were analyzed for a total of twenty cycles
including both initial and reload cores. For each parameter compared, a
brief description of the measurement technique utilized and the results
obtained is given below. Where applicable, the mean and standard
deviation of the differences between measurement and calculation are
provided by operating unit and for all comparisons combined. The mean
and standard deviation are defined as:

$$\text{Mean} = \bar{x} = \frac{\sum_{i=1}^{n} x_i}{n} \tag{3-1}$$

$$\text{Standard Deviation} = S = \sqrt{\frac{\sum_{i=1}^{n}(\bar{x}-x_i)^2}{n-1}} \tag{3-2}$$

where:

x_i = value for the ith observation

n = number of observations

3.1   Critical Boron Concentrations

3.1(a) 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.

3.1(b) Comparison of Results

Calculated and measured BOC, HZP, All-Rods-Out (ARO) critical boron concentrations are compared in Table 3-1. Calculated and measured HFP critical boron concentrations are compared in Tables 3-2 through 3-6. Plots of the difference between calculated and measured boron letdown versus burnups are shown in Figures 3-1 through 3-5.

3.2 Control Rod Worths

3.2(a) Measurement Technique

Control rod bank worths are measured at BOC, HZP conditions as part
of each cycle's startup physics testing. There are two basic ways
this is done. The first is 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. This technique was used for McGuire Unit 2 Cycle 1,
Catawba Unit 1 Cycle 1, and all Oconee cycles benchmarked.

The second method of measuring control rod bank worths is the rod
swap technique. In this technique the largest predicted (reference)
bank worth is measured by the boron swap technique. As the other
banks are inserted and withdrawn, their reactivity is compensated by
movement of the reference bank. Worths of these banks may then be
inferred relative to the reference bank worth, with allowances for
spatial redistribution due to the insertion of the bank being
measured. A more complete description of this method may be found
in Reference 6. This technique was used for all McGuire and Catawba
cycles except McGuire Unit 2 Cycle 1 and Catawba Unit 1 Cycle 1.
The measurement uncertainty for control rod worths is due to (1)
measured boron concentration errors, (2) Beta-effective data
utilized in the reactivity computer,

(3) control rod position uncertainty, and (4) the effect of spatial flux redistribution on the flux incident on the excore detectors.

3.2(b) Comparison of Results

Calculated and measured control rod worths are compared in Tables 3-7 through 3-11. These results are shown in terms of reactivity. The means and standard deviations of the percent differences between calculated and measured worths are listed in Table 3-12. The mean and standard deviation for all cycles combined are -1.7% and 6.4%, respectively.

3.3 Isothermal Temperature Coefficient

3.3(a) Measurement Technique

The HZP Isothermal Temperature Coefficient (ITC) is measured by executing average moderator temperature changes from a plateau of initial equilibrium critical conditions. After the change, steady state conditions are established and pertinent data are recorded by the reactivity computer at the resulting plateau. The HZP ITC is determined as the change in reactivity between plateaus divided by the change in temperature. This coefficient is calculated using both an increase and a decrease in average moderator temperature. The value reported is an average of these two measurements.

The McGuire and Catawba plants also measure the HFP ITC near the end of each cycle. The HFP ITC is measured by executing a change in average moderator temperature while maintaining control rod positions and power level. The reactivity change is compensated by a change in the soluble boron concentration. The boron concentration and moderator temperature are recorded following equilibrium at each plateau. The temperature is then returned to its initial value with a corresponding change in soluble boron concentration. The change in boron concentration is converted to reactivity using the calculated HFP boron worth. The HFP ITC is then reported as the average change in reactivity divided by the average change in moderator temperature.

The measurement uncertainty for HZP ITCs is due to (1) core average temperature measurement uncertainty and (2) Beta-effective data utilized in the reactivity computer. The measurement uncertainty for HFP ITCs is due to (1) measured boron concentration errors, (2) core average temperature measurement uncertainty and (3) calculated HFP boron worth uncertainties, and (4) flux and Xenon redistribution effects.

3.3(b) Comparison of Results

Calculated and measured HZP ITCs for all cycles analyzed are compared in Table 3-13. The mean of the differences for all cycles combined is 0.6 PCM/^{o}F with a standard deviation of 0.8 PCM/^{o}F.

Calculated and measured HFP ITCs for all McGuire and Catawba cycles analyzed are compared in Table 3-14. McGuire Unit 2, Cycle 1 data utilized a different measurement technique and is not included in this comparison. The mean of the differences for all cycles combined is 2.2 PCM/^{o}F with a standard deviation of 2.8 PCM/^{o}F.

3.4 Power Distributions

3.4(a) Measurement Technique

Measured radial and axial relative power distributions are determined from the appropriate incore detector system. Due to the quantity of data available, power distribution comparisons were only performed for a subset of the total cycles analyzed. For the McGuire and Catawba plants the incore detector system consists of movable incore fission chambers. The fission chambers measure a reaction rate which is proportional to the local flux in the instrument guide tubes. Cycle-specific calculated values of assembly power to reaction rate ratios are then used to convert the signals to relative assembly powers. A more detailed discussion of this system is included in Reference 3.

For the Oconee plants the incore detector system consists of fixed Rhodium emitters. The emitter current is proportional to the local flux in the instrument tube. The signals are corrected for Rhodium depletion, background current, and the specific core configuration.

Generic ratios of power in the surrounding pins to detector reaction rates are used to convert the corrected signals to "8 pin powers." Cycle-specific calculated values of assembly power to "8 pin powers" are then used to determine relative assembly powers. A more detailed discussion of this system is included in Reference 1.

3.4(b) Comparison of Results

Representative comparisons of calculated and measured relative radial assembly powers at approximately beginning, middle, and end of cycle for the most recently analyzed cycle at each station are provided in Figures 3-6 through 3-14. The comparisons shown here are only for qualitative purposes. A more detailed comparison, including the required statistical analysis is provided in Section 4.2.

Calculated and measured values of Axial Offset (A-0), as defined in (3-3) below, are compared in Table 3-15 for those cycles used to generate the uncertainties in Section 4.2.

$$A\text{-}0 = (P_T - P_B) / (P_T + P_B) \qquad (3\text{-}3)$$

where,

P_T = Power in the Top Half of the Core

P_B = Power in the Bottom Half of the Core

The means and standard deviations of the differences between calculated and measured A-Os are listed in Table 3-16. The mean and standard deviation for all cycles combined are 0.010 and 0.012 respectively.

Table 3-1

BOC, HZP, ARO CRITICAL BORON CONCENTRATIONS

McGuire

Unit	Cycle	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
2	1	1325	1295	30
2	2	1483	1413	70
2	3	1450	1379	71
2	4	1554	1499	55
2	5	1507	1460	47

Catawba

Unit	Cycle	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
1	1	964	975	-11
1	2	1404	1405	-1
1	3	1481	1441	40
1	4	1613	1570	43
2	1	964	975	-11
2	2	1389	1349	40
2	3	1462	1400	62

Oconee

Unit	Cycle	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
2	7	1583	1562	21
2	8	1602	1593	9
2	9	1547	1535	12
2	10	1595	1569	26
3	8	1574	1563	11
3	9	1617	1641	-24
3	10	1715	1709	6
3	11	1624	1606	18

Difference = SIMULATE-3P - Measured

Table 3-2

CATAWBA UNIT 1
HFP,ARO BORON LETDOWN

CYCLE 1

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.453	592	610	-18
0.921	597	621	-24
1.499	599	610	-11
2.898	567	598	-31
3.999	530	559	-29
5.159	477	506	-29
5.998	438	461	-23
6.862	393	415	-22
7.390	366	384	-18
7.930	338	367	-29
8.941	280	296	-16
9.370	255	270	-15
10.338	198	207	-9
12.305	75	72	3

CYCLE 2

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.544	945	918	27
0.963	908	882	26
1.786	842	832	10
2.616	774	764	10
3.489	703	687	16
4.285	637	623	14
5.399	546	538	8
6.272	474	466	8
7.348	387	376	11
8.444	301	292	9
9.766	198	184	14
10.712	126	104	22

Difference = SIMULATE-3P - Measured

Table 3-2(cont'd.)

CATAWBA UNIT 1
HFP,ARO BORON LETDOWN

CYCLE 3

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.173	1053	1003	50
0.704	1003	960	43
1.401	950	908	42
1.941	909	880	29
2.987	826	803	23
4.098	736	730	6
5.198	647	650	-3
6.448	546	548	-2
7.420	469	465	4
8.088	416	406	10
8.486	385	358	27
8.776	362	336	26
9.826	281	252	29
10.572	223	200	23

Table 3-3

CATAWBA UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 1

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
1.232	599	625	-26
1.860	598	620	-22
2.297	588	617	-29
3.471	548	587	-39
4.476	508	541	-33
5.689	452	487	-35
6.983	387	424	-37
8.096	328	355	-27
9.295	260	285	-25
10.648	179	194	-15
11.656	116	126	-10
12.723	48	51	-3

CYCLE 2

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.178	973	943	30
0.772	912	893	19
1.150	880	855	25
1.529	850	831	19
1.830	826	807	19
2.022	811	794	17
2.782	750	736	14
4.048	648	627	21
5.158	558	548	10
6.473	452	449	3
7.723	352	341	11
8.939	258	231	27
9.987	177	146	31
11.275	79	45	34

Difference = SIMULATE-3P - Measured

Table 3-4

MCGUIRE UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 1

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
1.079	863	846	17
1.865	859	836	23
2.119	853	837	16
3.142	809	794	15
3.872	778	758	20
4.929	720	704	16
5.963	662	652	10
7.075	593	582	11
7.926	541	520	21
8.737	485	461	24
9.257	449	429	20
9.818	411	389	22
10.517	359	337	22
11.248	304	284	20
13.327	146	125	21
14.261	73	38	35
14.582	48	11	37

Difference = SIMULATE-3P - Measured

Table 3-4(cont'd.)

MCGUIRE UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 2

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.182	1002	958	44
0.975	915	857	58
1.131	900	851	49
1.250	888	840	48
1.407	873	835	38
2.112	806	760	46
2.559	764	723	41
2.810	740	695	45
3.270	696	654	42
4.215	606	571	35
4.759	554	522	32
5.263	507	482	25
6.086	429	389	40
6.717	372	334	38
7.410	310	261	49
8.438	219	169	50

Table 3-4(cont'd.)

MCGUIRE UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 3

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.232	992	928	64
0.478	968	900	68
0.718	949	880	69
0.915	934	881	53
1.386	898	846	52
1.887	860	814	46
2.007	851	808	43
2.046	848	799	49
2.208	835	794	41
2.475	812	772	40
3.003	768	726	42
4.132	674	641	33
4.890	610	579	31
5.743	538	503	35
6.914	440	410	30
7.740	372	338	34
7.943	355	318	37
9.023	268	231	37
10.192	174	135	39

Table 3-4(cont'd.)

MCGUIRE UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 4

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.240	1129	1067	62
0.981	1062	1005	57
1.408	1027	979	48
1.532	1017	969	48
2.466	940	908	32
3.547	848	813	35
4.409	776	748	28
5.481	686	669	17
6.009	641	608	33
7.416	524	495	29
8.530	433	390	43
9.562	349	308	41
10.803	249	199	50
11.956	157	103	54

CYCLE 5

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.469	1051	1008	43
1.047	1006	959	47
1.256	990	943	47
1.510	972	932	40
2.322	910	882	28
3.358	826	807	19
4.192	759	734	25
5.190	678	664	14
6.392	581	563	18
7.591	485	465	20
8.726	394	364	30
9.456	336	309	27
10.276	271	235	36
10.895	223	188	35
11.704	160	116	44
12.830	73	24	49

Table 3-5

OCONEE UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 7

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.376	1067	1037	30
0.783	1045	1013	32
1.565	1003	980	23
3.130	899	894	5
4.695	777	775	2
6.261	645	644	1
7.826	508	512	-4
9.391	367	372	-5
10.956	225	231	-6
12.521	85	97	-12
13.492	0	17	-17

CYCLE 8

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.845	1132	1116	16
1.158	1117	1109	8
1.753	1018	994	24
2.723	952	952	0
3.537	882	887	-5
4.539	805	811	-6
5.478	726	742	-16
6.417	649	668	-19
7.387	566	575	-9
8.233	491	499	-8
9.172	405	413	-8
10.111	319	326	-7
11.081	232	243	-11
12.020	150	160	-10
12.959	68	75	-7
13.366	35	42	-7

Difference = SIMULATE-3P - Measured

Table 3-5(cont'd.)

OCONEE UNIT 2
HFP,ARO BORON LETDOWN

CYCLE 9

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.313	1041	1023	18
1.221	992	978	14
2.066	931	945	-14
2.911	860	888	-28
3.819	793	828	-35
4.727	714	751	-37
5.384	682	722	-40
6.229	606	646	-40
7.043	537	585	-48
7.888	462	509	-47
8.640	398	445	-47
9.453	323	362	-39
10.267	245	280	-35
10.956	178	218	-40
11.895	116	161	-45
12.772	33	66	-33

CYCLE 10

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.501	1050	1038	12
1.440	988	988	0
2.379	922	932	-10
3.036	869	886	-17
4.007	790	815	-25
4.946	709	743	-34
5.916	623	656	-33
6.855	539	577	-38
7.826	450	487	-37
8.765	362	402	-40
9.610	284	319	-35
10.580	197	228	-31
11.394	122	155	-33
11.989	69	100	-31

Table 3-6

OCONEE UNIT 3
HFP,ARO BORON LETDOWN

CYCLE 8

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.939	1023	992	31
1.909	966	958	8
2.817	900	898	2
3.756	826	830	-4
4.727	745	757	-12
5.228	707	705	2
6.198	623	622	1
7.168	537	540	-3
8.045	459	471	-12
8.452	423	427	-4
10.518	236	236	0
11.426	152	154	-2
11.832	115	118	-3
12.396	122	118	4

CYCLE 9

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.313	1106	1097	9
0.751	1076	1072	4
1.346	1043	1036	7
2.222	985	1000	-15
3.099	920	931	-11
4.101	843	878	-35
5.008	767	774	-7
5.979	680	703	-23
6.887	605	635	-30
7.794	520	533	-13
8.671	445	457	-12
9.641	353	368	-15
10.518	344	357	-13
10.893	307	326	-19

Difference = SIMULATE-3P - Measured

Table 3-6(cont'd.)

OCONEE UNIT 3
HFP,ARO BORON LETDOWN

CYCLE 10

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
1.096	1154	1143	11
2.066	1095	1102	-7
3.036	1019	1036	-17
4.007	940	961	-21
4.915	864	888	-24
5.916	774	798	-24
6.855	692	715	-23
7.826	602	629	-27
8.765	513	531	-18
9.673	429	448	-19
10.643	338	351	-13
11.112	322	337	-15
11.801	251	258	-7
12.709	167	170	-3
13.711	75	79	-4
14.024	49	52	-3

CYCLE 11

Exposure (GWD/MTU)	SIMULATE-3P(PPMB)	Measured(PPMB)	Difference(PPMB)
0.188	1103	1080	23
1.158	1031	1017	14
2.035	970	975	-5
3.005	892	907	-15
3.882	818	835	-17
4.789	741	761	-20
5.728	655	680	-25
6.667	569	597	-28
7.638	479	511	-32
8.577	390	417	-27

Table 3-7

CATAWBA UNIT 1
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
1	D	784	788	-0.5
1	C	1188	1203	-1.2
1	B	1233	1171	5.3
1	A	504	548	-8.0
1	SE	427	460	-7.2
1	SD	760	772	-1.6
1	SC	1079	1099	-1.8
1	TOTAL	5975	6041	-1.1

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
2	D	459	458	0.2
2	C	1010	1014	-0.4
2	B	714	718	-0.6
2	A	395	382	3.4
2	SE	437	406	7.6
2	SD	348	348	0.0
2	SC	347	352	-1.4
2	SB	811	787	3.0
2	SA	351	366	-4.1
2	TOTAL	4872	4831	0.8

$$\% \text{ Difference} = \frac{\text{SIMULATE-3P} - \text{Measured}}{\text{Measured}} \times 100$$

Table 3-7(cont'd.)

CATAWBA UNIT 1
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
3	D	546	559	-2.3
3	C	769	743	3.5
3	B	864	936	-7.7
3	A	279	251	11.2
3	SE	417	392	6.4
3	SD.	429	459	-6.5
3	SC	430	458	-6.1
3	SB	900	895	0.6
3	SA	341	378	-9.8
3	TOTAL	4975	5071	-1.9

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
4	D	554	572	-3.1
4	C	1046	1045	0.1
4	B	703	817	-14.0
4	A	394	365	7.9
4	SE	499	490	1.8
4	SD	375	418	-10.3
4	SC	375	421	-10.9
4	SB	855	919	-7.0
4	SA	261	325	-19.7
4	TOTAL	5062	5372	-5.8

Table 3-8

CATAWBA UNIT 2
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
1	D	774	795	-2.6
1	C	805	850	-5.3
1	B	889	882	0.8
1	A	242	261	-7.3
1	SE	376	403	-6.7
1	SD	522	524	-0.4
1	SC	522	521	0.2
1	SB	790	837	-5.6
1	SA	716	708	1.1
1	TOTAL	5636	5781	-2.5

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
2	D	465	478	-2.7
2	C	984	988	-0.4
2	B	743	799	-7.0
2	A	368	365	0.8
2	SE	421	411	2.4
2	SD	363	376	-3.5
2	SC	363	371	-2.2
2	SB	815	831	-1.9
2	SA	373	411	-9.2
2	TOTAL	4895	5030	-2.7

$$\% \text{ Difference} = \frac{\text{SIMULATE-3P} - \text{Measured}}{\text{Measured}} \times 100$$

Table 3-8(cont'd.)

CATAWBA UNIT 2
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
3	D	566	585	-3.2
3	C	1083	1106	-2.1
3	B	612	649	-5.7
3	A	317	290	9.3
3	SE	458	438	4.6
3	SD	359	379	-5.3
3	SC	360	382	-5.8
3	SB	858	869	-1.3
3	SA	289	328	-11.9
3	TOTAL	4902	5026	-2.5

Table 3-9

MCGUIRE UNIT 2
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
1	D	656	664	-1.2
1	C	1269	1283	-1.1
1	B	1053	1105	-4.7
1	A	677	678	-0.1
1	SE	853	853	0.0
1	SD	759	771	-1.6
1	SC	1012	1026	-1.4
1	TOTAL	6279	6380	-1.6

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
2	D	642	668	-3.9
2	C	931	871	6.9
2	B	445	490	-9.2
2	A	489	426	14.8
2	SE	323	279	15.8
2	SD	362	386	-6.2
2	SC	361	388	-7.0
2	SB	504	487	3.5
2	SA	371	444	-16.4
2	TOTAL	4428	4439	-0.2

$$\% \text{ Difference} = \frac{\text{SIMULATE-3P} - \text{Measured}}{\text{Measured}} \times 100$$

Table 3-9(cont'd.)

MCGUIRE UNIT 2
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
3	D	662	624	6.1
3	C	865	787	9.9
3	B	646	654	-1.2
3	A	349	305	14.4
3	SE	483	428	12.9
3	SD	403	414	-2.7
3	SC	398	400	-0.5
3	SB	885	842	5.1
3	SA	348	387	-10.1
3	TOTAL	5039	4841	4.1

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
4	D	634	643	-1.4
4	C	796	717	11.0
4	B	722	750	-3.7
4	A	356	288	23.6
4	SE	515	461	11.7
4	SD	465	478	-2.7
4	SC	459	472	-2.8
4	SB	920	899	2.3
4	SA	307	343	-10.5
4	TOTAL	5174	5051	2.4

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
5	D	619	637	-2.8
5	C	855	826	3.5
5	B	679	699	-2.9
5	A	364	329	10.6
5	SE	433	410	5.6
5	SD	439	466	-5.8
5	SC	441	463	-4.8
5	SB	800	748	7.0
5	SA	304	328	-7.3
5	TOTAL	4934	4906	0.6

Table 3-10

OCONEE UNIT 2
BOC. HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
7	7	1290	1420	-9.2
7	6	1000	1040	-3.8
7	5	1130	1110	1.8
7	TOTAL	3420	3570	-4.2

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
8	7	960	970	-1.0
8	6	840	870	-3.4
8	5	1170	1200	-2.5
8	TOTAL	2970	3040	-2.3

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
9	7	970	1010	-4.0
9	6	1080	1120	-3.6
9	5	1330	1440	-7.6
9	TOTAL	3380	3570	-5.3

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
10	7	880	920	-4.3
10	6	880	910	-3.3
10	5	1250	1350	-7.4
10	TOTAL	3010	3180	-5.3

$$\% \text{ Difference} = \frac{\text{SIMULATE-3P} - \text{Measured}}{\text{Measured}} \times 100$$

Table 3-11

OCONEE UNIT 3
BOC, HZP CONTROL ROD WORTHS

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
8	7	1170	1240	-5.6
8	6	990	1070	-7.5
8	5	1070	1030	3.9
8	TOTAL	3230	3340	-3.3

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
9	7	1110	1150	-3.5
9	6	860	900	-4.4
9	5	1350	1410	-4.3
9	TOTAL	3320	3460	-4.0

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
10	7	860	900	-4.4
10	6	1110	1160	-4.3
10	5	1280	1480	-13.5
10	TOTAL	3250	3540	-8.2

Cycle	Bank	SIMULATE-3P(PCM)	Measured(PCM)	% Difference
11	7	890	960	-7.3
11	6	1000	1030	-2.9
11	5	1240	1380	-10.1
11	TOTAL	3130	3370	-7.1

$$\% \text{ Difference} = \frac{\text{SIMULATE-3P} - \text{Measured}}{\text{Measured}} \times 100$$

Table 3-12

SUMMARY STATISTICS FOR
BOC, HZP CONTROL ROD WORTHS

Station	Unit	Mean % Difference	Standard Deviation of % Differences
Catawba	1	-2.1	6.3
Catawba	2	-2.6	4.2
McGuire	2	1.2	7.9
Oconee	2	-4.1	2.6
Oconee	3	-5.4	3.8
All	All	-1.7	6.4

Table 3-13

BOC. HZP.ARO ISOTHERMAL TEMPERATURE COEFFICIENTS

McGuire

Unit	Cycle	SIMULATE-3P(PCM/°F)	Measured(PCM/°F)	Difference(PCM/°F)
2	1	-0.7	-1.4	0.7
2	2	-1.0	-1.7	0.7
2	3	0.2	-0.6	0.8
2	4	3.8	2.4	1.4
2	5	2.9	1.9	1.0

Catawba

Unit	Cycle	SIMULATE-3P(PCM/°F)	Measured(PCM/°F)	Difference(PCM/°F)
1	1	-1.7	-1.7	0.0
1	2	3.2	1.7	1.5
1	3	2.5	1.3	1.2
1	4	4.3	2.8	1.5
2	1	-1.6	-1.8	0.2
2	2	2.4	1.7	0.7
2	3	1.3	0.3	1.0

Oconee

Unit	Cycle	SIMULATE-3P(PCM/°F)	Measured(PCM/°F)	Difference(PCM/°F)
2	7	0.3	-0.8	1.1
2	8	1.0	2.9	-1.9
2	9	-0.1	-0.3	0.2
2	10	0.0	0.4	-0.4
3	8	0.2	-0.5	0.7
3	9	1.1	0.0	1.1
3	10	2.0	1.9	0.1
3	11	0.3	0.1	0.2

Difference = SIMULATE-3P - Measured

Mean Difference = 0.6
Standard Deviation = 0.8

Table 3-14

EOC, HFP ISOTHERMAL TEMPERATURE COEFFICIENTS

McGuire

Unit	Cycle	SIMULATE-3P(PCM/°F)	Measured(PCM/°F)	Difference(PCM/°F)
2	2	-25.1	-24.1	-1.0
2	3	-24.3	-22.8	-1.5
2	4	-22.7	-23.6	0.9
2	5	-22.5	-28.2	5.7

Catawba

Unit	Cycle	SIMULATE-3P(PCM/°F)	Measured(PCM/°F)	Difference(PCM/°F)
1	1	-16.5	-22.3	5.8
1	2	-21.3	-23.7	2.4
1	3	-22.4	-27.1	4.7
2	1	-16.3	-16.4	0.1
2	2	-20.6	-23.2	2.6

Difference = SIMULATE-3P - Measured

Mean Difference = 2.2
Standard Deviation = 2.8

Table 3-15

HFP AXIAL OFFSET COMPARISONS

CATAWBA 1 CYCLE 3

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
1.077	0.019	0.000	0.019
2.062	0.008	-0.009	0.017
3.141	-0.008	-0.022	0.014
4.341	-0.023	-0.025	0.002
6.638	-0.031	-0.047	0.016
7.657	-0.025	-0.036	0.011
8.890	-0.027	-0.039	0.012
9.940	-0.034	-0.046	0.012
10.798	-0.032	-0.044	0.012

CATAWBA 2 CYCLE 2

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
0.174	0.070	0.053	0.017
0.811	0.041	0.022	0.019
1.981	0.010	0.000	0.010
2.527	-0.003	-0.012	0.009
3.711	-0.015	-0.022	0.007
4.839	-0.017	-0.032	0.015
5.974	-0.026	-0.038	0.012
7.186	-0.034	-0.040	0.006
8.071	-0.023	-0.041	0.018
8.931	-0.030	-0.041	0.011
9.981	-0.014	-0.031	0.017
10.856	-0.028	-0.046	0.018

Difference = SIMULATE-3P - Measured

Table 3-15(cont'd.)

HFP AXIAL OFFSET COMPARISONS

MCGUIRE 2 CYCLE 4

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
0.249	0.060	0.042	0.018
1.418	0.029	0.016	0.013
3.019	0.001	-0.010	0.011
4.418	-0.021	-0.026	0.005
6.219	-0.030	-0.039	0.009
7.415	-0.032	-0.042	0.010
8.653	-0.026	-0.038	0.012
9.642	-0.043	-0.059	0.016
10.804	-0.027	-0.049	0.022
11.964	-0.021	-0.047	0.026

MCGUIRE 2 CYCLE 5

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
0.471	0.036	0.034	0.002
1.789	0.012	0.002	0.010
2.987	-0.003	-0.008	0.005
4.206	-0.016	-0.027	0.011
5.194	-0.026	-0.034	0.008
6.360	-0.030	-0.048	0.018
7.578	-0.024	-0.044	0.020
8.724	-0.036	-0.051	0.015
9.098	-0.035	-0.049	0.014
9.455	-0.041	-0.050	0.009
10.281	-0.028	-0.047	0.019
11.421	-0.021	-0.047	0.026
13.082	-0.015	0.015	-0.030

Table 3-15(cont'd.)

HFP AXIAL OFFSET COMPARISONS

OCONEE 2 CYCLE 8

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
0.857	-0.014	-0.014	0.000
1.158	-0.011	-0.007	-0.004
1.758	-0.032	-0.028	-0.004
2.723	-0.025	-0.034	0.009
3.539	-0.036	-0.053	0.017
4.534	-0.024	-0.056	0.032
6.411	-0.018	-0.047	0.029
7.381	-0.015	-0.043	0.028
8.220	-0.011	-0.038	0.027
9.179	-0.014	-0.028	0.014
10.113	-0.023	-0.029	0.006
11.093	-0.015	-0.016	0.001
12.013	-0.016	-0.014	-0.002
12.948	-0.010	0.007	-0.017
13.428	-0.007	0.011	-0.018

OCONEE 2 CYCLE 9

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
0.319	-0.017	0.002	-0.019
1.211	-0.020	-0.013	-0.007
2.069	-0.008	0.012	-0.020
2.908	-0.030	-0.017	-0.013
3.825	-0.015	-0.018	0.003
4.733	-0.021	-0.019	-0.002
5.372	-0.007	-0.011	0.004
6.236	-0.009	-0.012	0.003
7.046	-0.009	-0.011	0.002
7.885	-0.007	-0.015	0.008
8.627	0.003	-0.009	0.012
9.422	-0.008	-0.022	0.014
10.274	-0.011	-0.031	0.020
10.968	-0.012	-0.021	0.009
11.896	0.021	0.033	-0.012
12.797	0.006	0.002	0.004

Table 3-15(cont'd.)

HFP AXIAL OFFSET COMPARISONS

OCONEE 3 CYCLE 9

Exposure (GWD/MTU)	SIMULATE-3P	– Measured	Difference
0.320	-0.050	-0.040	-0.010
0.746	-0.057	-0.047	-0.010
1.337	-0.057	-0.034	-0.023
2.236	-0.045	-0.055	0.010
3.104	-0.033	-0.054	0.021
4.087	-0.021	-0.048	0.027
5.006	-0.020	-0.046	0.026
6.885	-0.010	-0.036	0.026
7.803	-0.019	-0.048	0.029
8.670	-0.011	-0.030	0.019
9.648	-0.021	-0.032	0.011
10.510	0.015	0.001	0.014
10.894	0.008	-0.007	0.015

OCONEE 3 CYCLE 10

Exposure (GWD/MTU)	SIMULATE-3P	Measured	Difference
1.102	-0.020	-0.026	0.006
3.039	-0.020	-0.027	0.007
4.010	-0.020	-0.034	0.014
4.918	-0.013	-0.030	0.017
5.923	-0.019	-0.037	0.018
6.858	-0.013	-0.028	0.015
7.827	-0.017	-0.032	0.015
8.768	-0.020	-0.033	0.013
9.682	-0.011	-0.027	0.016
10.646	-0.016	-0.036	0.020
11.109	0.017	-0.007	0.024
11.795	0.017	-0.002	0.019
12.703	0.008	-0.006	0.014
13.708	-0.004	-0.015	0.011
14.031	0.002	-0.010	0.012

Table 3-16

SUMMARY STATISTICS FOR
HFP AXIAL OFFSETS

Station	Unit	Cycle	Mean Difference	Standard Deviation of Differences
Catawba	1	3	0.013	0.005
Catawba	2	2	0.013	0.005
McGuire	2	4	0.014	0.006
McGuire	2	5	0.010	0.014
Oconee	2	8	0.008	0.016
Oconee	2	9	0.000	0.012
Oconee	3	9	0.012	0.016
Oconee	3	10	0.015	0.005
All	All	All	0.010	0.012

Catawba Unit 1 HFP, ARO Boron Differences
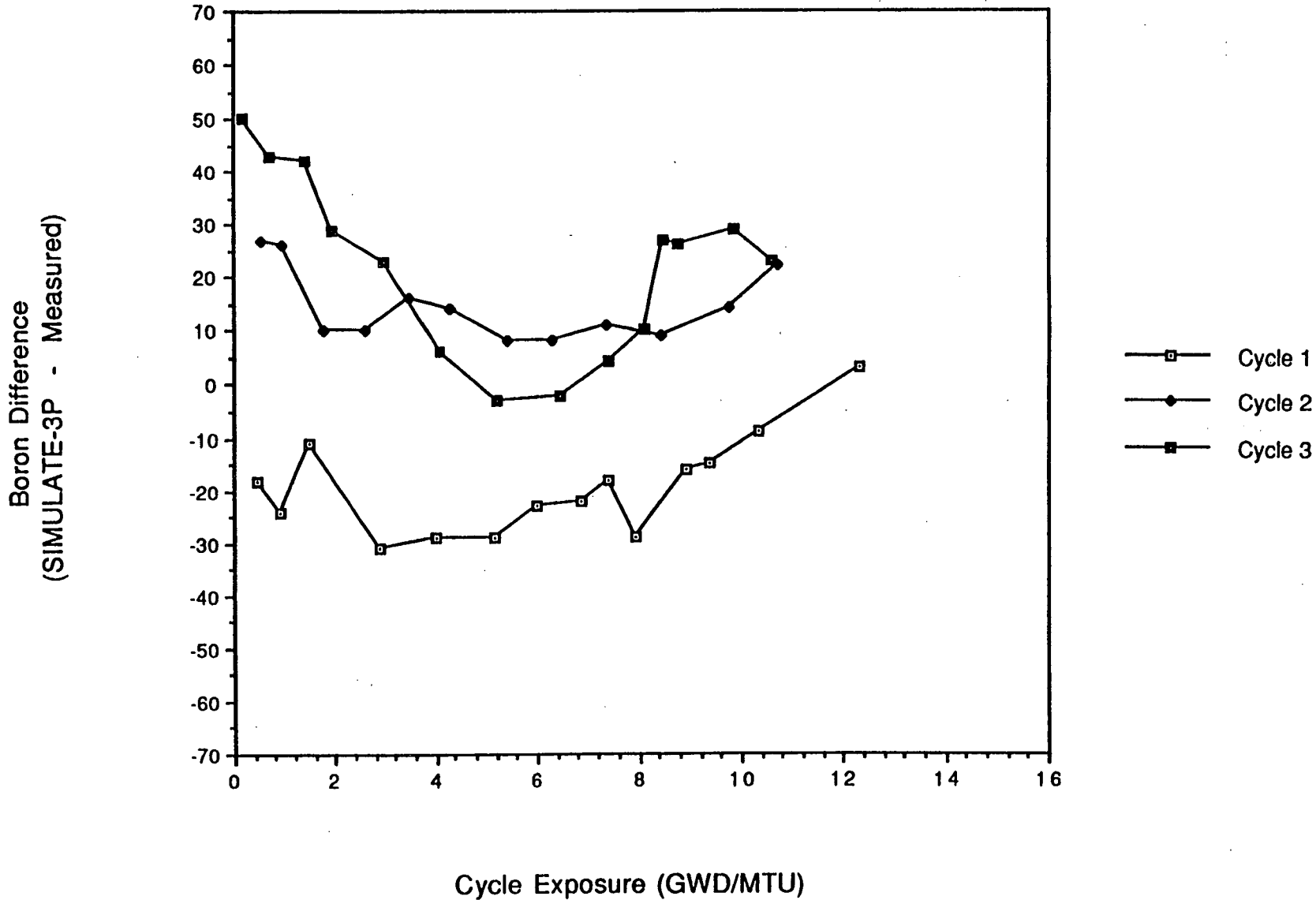
Figure 3-1

Cycle 1
Cycle 2
Cycle 3

Boron Difference
(SIMULATE-3P - Measured)

Cycle Exposure (GWD/MTU)

3-37

Catawba Unit 2 HFP, ARO Boron Differences

Figure 3-2

3-38

McGuire Unit 2 HFP, ARO Boron Differences

Boron Difference (SIMULATE-3P - Measured)

Cycle Exposure (GWD/MTU)
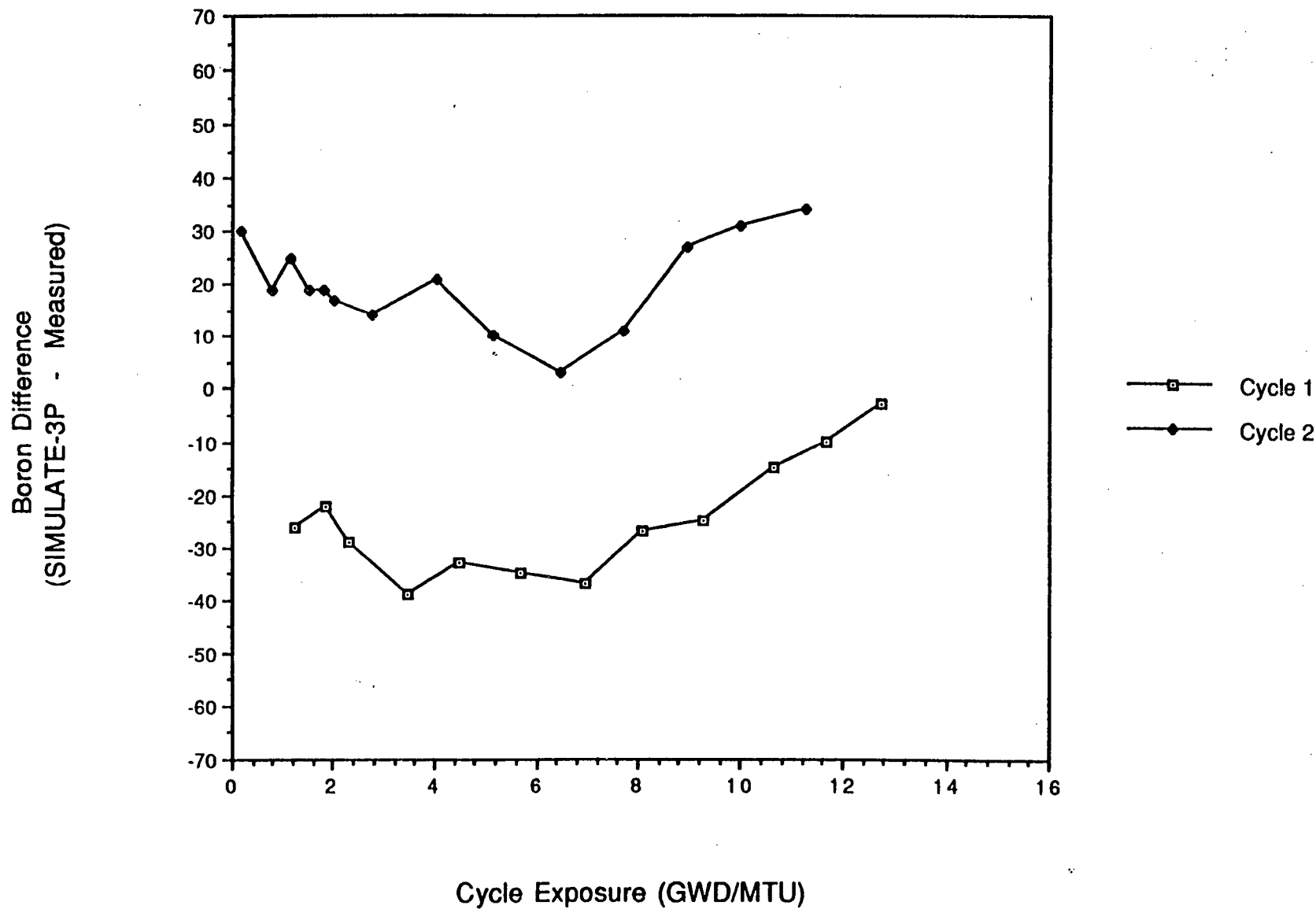
Cycle 1
Cycle 2
Cycle 3
Cycle 4
Cycle 5
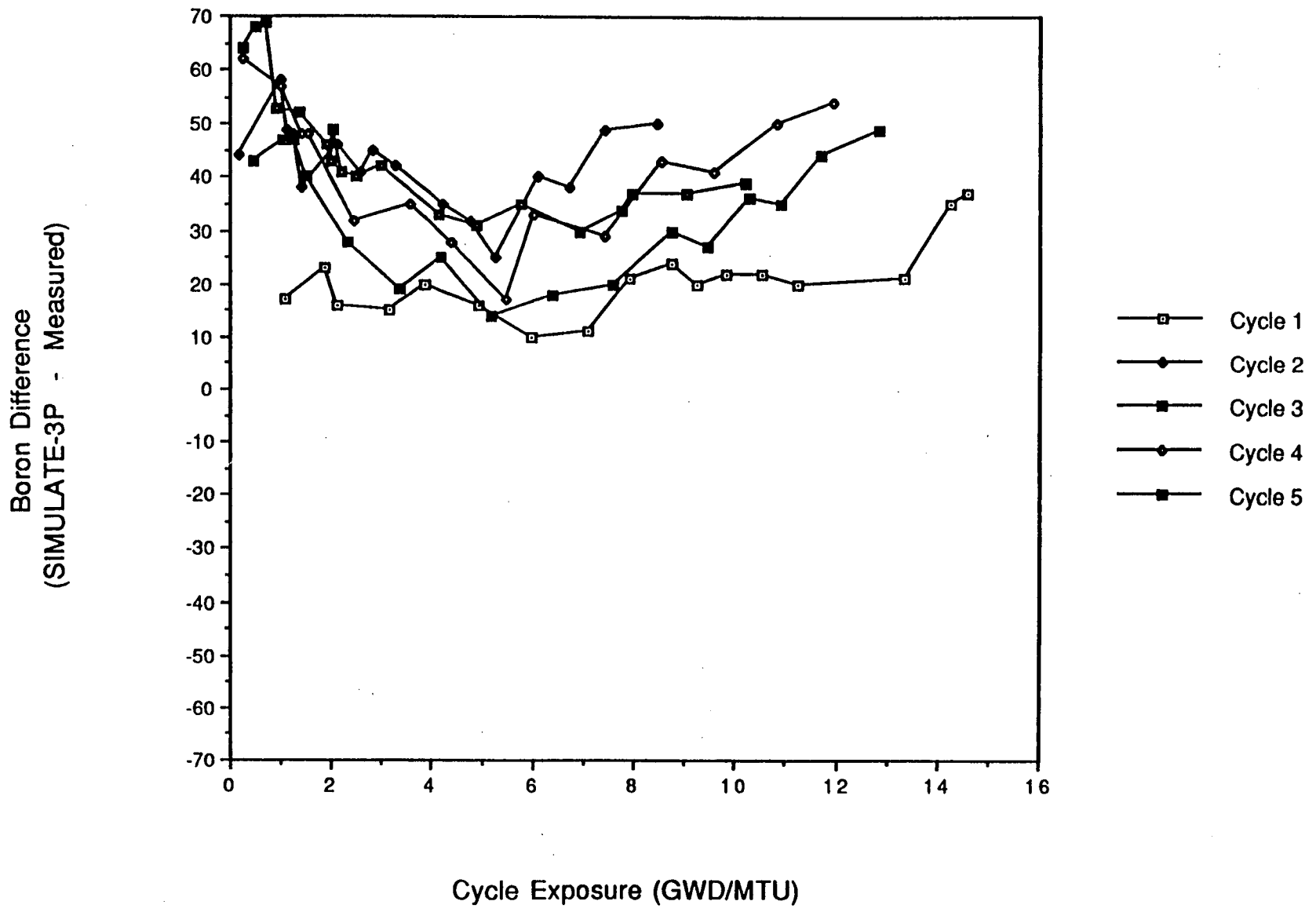
Figure 3-3

Oconee Unit 2 HFP, ARO Boron Differences

Boron Difference
(SIMULATE-3P - Measured)

Cycle Exposure (GWD/MTU)
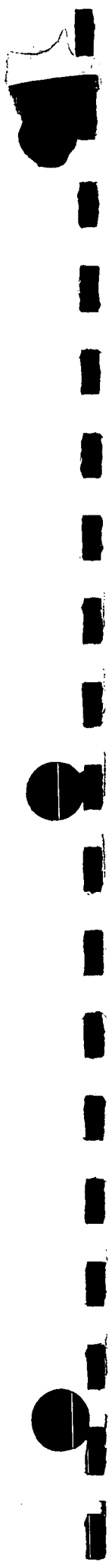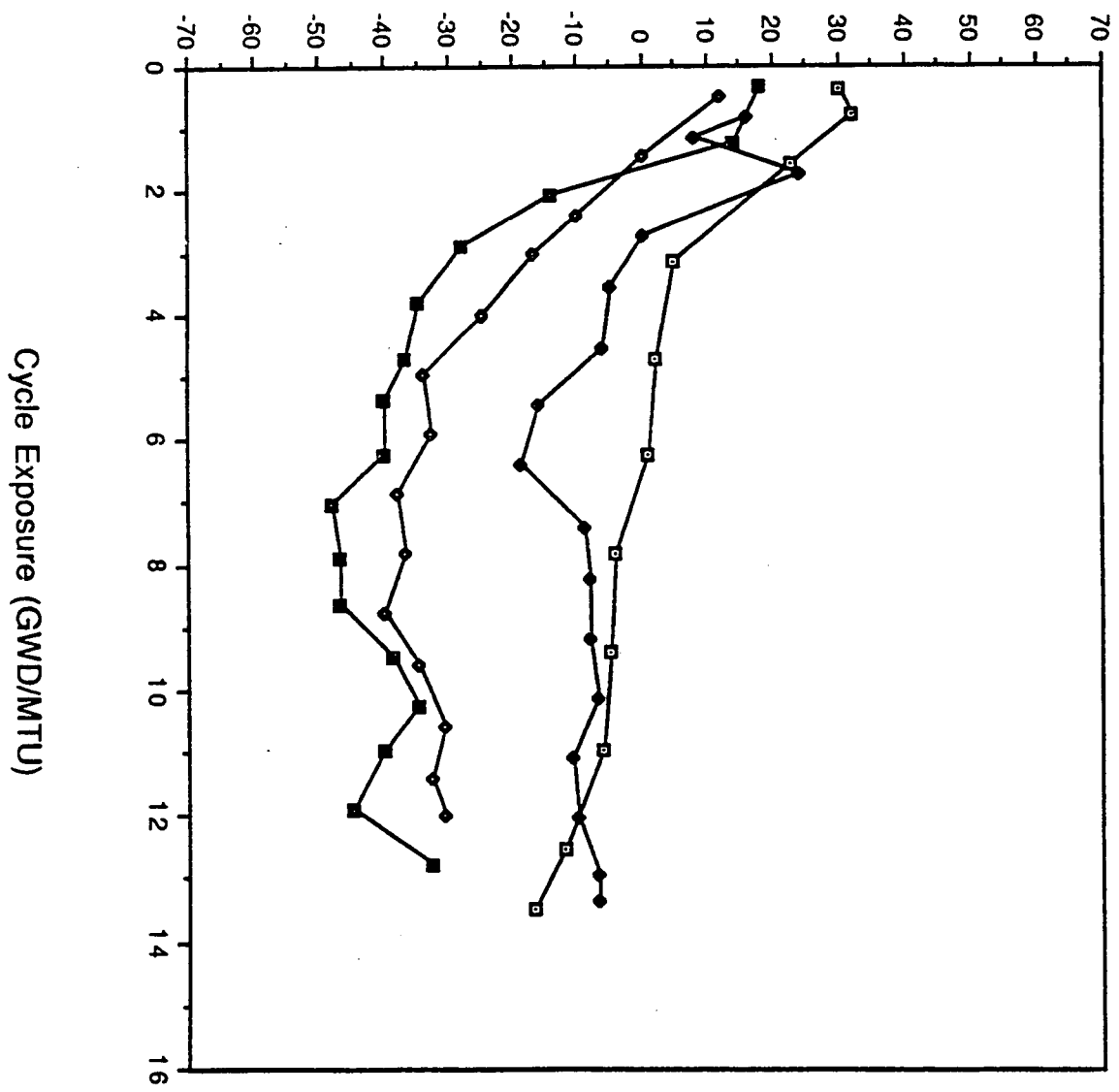
Cycle 7
Cycle 8
Cycle 9
Cycle 10

Figure 3-4

Oconee Unit 3 HFP, ARO Boron Differences

Boron Difference
(SIMULATE-3P - Measured)

Cycle Exposure (GWD/MTU)
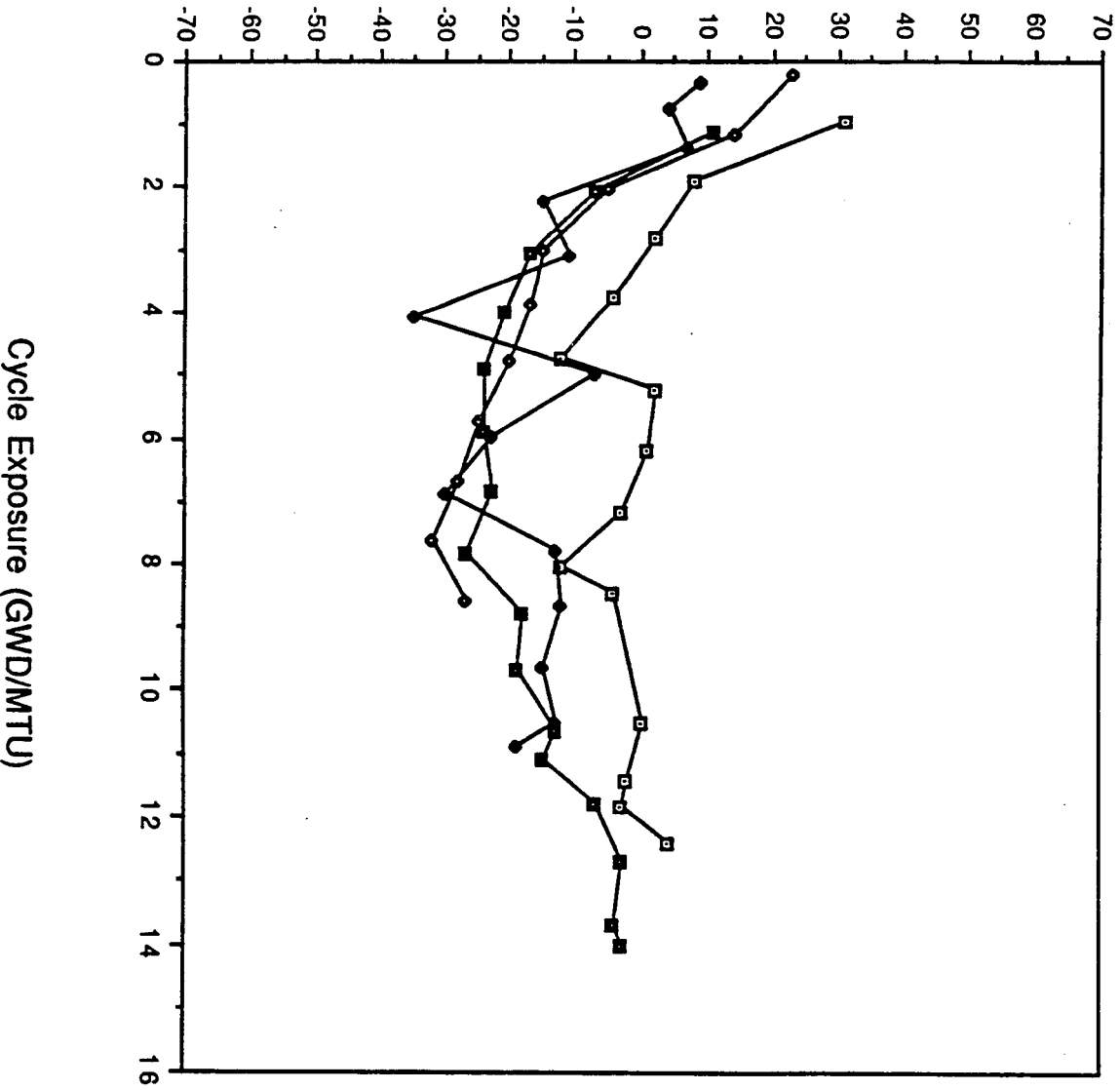
Cycle 8
Cycle 9
Cycle 10
Cycle 11

Figure 3-5

Figure 3-6

Catawba Unit 1 BOC-3
Assembly Power Distribution

```
        H           G           F           E           D           C           B           A

   ************************************************************************************************
   *    .95  *   1.25  *    .97  *  1.25  *    .96  *  1.18  *    .91  *    .85  *
8  *    .94  *   1.25  *   1.00  *  1.24  *    .96  *  1.18  *    .92  *    .84  *
   *   1.06  *    .00  *  -3.00  *   .81  *    .00  *    .00  *  -1.09  *   1.19  *
   ************************************************************************************************
             *    .97  *   1.12  *    .97  *  1.28  *   1.03  *  1.21  *    .76  *
          9  *    .98  *   1.12  *    .97  *  1.27  *   1.04  *  1.21  *    .76  *
             *  -1.02  *    .00  *    .00  *   .79  *   -.96  *    .00  *    .00  *
             ************************************************************************************
                       *    .94  *   1.27  *  1.00  *   1.24  *  1.11  *    .84  *
                   10  *    .93  *   1.25  *  1.01  *   1.25  *  1.14  *    .83  *
                       *   1.08  *   1.60  *  -.99  *   -.80  *  -2.63  *  1.20  *
                       ****************************************************************************
                                 *    .99  *  1.28  *   1.00  *  1.11  *    .50  *
                             11  *   1.00  *  1.28  *   1.03  *  1.12  *    .50  *
                                 *  -1.00  *   .00  *  -2.91  *  -.89  *    .00  *
                                 **********************************************************
                                           *    .95  *  1.19  *    .67  *
                                       12  *    .96  *  1.21  *    .69  *
                                           *  -1.04  *  -1.65  *  -2.90  *
                                           ************************************
                                                     *    .84  *    .39  *  SIMULATE-3
                                                 13  *    .87  *    .40  *  SNA-CORE
                                                     *  -3.45  *  -2.50  *  PERCENT ERROR
                                                     ***********************
```

Figure 3-7

Catawba Unit 1 MOC-3
Assembly Power Distribution

	H	G	F	E	D	C	B	A
	.97	1.29	.98	1.28	.96	1.15	.90	.86
8	.96	1.30	1.00	1.29	.96	1.15	.91	.83
	1.04	-.77	-2.00	-.78	.00	.00	-1.10	3.61
		.98	1.11	.98	1.30	1.00	1.21	.76
9		.99	1.12	.99	1.30	1.01	1.21	.75
		-1.01	-.89	-1.01	.00	-.99	.00	1.33
			.96	1.31	1.00	1.19	1.07	.83
10			.95	1.31	1.00	1.20	1.07	.81
			1.05	.00	.00	-.83	.00	2.47
				1.00	1.30	.99	1.09	.51
11				1.01	1.31	.99	1.09	.50
				-.99	-.76	.00	.00	2.00
					.95	1.19	.69	
12					.94	1.19	.69	
					1.06	.00	.00	
						.86	.43	SIMULATE-3
13						.86	.42	SNA-CORE
						.00	2.38	PERCENT ERROR

Figure 3-8

Catawba Unit 1 EOC-3
Assembly Power Distribution

	H	G	F	E	D	C	B	A	
	.97	1.28	.98	1.28	.96	1.13	.90	.87	
8	.96	1.29	.99	1.29	.96	1.14	.91	.84	
	1.04	-.78	-1.01	-.78	.00	-.88	-1.10	3.57	
		.98	1.10	.98	1.29	.99	1.21	.77	
9		.99	1.11	.99	1.29	1.00	1.21	.75	
		-1.01	-.90	-1.01	.00	-1.00	.00	2.67	
		.96	1.30	.99	1.16	1.05	.84		
10		.95	1.30	1.00	1.17	1.05	.82		
		1.05	.00	-1.00	-.85	.00	2.44		
			1.00	1.29	.98	1.09	.53		
11			1.01	1.30	.99	1.08	.52		
			-.99	-.77	-1.01	.93	1.92		
				.95	1.18	.71			
12				.95	1.19	.71			
				.00	-.84	.00			
					.87	.46			SIMULATE-3
13					.88	.45			SNA-CORE
					-1.14	2.22			PERCENT ERROR

Figure 3-9

McGuire Unit 2 BOC-5
Assembly Power Distribution

```
        H          G          F          E          D          C          B          A

    ****************************************************************************************
    *   1.08  *   1.14  *   1.02  *   1.29  *    .99  *   1.04  *    .98  *    .66  *
 8  *   1.07  *   1.13  *   1.10  *   1.27  *    .99  *   1.05  *    .99  *    .65  *
    *    .93  *    .88  *  -7.27  *   1.57  *    .00  *   -.95  *  -1.01  *   1.54  *
    ****************************************************************************************
              *   1.05  *   1.32  *   1.10  *   1.29  *   1.02  *   1.16  *    .64  *
           9  *   1.03  *   1.31  *   1.09  *   1.27  *   1.01  *   1.16  *    .64  *
              *   1.94  *    .76  *    .92  *   1.57  *    .99  *    .00  *    .00  *
              **********************************************************************
                        *   1.05  *   1.31  *    .99  *   1.25  *   1.02  *    .75  *
                    10  *   1.04  *   1.30  *    .99  *   1.25  *   1.03  *    .75  *
                        *    .96  *    .77  *    .00  *    .00  *   -.97  *    .00  *
                        ****************************************************************
                                  *   1.02  *   1.23  *   1.15  *   1.08  *    .43  *
                              11  *   1.03  *   1.24  *   1.17  *   1.09  *    .43  *
                                  *   -.97  *   -.81  *  -1.71  *   -.92  *    .00  *
                                  ******************************************************
                                            *   1.16  *   1.17  *    .67  *
                                        12  *   1.17  *   1.17  *    .68  *
                                            *   -.85  *    .00  *  -1.47  *
                                            ******************************************
                                                      *    .79  *    .34  * SIMULATE-3
                                                  13  *    .80  *    .34  * SNA-CORE
                                                      *  -1.25  *    .00  * PERCENT ERROR
                                                      ******************************
```

Figure 3-10

McGuire Unit 2 MOC-5
Assembly Power Distribution

	H	G	F	E	D	C	B	A	
8	1.01	1.09	1.00	1.32	.99	1.03	.97	.68	
	1.02	1.10	1.02	1.33	1.00	1.05	.98	.67	
	-.98	-.91	-1.96	-.75	-1.00	-1.90	-1.02	1.49	
9		1.01	1.34	1.09	1.31	1.01	1.17	.66	
		1.01	1.34	1.09	1.31	1.00	1.16	.65	
		.00	.00	.00	.00	1.00	.86	1.54	
10			1.04	1.32	.98	1.27	1.01	.78	
			1.04	1.33	.97	1.26	1.01	.76	
			.00	-.75	1.03	.79	.00	2.63	
11				.99	1.16	1.10	1.09	.46	
				.99	1.17	1.12	1.09	.45	
				.00	-.85	-1.79	.00	2.22	
12				1.09	1.18	.69			
				1.10	1.18	.70			
				-.91	.00	-1.43			
13					.81	.38	SIMULATE-3		
					.81	.37	SNA-CORE		
					.00	2.70	PERCENT ERROR		

Figure 3-11

McGuire Unit 2 EOC-5
Assembly Power Distribution

	H	G	F	E	D	C	B	A	
8	1.01	1.06	.98	1.28	.98	1.03	.97	.72	SIMULATE-3
	.99	1.07	.99	1.28	.99	1.05	.99	.72	SNA-CORE
	2.02	-.93	-1.01	.00	-1.01	-1.90	-2.02	.00	PERCENT ERROR
9		.99	1.29	1.06	1.28	1.00	1.16	.69	
		.98	1.30	1.05	1.29	1.00	1.15	.69	
		1.02	-.77	.95	-.78	.00	.87	.00	
10			1.01	1.29	.97	1.25	1.01	.81	
			1.01	1.31	.97	1.25	1.01	.80	
			.00	-1.53	.00	.00	.00	1.25	
11				.98	1.14	1.09	1.10	.50	
				.98	1.15	1.10	1.10	.49	
				.00	-.87	-.91	.00	2.04	
12					1.09	1.21	.74		
					1.08	1.20	.74		
					.93	.83	.00		
13						.86	.44		SIMULATE-3
						.86	.43		SNA-CORE
						.00	2.33		PERCENT ERROR

Figure 3-12

Oconee Unit 2 BOC-9
Assembly Power Distribution

	8	9	10	11	12	13	14	15
H	1.02	1.24	1.04	1.00	1.19	1.18	1.00	.51
	1.01	1.20	1.04	1.00	1.18	1.19	1.00	.48
	.99	3.33	.00	.00	.85	-.84	.00	6.25
K		1.11	1.26	1.08	1.27	1.20	1.13	.47
		1.11	1.23	1.06	1.28	1.23	1.12	.49
		.00	2.44	1.89	-.78	-2.44	.89	-4.08
L			1.27	1.29	1.00	1.25	.92	.34
			1.24	1.33	1.03	1.26	.92	.35
			2.42	-3.01	-2.91	-.79	.00	-2.86
M				1.13	1.29	1.14	.66	
				1.13	1.27	1.14	.69	
				.00	1.57	.00	-4.35	
N					1.20	1.05	.43	
					1.18	1.03	.44	
					1.69	1.94	-2.27	
O						.54	SIMULATE-3	
						.54	OAC	
						.00	PERCENT ERROR	

Figure 3-13

Oconee Unit 2 MOC-9
Assembly Power Distribution

Each cell lists three values: SIMULATE-3 (top), OAC (middle), PERCENT ERROR (bottom).

	8	9	10	11	12	13	14	15
H	1.09 / 1.11 / -1.80	1.37 / 1.35 / 1.48	1.08 / 1.10 / -1.82	1.00 / 1.01 / -.99	1.13 / 1.13 / .00	1.09 / 1.08 / .93	.93 / .90 / 3.33	.50 / .48 / 4.17
K		1.15 / 1.21 / -4.96	1.36 / 1.35 / .74	1.08 / 1.09 / -.92	1.30 / 1.29 / .78	1.12 / 1.12 / .00	1.07 / 1.05 / 1.90	.47 / .47 / .00
L			1.28 / 1.27 / .79	1.37 / 1.41 / -2.84	1.01 / 1.05 / -3.81	1.25 / 1.24 / .81	.89 / .89 / .00	.35 / .35 / .00
M				1.14 / 1.15 / -.87	1.33 / 1.30 / 2.31	1.09 / 1.10 / -.91	.66 / .66 / .00	
N					1.14 / 1.13 / .88	1.02 / 1.04 / -1.92	.44 / .43 / 2.33	
O						.54 / .52 / 3.85		

Legend:
.54 * SIMULATE-3
.52 * OAC
3.85 * PERCENT ERROR

Figure 3-14

Oconee Unit 2 EOC-9
Assembly Power Distribution

	8	9	10	11	12	13	14	15

```
****************************************************************************
*    1.02  *   1.28  *   1.03  *    .99  *   1.10  *   1.08  *    .96  *    .57  *
H  *    1.02  *   1.24  *   1.04  *   1.00  *   1.12  *   1.08  *    .93  *    .55  *
*     .00  *   3.23  *   -.96  *  -1.00  *  -1.79  *    .00  *   3.23  *   3.64  *
****************************************************************************
        *   1.08  *   1.28  *   1.05  *   1.28  *   1.11  *   1.09  *    .53  *
    K   *   1.11  *   1.26  *   1.06  *   1.28  *   1.11  *   1.08  *    .54  *
        *  -2.70  *   1.59  *   -.94  *    .00  *    .00  *    .93  *  -1.85  *
        ****************************************************************************
                *   1.19  *   1.32  *   1.06  *   1.25  *    .93  *    .41  *
            L   *   1.18  *   1.37  *   1.10  *   1.26  *    .93  *    .40  *
                *    .85  *  -3.65  *  -3.64  *   -.79  *    .00  *   2.50  *
                ****************************************************************************
                        *   1.11  *   1.31  *   1.09  *    .72  *
                    M   *   1.11  *   1.30  *   1.11  *    .71  *
                        *    .00  *    .77  *  -1.80  *   1.41  *
                        ****************************************************
                                *   1.11  *   1.03  *    .49  *
                            N   *   1.11  *   1.04  *    .47  *
                                *    .00  *   -.96  *   4.26  *
                                ********************************
                                        *    .58  * SIMULATE-3
                                    O   *    .55  * OAC
                                        *   5.45  * PERCENT ERROR
                                        ************
```

4.0 POWER DISTRIBUTION UNCERTAINTY

4.1 Qualitative Comparisons

4.1(a) General Description

Representative power distribution comparisons between measured and calculated values were previously shown in Figures 3-6 through 3-14. These and other comparisons throughout the cycles analyzed demonstrate no significant trends with assembly burnup, enrichment, or burnable poison loading. Since SIMULATE-3P does not require normalization or other empirical adjustments to achieve these comparisons a high degree of confidence may be placed in its predictive capability. The EPRI-NODE-P model to be used as a supplement to SIMULATE-3P is similar to that described in Reference 1. However, since it will use data from CASMO-3 and be normalized to SIMULATE-3P instead of PDQ, new uncertainties shall be derived for its application to development of cycle-specific operating limits.

4.1(b) SIMULATE-3P versus PDQ

Because typical power reactors, such as those operated by Duke Power Company, do not measure actual pin power distributions, the accuracy of pin power predictions must be otherwise inferred. Since the measured data used to determine the radial-local reliability factor is taken from small, low power critical experiments the applicability of the model to large power reactors is demonstrated by comparison to other calculational methods.

4-1

Supplemental to the SIMULATE-3P benchmark executions, a CASMO-3 based PDQ model for Catawba Unit 1 Cycles 1 and 2 was generated. By using the same cross section generator, differences in the two models are reduced to the calculational techniques involved. While PDQ is not presented as an exact solution, comparisons demonstrate the pin power reconstruction technique in SIMULATE-3P match a fine mesh, two-group, diffusion theory solution of a large power reactor problem. A severe test of calculational techniques is involved by modeling large intra-assembly burnup gradients. The Cycle 2 cases demonstrate the ability to accurately model fuel shuffled from core peripheral locations (where large gradients are present) to interior locations. Comparisons of assembly radial and peak pin powers are shown in Figures 4-1 through 4-12. The comparison of these PDQ and SIMULATE-3P pin power calculations show no significant or systematic differences. This supports the applicability of the reconstruction technique to large power reactors.

4.2 Quantitative Comparisons

In order to apply the SIMULATE-3P methodology to reload design, appropriate uncertainty factors are required for use in generating operating limits. When EPRI-NODE-P is used in conjunction with SIMULATE-3P to generate operating limits all peaking information except local pin peaking data is calculated with EPRI-NODE-P. Therefore, only the uncertainty for radial-local factors is based on SIMULATE-3P and all other uncertainties are based on EPRI-NODE-P. When SIMULATE-3P is used as a stand-alone model to generate operating limits, all uncertainties are based on SIMULATE-3P.

4.2(a) Radial-Local Power Distribution Uncertainty

Extensive benchmarking of the pin power prediction capability of
SIMULATE-3P was provided by Yankee Atomic Electric Company in
Section 5 of Reference 5. Comparisons were made to measured
critical experiments, colorset transport theory calculations, and
quarter core PDQ calculations. Section 6 of Reference 5 concludes
that SIMULATE-3P calculates the reference transport theory pinwise
powers within 1%. Duke Power previously demonstrated the ability of
CASMO-2 to calculate measured critical pin powers within 1.7%
(Reference 1). In order to maintain conservatism and be consistant
with previously approved values (e.g. Reference 3), an uncertainty
of 2% will be utilized.

4.2(b) Assembly Power Distribution Uncertainties

Assembly uncertainties are based upon comparisons of the appropriate
reactor model to measured data. The derivation of the statistical
model used to determine the Observed Nuclear Reliability Factors
(ONRF) is given in Supplement 2, Section 5.1 of Reference 2. The
ONRF is defined by the relationship:

$$ONRF = \frac{\overline{M} - \overline{D} + K * S(D)}{\overline{M}} \qquad (4-1)$$

where:

$$\overline{D} = \overline{C} - \overline{M} = (\sum_{i=1}^{n} D_i) \div n \qquad (4\text{-}2)$$

$$\overline{D}_i = \overline{C}_i - \overline{M}_i \text{ the } i\text{th difference, } 1 \leq i \leq N \qquad (4\text{-}3)$$

C_i is the ith calculated value (radial or peak)

M_i is the ith measured value (radial or peak)

$$\overline{C} = (\sum_{i=1}^{n} c_i) \div n \qquad (4\text{-}4)$$

$$\overline{M} = (\sum_{i=1}^{n} M_i) \div n \qquad (4\text{-}5)$$

K is the one-sided 95/95 tolerance factor

S(D) is the standard deviation of the differences

Using the engineering judgement that only peaking factors greater than the core average are of concern, only pairs of C and M where both are ≥ 1.0 are evaluated. To ensure that the statistical treatment of ONRFs derived in Reference 2 is applicable, the normality D' test from ANSI N15.15-1974 (Reference 7) is applied to each set of data. Separate sets of ONRFs are generated for SIMULATE -3P and EPRI-NODE-P.

Application of these two models to generation of operating limits requires ONRFs for assembly radial, axial, and total peaking factors. The radial and total comparisons are based on power distributions normalized to the core average while axial distributions are normalized to the assembly average. Since McGuire and Catawba are both Westinghouse 193-assembly plants their data is combined to provide 45 reactor statepoints covering 4 cycles of operation. The Oconee data consists of 59 reactor statepoints over 4 cycles of operation.

The results of the normality tests include the calculated value of D' and the critical values between which it must fall for the data to be deemed normal. With a level of significance of 5% these critical values were determined from the 2.5% and 97.5% values in Table 5 of Reference 7. The normality test results for radial, axial, and total peaking factors are given in Tables 4-1 through 4-4. The data used in the calculation of the ONRF as well as the ONRF itself for radial, axial, and total peaking are shown in Table 4.5. The application of these ONRFs to generating operating limits is discussed in Sections 5.1, 5.2, and 5.3.

Table 4-1

SIMULATE-3P
MCGUIRE/CATAWBA NORMALITY TEST RESULTS

Radial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
McGuire 2	4	384	2097	2028.5	2142	Nearly Normal
McGuire 2	5	447	2636	2630.8	2688	Nearly Normal
Catawba 1	3	236	1006	1018.8	1034	Normal
Catawba 2	2	388	2130	2056.1	2173	Nearly Normal
All	All	1455	15561	15180	15732	Nearly Normal.

Axial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
McGuire 2	4	616	4272	4341.3	4345	Normal
McGuire 2	5	728	5494	5616.4	5579	Nearly Normal
Catawba 1	3	504	3158	3231.1	3217	Nearly Normal
Catawba 2	2	672	4870	4910.4	4949	Normal
All	All	2520	35500	36048	35800	Nearly Normal

Total

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
McGuire 2	4	471	2852	2801.8	2904	Nearly Normal
McGuire 2	5	559	3691	3782.4	3757	Nearly Normal
Catawba 1	3	379	2056	2092.0	2100	Normal
Catawba 2	2	589	3994	4024.7	4063	Normal
All	All	1998	25072	25260	25302	Normal

Table 4-2

SIMULATE-3P
OCONEE NORMALITY TEST RESULTS

Radial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
Oconee 2	8	241	1039	1068.9	1067	Nearly Normal
Oconee 2	9	263	1185	1185.1	1216	Normal
Oconee 3	9	191	731.7	738.94	753.9	Normal
Oconee 3	10	277	1281	1288.1	1314	Normal
All	All	972	8487	8499.7	8602	Normal

Axial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
Oconee 2	8	390	2147	2126.4	2192	Nearly Normal
Oconee 2	9	400	2230	2177.7	2276	Nearly Normal
Oconee 3	9	335	1707	1738.3	1746	Normal
Oconee 3	10	405	2273	2257.1	2319	Nearly Normal
All	All	1530	16788	16459	16968	Nearly Normal

Total

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
Oconee 2	8	302	1460	1468.6	1495	Normal
Oconee 2	9	280	1302	1314.5	1335	Normal
Oconee 3	9	238	1019	1035.9	1047	Normal
Oconee 3	10	302	1460	1451.3	1495	Nearly Normal
All	All	1122	10532	10555	10662	Normal

Table 4-3

EPRI-NODE-P
MCGUIRE/CATAWBA NORMALITY TEST RESULTS

Radial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
McGuire 2	4	376	2032	1977.4	2075	Nearly Normal
McGuire 2	5	426	2452	2500.7	2502	Normal
Catawba 1	3	255	1131	1159.9	1161	Normal
Catawba 2	2	403	2256	2274.6	2302	Normal
All	All	1460	15642	15656	15814	Normal

Axial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
McGuire 2	4	616	4272	4374.3	4345	Nearly Normal
McGuire 2	5	728	5494	5622.0	5579	Nearly Normal
Catawba 1	3	504	3158	3252.4	3217	Nearly Normal
Catawba 2	2	672	4870	4882.6	4949	Normal
All	All	2520	35500	35539	35800	Normal

Total

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
McGuire 2	4	475	2889	2907.3	2945	Normal
McGuire 2	5	559	3691	3741.7	3757	Normal
Catawba 1	3	392	2164	2154.1	2209	Nearly Normal
Catawba 2	2	589	3994	3961.1	4063	Nearly Normal
All	All	2015	25390	25428	25622	Normal

Table 4-4

EPRI-NODE-P
OCONEE NORMALITY TEST RESULTS

Radial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
Oconee 2	8	241	1039	1053.0	1067	Normal
Oconee 2	9	255	1131	1123.6	1161	Nearly Normal
Oconee 3	9	191	731.7	723.16	753.9	Nearly Normal
Oconee 3	10	267	1212	1241.5	1244	Normal
All	All	954	8251	8249.9	8363	Nearly Normal

Axial

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
Oconee 2	8	390	2147	2137.2	2192	Nearly Normal
Oconee 2	9	400	2230	2251.9	2276	Normal
Oconee 3	9	335	1707	1748.4	1746	Nearly Normal
Oconee 3	10	405	2273	2318.2	2319	Normal
All	All	1530	16788	16857	16968	Normal

Total

Unit	Cycle	N	D'(P=.025)	D'	D'(P=.975)	Remarks
Oconee 2	8	303	1467	1502.7	1502	Nearly Normal
Oconee 2	9	279	1295	1305.0	1328	Normal
Oconee 3	9	241	1039	1053.1	1067	Normal
Oconee 3	10	305	1482	1488.1	1517	Normal
All	All	1128	10618	10660	10748	Normal

Table 4-5

OBSERVED NUCLEAR RELIABILITY FACTORS

SIMULATE-3P FACTORS
FOR B&W PLANTS

Type	N	$\overline{M}$	$\overline{D}$	K	S(D)	ONRF
Radial	972	1.166	0.000	1.728	0.018	1.027
Axial	1530	1.108	-0.011	1.711	0.022	1.044
Total	1122	1.265	-0.017	1.723	0.028	1.052

SIMULATE-3P FACTORS
FOR WESTINGHOUSE PLANTS

Type	N	$\overline{M}$	$\overline{D}$	K	S(D)	ONRF
Radial	1455	1.145	0.000	1.713	0.011	1.017
Axial	2520	1.148	-0.027	1.697	0.020	1.053
Total	1998	1.257	-0.027	1.703	0.026	1.057

EPRI-NODE-P FACTORS
FOR B&W PLANTS

Type	N	$\overline{M}$	$\overline{D}$	K	S(D)	ONRF
Radial	954	1.169	-0.005	1.729	0.027	1.044
Axial	1530	1.108	0.015	1.711	0.028	1.030
Total	1128	1.264	0.012	1.723	0.046	1.053

EPRI-NODE-P FACTORS
FOR WESTINGHOUSE PLANTS

Type	N	$\overline{M}$	$\overline{D}$	K	S(D)	ONRF
Radial	1460	1.144	-0.005	1.713	0.019	1.033
Axial	2520	1.148	-0.005	1.697	0.026	1.043
Total	2015	1.255	-0.007	1.703	0.035	1.053

Figure 4-1

Catawba Unit 1 BOC-1
PDQ vs. SIMULATE-3P Assembly Powers

	H	G	F	E	D	C	B	A
8	.8480	.9480	.9380	1.0960	.9500	.9960	.9200	.9310
	.8650	.9826	.9509	1.1217	.9586	1.0199	.9120	.9132
	-1.9653	-3.5213	-1.3566	-2.2912	-.8971	-2.3434	.8772	1.9492
9	.9480	.9060	1.0810	.9760	1.0850	.9260	1.0860	.9740
	.9826	.9203	1.1092	.9853	1.1067	.9297	1.1083	.9532
	-3.5213	-1.5538	-2.5424	-.9439	-1.9608	-.3980	-2.0121	2.1821
10	.9380	1.0810	.9820	1.1210	.9670	1.0150	.9040	.8590
	.9511	1.1093	.9909	1.1416	.9686	1.0305	.8902	.8388
	-1.3774	-2.5512	-.8982	-1.8045	-.1652	-1.5041	1.5502	2.4082
11	1.0960	.9760	1.1210	1.0110	1.1090	.9980	1.0920	.6970
	1.1220	.9855	1.1418	1.0105	1.1207	.9837	1.0897	.6747
	-2.3173	-.9640	-1.8217	.0495	-1.0440	1.4537	.2111	3.3052
12	.9500	1.0850	.9670	1.1090	1.3870	1.1150	1.1180	
	.9589	1.1071	.9688	1.1208	1.3690	1.1004	1.0893	
	-.9281	-1.9962	-.1858	-1.0528	1.3148	1.3268	2.6347	
13	.9960	.9260	1.0150	.9980	1.1150	1.1240	.8000	
	1.0204	.9301	1.0309	.9839	1.1005	1.0899	.7735	
	-2.3912	-.4408	-1.5423	1.4331	1.3176	3.1287	3.4260	
14	.9200	1.0860	.9040	1.0920	1.1180	.8000		
	.9125	1.1089	.8906	1.0900	1.0894	.7735		
	.8219	-2.0651	1.5046	.1835	2.6253	3.4260		
15	.9310	.9740	.8590	.6970	SIMULATE-3			
	.9137	.9537	.8392	.6750	PDQ			
	1.8934	2.1286	2.3594	3.2593	PERCENT ERROR			

Figure 4-2

Catawba Unit 1 MOC-1
PDQ vs. SIMULATE-3P Assembly Powers

	H	G	F	E	D	C	B	A
8	1.0380	1.1540	1.0590	1.1980	1.0380	1.0960	.9240	.8270
	1.0252	1.1477	1.0476	1.1914	1.0345	1.1054	.9228	.8189
	1.2485	.5489	1.0882	.5540	.3383	-.8504	.1300	.9891
9	1.1540	1.0520	1.2020	1.0600	1.1710	.9930	1.0840	.8460
	1.1477	1.0403	1.1926	1.0529	1.1728	.9952	1.0992	.8371
	.5489	1.1247	.7882	.6743	-.1535	-.2211	-1.3828	1.0632
10	1.0590	1.2020	1.0660	1.1970	1.0320	1.0790	.8850	.7600
	1.0476	1.1926	1.0572	1.1951	1.0325	1.0921	.8870	.7548
	1.0882	.7882	.8324	.1590	-.0484	-1.1995	-.2255	.6889
11	1.1980	1.0600	1.1970	1.0580	1.1510	.9800	.9980	.6090
	1.1914	1.0528	1.1950	1.0574	1.1616	.9848	1.0088	.6014
	.5540	.6839	.1674	.0567	-.9125	-.4874	-1.0706	1.2637
12	1.0380	1.1710	1.0320	1.1510	1.2360	1.0450	.9150	
	1.0345	1.1728	1.0324	1.1616	1.2376	1.0530	.9143	
	.3383	-.1535	-.0387	-.9125	-.1293	-.7597	.0766	
13	1.0960	.9930	1.0790	.9800	1.0450	.9780	.6450	
	1.1053	.9951	1.0920	.9848	1.0530	.9795	.6420	
	-.8414	-.2110	-1.1905	-.4874	-.7597	-.1531	.4673	
14	.9240	1.0840	.8850	.9980	.9150	.6450		
	.9227	1.0991	.8869	1.0088	.9143	.6420		
	.1409	-1.3739	-.2142	-1.0706	.0766	.4673		
15	.8270	.8460	.7600	.6090	SIMULATE-3			
	.8189	.8370	.7547	.6014	PDQ			
	.9891	1.0753	.7023	1.2637	PERCENT ERROR			

Figure 4-3

Catawba Unit 1 EOC-1
PDQ vs. SIMULATE-3P Assembly Powers

```
        H           G           F           E           D           C           B      ↲     A

****************************************************************************************************
   *    .9210  *  1.0350  *    .9220  *  1.0620 ⚹    .9760  *  1.1410  *    .9910  *    .9000 ⚹
 8 *    .9158  *  1.0317  ⚹    .9170  ⚹  1.0567  ⚹    .9702  *  1.1397  ⚹    .9904  *    .8955  ⚹
   *    .5678  *    .3199  ⚹    .5453  ⚹    .5016  ⚹    .5978  *    .1141  *    .0606  *    .5025  ⚹
****************************************************************************************************
   *  1.0350  *    .9190  *  1.0440  *    .9430  ⚹  1.1060  *  1.0130  *  1.2030  *    .9070  *
 9 *  1.0317  *    .9140  ⚹  1.0387  ⚹    .9382  *  1.1014  *  1.0108  *  1.2132  *    .9021  *
   *    .3199  *    .5470  ⚹    .5103  *    .5116  ⚹    .4177  *    .2176  ⚹  -.8408  *    .5432  ⚹
****************************************************************************************************
   *    .9220  ⚹  1.0440  *    .9330  *  1.0800  *    .9890  *  1.1360  *    .9590  *    .8430  ⚹
10 ⚹    .9170  ⚹  1.0387  ⚹    .9284  ⚹  1.0765  *    .9857  *  1.1392  ⚹    .9613  *    .8414  ⚹
   *    .5453  *    .5103  *    .4955  *    .3251  *    .3348  ⚹  -.2809  ⚹  -.2393  *    .1902  ⚹
****************************************************************************************************
   *  1.0620  *    .9430  *  1.0800  *    .9850  *  1.1430  *  1.0110  *  1.0950  *    .6830  ⚹
11 *  1.0567  *    .9382  *  1.0765  *    .9824  *  1.1461  *  1.0146  *  1.1061  *    .6760  ⚹
   *    .5016  *    .5116  *    .3251  ⚹    .2647  ⚹  -.2705  *  -.3548  * -1.0035  *  1.0355  ⚹
****************************************************************************************************
   *    .9760  *  1.1060  *    .9890  *  1.1430  *  1.1890  *  1.1240  *    .9610  ⚹
12 *    .9701  *  1.1014  *    .9857  *  1.1461  *  1.1902  *  1.1368  *    .9626  ⚹
   ⚹    .6082  *    .4177  *    .3348  *  -.2705  *  -.1008  ⚹ -1.1260  *  -.1662  *
****************************************************************************************************
   *  1.1410  *  1.0130  *  1.1360  *  1.0110  ⚹  1.1240  ⚹  1.0850  *    .7010  *
13 ⚹  1.1397  *  1.0108  *  1.1392  *  1.0146  ⚹  1.1368  *  1.1013  ⚹    .7010  *
   ⚹    .1141  *    .2176  ⚹  -.2809  *  -.3548  * -1.1260  * -1.4801  *    .0000  *
****************************************************************************************************
   ⚹    .9910  *  1.2030  *    .9590  ⚹  1.0950  *    .9610  *    .7010  ⚹
14 ⚹    .9904  ⚹  1.2132  *    .9613  *  1.1061  *    .9626  ⚹    .7010  *
   ⚹    .0606  ⚹  -.8408  ⚹  -.2393  * -1.0035  ⚹  -.1662  *    .0000  ⚹
****************************************************************************************************
   ⚹    .9000  ⚹    .9070  *    .8430  *    .6830  * SIMULATE-3
15 ⚹    .8955  ⚹    .9021  *    .8414  ⚹    .6760  * PDQ
   *    .5025  ⚹    .5432  ⚹    .1902  ⚹  1.0355  * PERCENT ERROR
****************************************
```

Figure 4-4

Catawba Unit 1 BOC-2
PDQ vs. SIMULATE-3P Assembly Powers

	H	G	F	E	D	C	B	A
	***							
	.9400	1.3100	1.2300	1.2180	.9600	1.1320	1.1140	1.0320
8	.9269	1.3247	1.2455	1.2354	.9624	1.1331	1.1085	1.0036
	1.4133	-1.1097	-1.2445	-1.4085	-.2494	-.0971	.4962	2.8298
	***							
	1.3100	1.2630	1.2590	1.0390	1.1920	.9340	1.0970	.9640
9	1.3297	1.2797	1.2735	1.0453	1.2043	.9314	1.0870	.9373
	-1.4815	-1.3050	-1.1386	-.6027	-1.0213	.2791	.9200	2.8486
	***							
	1.2300	1.2560	1.0450	1.2700	.9360	1.2100	1.0530	.8460
10	1.2512	1.2767	1.0512	1.2885	.9359	1.2156	1.0460	.8266
	-1.6944	-1.6214	-.5898	-1.4358	.0107	-.4607	.6692	2.3470
	***							
	1.2180	1.0300	1.2670	.9040	1.0910	.9050	1.0510	.3680
11	1.2382	1.0468	1.2888	.9044	1.1079	.9000	1.0479	.3571
	-1.6314	-1.6049	-1.6915	-.0442	-1.5254	.5556	.2958	3.0524
	***							
	.9600	1.1930	.9370	1.0920	.8450	1.0510	.8390	
12	.9637	1.2057	.9365	1.1082	.8446	1.0551	.8354	
	-.3839	-1.0533	.0534	-1.4618	.0474	-.3886	.4309	
	***							
	1.1320	.9430	1.2150	.9070	1.0520	1.0270	.3450	
13	1.1344	.9324	1.2165	.9004	1.0553	1.0333	.3390	
	-.2116	1.1369	-.1233	.7330	-.3127	-.6097	1.7699	
	***							
	1.1140	1.1050	1.0590	1.0550	.8400	.3450		
14	1.1097	1.0881	1.0469	1.0485	.8357	.3391		
	.3875	1.5532	1.1558	.6199	.5145	1.7399		
	**							
	1.0320	.9690	.8500	.3690	SIMULATE-3			
15	1.0047	.9383	.8274	.3574	PDQ			
	2.7172	3.2719	2.7314	3.2457	PERCENT ERROR			
	***							

Figure 4-5

Catawba Unit 1 MOC-2
PDQ vs. SIMULATE-3P Assembly Powers

	H	G	F	E	D	C	B	A
8	.8140	1.0750	1.0480	1.1010	.9250	1.0850	1.0400	.9480
	.8090	1.0809	1.0602	1.1132	.9380	1.0892	1.0381	.9240
	.6180	-.5458	-1.1507	-1.0959	-1.3859	-.3856	.1830	2.5974
9	1.0750	1.0490	1.1030	.9830	1.1560	.9380	1.0510	.9110
	1.0829	1.0599	1.1106	.9934	1.1640	.9461	1.0450	.8870
	-.7295	-1.0284	-.6843	-1.0469	-.6873	-.8561	.5742	2.7057
10	1.0480	1.1010	.9930	1.3090	.9970	1.2840	1.0510	.8490
	1.0621	1.1115	1.0060	1.3227	1.0061	1.2896	1.0472	.8293
	-1.3276	-.9447	-1.2922	-1.0358	-.9045	-.4342	.3629	2.3755
11	1.1010	.9770	1.3080	1.0050	1.2700	.9900	1.1140	.4160
	1.1135	.9934	1.3224	1.0106	1.2835	.9935	1.1037	.4070
	-1.1226	-1.6509	-1.0889	-.5541	-1.0518	-.3523	.9332	2.2113
12	.9250	1.1560	.9970	1.2700	.9640	1.1230	.9090	
	.9379	1.1639	1.0060	1.2834	.9698	1.1178	.8944	
	-1.3754	-.6788	-.8946	-1.0441	-.5981	.4652	1.6324	
13	1.0850	.9430	1.2860	.9910	1.1230	1.0890	.4040	
	1.0890	.9460	1.2895	.9934	1.1177	1.0815	.3967	
	-.3673	-.3171	-.2714	-.2416	.4742	.6935	1.8402	
14	1.0400	1.0560	1.0530	1.1150	.9090	.4040		
	1.0380	1.0449	1.0471	1.1036	.8943	.3966		
	.1927	1.0623	.5635	1.0330	1.6437	1.8659		
15	.9480	.9130	.8500	.4170	SIMULATE-3			
	.9239	.8869	.8292	.4070	PDQ			
	2.6085	2.9428	2.5084	2.4570	PERCENT ERROR			

Figure 4-6

Catawba Unit 1 EOC-2
PDQ vs. SIMULATE-3P Assembly Powers

```
            H          G          F          E          D          C          B     🖝    A

    ***************************************************************************************
    *   .8290 *  1.0520 *  1.0240 *  1.0810 *   .9250 *  1.0660 *  1.0100 *   .9190 *
  8 *   .8079 *  1.0359 *  1.0191 *  1.0765 *   .9303 *  1.0675 *  1.0144 *   .9105 *
    *  2.6117 *  1.5542 *   .4808 *   .4180 *  -.5697 *  -.1405 *  -.4338 *   .9336 *
    ***************************************************************************************
    *  1.0520 *  1.0250 *  1.0810 *   .9810 *  1.1400 *   .9390 *  1.0260 *   .8900 *
  9 *  1.0373 *  1.0163 *  1.0706 *   .9791 *  1.1385 *   .9479 *  1.0278 *   .8816 *
    *  1.4171 *   .8560 *   .9714 *   .1941 *   .1318 *  -.9389 *  -.1751 *   .9528 *
    ***************************************************************************************
    *  1.0240 *  1.0800 *   .9960 *  1.3260 *  1.0170 *  1.2930 *  1.0340 *   .8490 *
 10 *  1.0202 *  1.0711 *   .9964 *  1.3261 *  1.0233 *  1.3017 *  1.0401 *   .8432 *
    *   .3725 *   .8309 *  -.0401 *  -.0075 *  -.6157 *  -.6684 *  -.5865 *   .6879 *
    ***************************************************************************************
    *  1.0810 *   .9770 *  1.3250 *  1.0420 *  1.3310 *  1.0040 *  1.1120 *   .4440 *
 11 *  1.0766 *   .9790 *  1.3258 *  1.0422 *  1.3415 *  1.0129 *  1.1132 *   .4397 *
    *   .4087 *  -.2043 *  -.0603 *  -.0192 *  -.7827 *  -.8787 *  -.1078 *   .9779 *
    ***************************************************************************************
    *   .9250 *  1.1400 *  1.0170 *  1.3310 *   .9920 *  1.1090 *   .9130 *
 12 *   .9303 *  1.1384 *  1.0232 *  1.3414 *  1.0016 *  1.1114 *   .9070 *
    *  -.5697 *   .1405 *  -.6059 *  -.7753 *  -.9585 *  -.2159 *   .6615 *
    ***************************************************************************************
    *  1.0660 *   .9430 *  1.2940 *  1.0040 *  1.1090 *  1.0700 *   .4290 *
 13 *  1.0675 *   .9479 *  1.3017 *  1.0129 *  1.1114 *  1.0741 *   .4256 *
    *  -.1405 *  -.5169 *  -.5915 *  -.8787 *  -.2159 *  -.3817 *   .7989 *
    ***************************************************************************************
    *  1.0100 *  1.0290 *  1.0340 *  1.1120 *   .9130 *   .4290 *
 14 *  1.0144 *  1.0278 *  1.0401 *  1.1132 *   .9070 *   .4256 *
    *  -.4338 *   .1168 *  -.5865 *  -.1078 *   .6615 *   .7989 *
    ***************************************************************************************
    *   .9190 *   .8910 *   .8500 *   .4440 * SIMULATE-3
 15 *   .9105 *   .8816 *   .8432 *   .4397 * PDQ
    *   .9336 *  1.0662 *   .8065 *   .9779 * PERCENT ERROR
    ***********************************************
```

Figure 4-7

Catawba Unit 1 BOC-1
PDQ vs. SIMULATE-3P Peak Pin Powers

```
            H          G          F          E          D          C          B          A

    ****************************************************************************************
    *    .9010 *  1.1640 *  1.0090 *  1.2360 *  1.0170 *  1.1920 *    .9940 *  1.3040 *
 8  *    .9186 *  1.1595 *  1.0146 *  1.2288 *  1.0212 *  1.1783 *    .9867 *  1.2793 *
    *  -1.9160 *    .3881 *  -.5519 *    .5859 *  -.4113 *  1.1627 *    .7398 *  1.9307 *
    ****************************************************************************************
    *   1.1640 *    .9880 *  1.2500 *  1.0370 *  1.2470 *    .9900 *  1.3940 *  1.2610 *
 9  *   1.1596 *    .9912 *  1.2436 *  1.0430 *  1.2362 *    .9916 *  1.3816 *  1.2207 *
    *    .3794 *  -.3228 *    .5146 *  -.5753 *    .8736 *  -.1614 *    .8975 *  3.3014 *
    ****************************************************************************************
    *   1.0090 *  1.2500 *  1.0480 *  1.2730 *  1.0360 *  1.2350 *    .9840 *  1.2590 *
10  *   1.0148 *  1.2438 *  1.0494 *  1.2611 *  1.0303 *  1.2061 *    .9657 *  1.2085 *
    *   -.5715 *    .4985 *  -.1334 *    .9436 *    .5532 *  2.3962 *  1.8950 *  4.1787 *
    ****************************************************************************************
    *   1.2360 *  1.0370 *  1.2730 *  1.0890 *  1.3400 *  1.0960 *  1.4410 *  1.1250 *
11  *   1.2292 *  1.0432 *  1.2611 *  1.0751 *  1.2984 *  1.0677 *  1.4031 *  1.0921 *
    *    .5532 *  -.5943 *    .9436 *  1.2929 *  3.2039 *  2.6506 *  2.7012 *  3.0125 *
    ****************************************************************************************
    *   1.0170 *  1.2470 *  1.0360 *  1.3400 *  1.4950 *  1.3450 *  1.4210 *
12  *   1.0215 *  1.2365 *  1.0305 *  1.2984 *  1.4643 *  1.2989 *  1.3823 *
    *   -.4405 *    .8492 *    .5337 *  3.2039 *  2.0966 *  3.5492 *  2.7997 *
    ****************************************************************************************
    *   1.1920 *    .9900 *  1.2350 *  1.0960 *  1.3450 *  1.5000 *  1.2930 *
13  *   1.1788 *    .9920 *  1.2063 *  1.0679 *  1.2990 *  1.4253 *  1.2544 *
    *   1.1198 *  -.2016 *  2.3792 *  2.6313 *  3.5412 *  5.2410 *  3.0772 *
    ****************************************************************************************
    *    .9940 *  1.3940 *    .9840 *  1.4410 *  1.4210 *  1.2930 *
14  *    .9873 *  1.3823 *    .9661 *  1.4036 *  1.3826 *  1.2545 *
    *    .6786 *    .8464 *  1.8528 *  2.6646 *  2.7774 *  3.0690 *
    ****************************************************************************************
    *   1.3040 *  1.2610 *  1.2620 *  1.1250 * SIMULATE-3
15  *   1.2800 *  1.2213 *  1.2091 *  1.0925 * PDQ
    *   1.8750 *  3.2506 *  4.3752 *  2.9748 * PERCENT ERROR
    ***************************************************
```

4-17

Figure 4-8

Catawba Unit 1 MOC-1
PDQ vs. SIMULATE-3P Peak Pin Powers

	H	G	F	E	D	C	B	A
8	1.0880	1.2510	1.1130	1.2630	1.0990	1.2130	1.0070	1.1450
	1.0619	1.2139	1.0846	1.2377	1.0760	1.1896	.9928	1.1216
	2.4579	3.0563	2.6185	2.0441	2.1375	1.9670	1.4303	2.0863
9	1.2510	1.1100	1.2640	1.1130	1.2480	1.0600	1.3300	1.1070
	1.2138	1.0814	1.2348	1.0885	1.2230	1.0455	1.2994	1.0800
	3.0648	2.6447	2.3648	2.2508	2.0442	1.3869	2.3549	2.5000
10	1.1130	1.2640	1.1150	1.2610	1.0960	1.2070	.9820	1.0930
	1.0846	1.2348	1.0916	1.2383	1.0765	1.1862	.9718	1.0647
	2.6185	2.3648	2.1436	1.8332	1.8114	1.7535	1.0496	2.6580
11	1.2630	1.1130	1.2610	1.1110	1.2480	1.0600	1.2870	.9850
	1.2377	1.0884	1.2383	1.0963	1.2318	1.0496	1.2625	.9624
	2.0441	2.2602	1.8332	1.3409	1.3151	.9909	1.9406	2.3483
12	1.0990	1.2480	1.0960	1.2480	1.3130	1.2040	1.2030	
	1.0759	1.2230	1.0765	1.2318	1.2973	1.1935	1.1869	
	2.1470	2.0442	1.8114	1.3151	1.2102	.8798	1.3565	
13	1.2130	1.0600	1.2070	1.0600	1.2040	1.2920	1.0510	
	1.1895	1.0455	1.1861	1.0495	1.1935	1.2509	1.0398	
	1.9756	1.3869	1.7621	1.0005	.8798	3.2856	1.0771	
14	1.0070	1.3300	.9820	1.2870	1.2030	1.0510		
	.9927	1.2993	.9717	1.2625	1.1868	1.0398		
	1.4405	2.3628	1.0600	1.9406	1.3650	1.0771		
15	1.1450	1.1060	1.0950	.9850	SIMULATE-3			
	1.1215	1.0799	1.0646	.9623	PDQ			
	2.0954	2.4169	2.8555	2.3589	PERCENT ERROR			

Figure 4-9

Catawba Unit 1 EOC-1
PDQ vs. SIMULATE-3P Peak Pin Powers

	H	G	F	E	D	C	B	A
8	.9540	1.0650	.9560	1.0980	1.0210	1.1800	1.0570	1.1840
	.9322	1.0478	.9346	1.0757	.9991	1.1619	1.0369	1.1607
	2.3386	1.6415	2.2897	2.0731	2.1920	1.5578	1.9385	2.0074
9	1.0650	.9500	1.0710	.9850	1.1410	1.0580	1.2970	1.1730
	1.0478	.9302	1.0535	.9639	1.1262	1.0402	1.2795	1.1352
	1.6415	2.1286	1.6611	2.1890	1.3142	1.7112	1.3677	3.3298
10	.9560	1.0710	.9730	1.1140	1.0340	1.1690	1.0390	1.1350
	.9346	1.0535	.9525	1.1028	1.0165	1.1578	1.0237	1.1211
	2.2897	1.6611	2.1522	1.0156	1.7216	.9674	1.4946	1.2399
11	1.0980	.9850	1.1140	1.0380	1.1820	1.0610	1.2640	1.0680
	1.0757	.9639	1.1028	1.0211	1.1773	1.0483	1.2524	1.0337
	2.0731	2.1890	1.0156	1.6551	.3992	1.2115	.9262	3.3182
12	1.0210	1.1410	1.0340	1.1820	1.2320	1.1880	1.2310	
	.9991	1.1262	1.0165	1.1773	1.2212	1.1850	1.2044	
	2.1920	1.3142	1.7216	.3992	.8844	.2532	2.2086	
13	1.1800	1.0580	1.1690	1.0610	1.1880	1.3050	1.1080	
	1.1619	1.0402	1.1578	1.0483	1.1850	1.2783	1.0928	
	1.5578	1.7112	.9674	1.2115	.2532	2.0887	1.3909	
14	1.0570	1.2970	1.0390	1.2640	1.2310	1.1080		
	1.0369	1.2795	1.0237	1.2524	1.2044	1.0928		
	1.9385	1.3677	1.4946	.9262	2.2086	1.3909		
15	1.1840	1.1730	1.1360	1.0680	SIMULATE-3			
	1.1607	1.1352	1.1211	1.0337	PDQ			
	2.0074	3.3298	1.3291	3.3182	PERCENT ERROR			

Figure 4-10

Catawba Unit 1 BOC-2
PDQ vs. SIMULATE-3P Peak Pin Powers

	H	G	F	E	D	C	B	A
8	.9700	1.3920	1.3560	1.3070	1.0510	1.1940	1.1800	1.3310
	.9427	1.3931	1.3602	1.3156	1.0416	1.1809	1.1591	1.2852
	2.8959	-.0790	-.3088	-.6537	.9025	1.1093	1.8031	3.5636
9	1.3920	1.3770	1.3440	1.1200	1.2990	1.0190	1.1690	1.2650
	1.3983	1.3815	1.3558	1.1159	1.3098	.9964	1.1700	1.2236
	-.4505	-.3257	-.8703	.3674	-.8246	2.2682	-.0855	3.3835
10	1.3560	1.3440	1.1450	1.3830	1.0370	1.3030	1.1710	1.1970
	1.3678	1.3578	1.1355	1.3793	1.0240	1.2773	1.1420	1.1731
	-.8627	-1.0163	.8366	.2683	1.2695	2.0121	2.5394	2.0373
11	1.3070	1.1140	1.3800	.9830	1.2210	1.0020	1.2300	.6790
	1.3191	1.1169	1.3790	.9632	1.2239	.9800	1.2099	.6582
	-.9173	-.2596	.0725	2.0556	-.2369	2.2449	1.6613	3.1601
12	1.0510	1.2940	1.0370	1.2210	.9210	1.1100	1.1550	
	1.0432	1.3116	1.0248	1.2243	.9073	1.1077	1.1380	
	.7477	-1.3419	1.1905	-.2695	1.5100	.2076	1.4938	
13	1.1940	1.0260	1.3090	1.0050	1.1100	1.2820	.6770	
	1.1824	.9974	1.2784	.9806	1.1078	1.2731	.6669	
	.9811	2.8675	2.3936	2.4883	.1986	.6991	1.5145	
14	1.1800	1.1770	1.1770	1.2340	1.1570	.6780		
	1.1604	1.1711	1.1430	1.2106	1.1384	.6670		
	1.6891	.5038	2.9746	1.9329	1.6339	1.6492		
15	1.3310	1.2710	1.2030	.6810	SIMULATE-3			
	1.2866	1.2249	1.1743	.6587	PDQ			
	3.4510	3.7636	2.4440	3.3855	PERCENT ERROR			

Figure 4-11

Catawba Unit 1 MOC-2
PDQ vs. SIMULATE-3P Peak Pin Powers

	H	G	F	E	D	C	B	A
8	.8380	1.1380	1.1250	1.1540	.9970	1.1370	1.1110	1.2030
	.8180	1.1293	1.1185	1.1517	.9889	1.1273	1.0852	1.1631
	2.4450	.7704	.5811	.1997	.8191	.8605	2.3774	3.4305
9	1.1380	1.1330	1.1940	1.0740	1.2260	1.0260	1.1630	1.1670
	1.1313	1.1197	1.1996	1.0708	1.2287	1.0115	1.1595	1.1261
	.5922	1.1878	-.4668	.2988	-.2197	1.4335	.3019	3.6320
10	1.1250	1.1910	1.0960	1.3690	1.0900	1.3490	1.1710	1.1580
	1.1212	1.1994	1.0842	1.3695	1.0752	1.3380	1.1436	1.1296
	.3389	-.7004	1.0884	-.0365	1.3765	.8221	2.3959	2.5142
11	1.1540	1.0710	1.3680	1.0380	1.3540	1.0870	1.2850	.7020
	1.1522	1.0707	1.3694	1.0279	1.3461	1.0683	1.2657	.6976
	.1562	.0280	-.1022	.9826	.5869	1.7504	1.5248	.6307
12	.9970	1.2240	1.0900	1.3540	1.0620	1.1770	1.2170	
	.9889	1.2285	1.0751	1.3459	1.0441	1.1582	1.1874	
	.8191	-.3663	1.3859	.6018	1.7144	1.6232	2.4928	
13	1.1370	1.0290	1.3500	1.0880	1.1770	1.3290	.7200	
	1.1272	1.0113	1.3379	1.0681	1.1581	1.2982	.7203	
	.8694	1.7502	.9044	1.8631	1.6320	2.3725	-.0416	
14	1.1110	1.1660	1.1730	1.2860	1.2170	.7200		
	1.0850	1.1593	1.1435	1.2656	1.1873	.7203		
	2.3963	.5779	2.5798	1.6119	2.5015	-.0416		
15	1.2030	1.1700	1.1600	.7020	SIMULATE-3			
	1.1630	1.1259	1.1295	.6975	PDQ			
	3.4394	3.9169	2.7003	.6452	PERCENT ERROR			

Figure 4-12

Catawba Unit 1 EOC-2
PDQ vs. SIMULATE-3P Peak Pin Powers

	H	G	F	E	D	C	B	A
8	.8520	1.1040	1.0900	1.1270	.9880	1.1160	1.0770	1.1460
	.8143	1.0809	1.0614	1.1032	.9674	1.0981	1.0557	1.1291
	4.6297	2.1371	2.6946	2.1574	2.1294	1.6301	2.0176	1.4968
9	1.1040	1.0990	1.1760	1.0680	1.2110	1.0240	1.1440	1.1170
	1.0819	1.0632	1.1636	1.0497	1.2025	1.0053	1.1425	1.0998
	2.0427	3.3672	1.0657	1.7434	.7069	1.8601	.1313	1.5639
10	1.0900	1.1750	1.0980	1.3780	1.1110	1.3510	1.1480	1.1280
	1.0630	1.1633	1.0699	1.3664	1.0864	1.3444	1.1265	1.1124
	2.5400	1.0058	2.6264	.8489	2.2644	.4909	1.9086	1.4024
11	1.1270	1.0660	1.3780	1.0750	1.3800	1.1010	1.2670	.7230
	1.1033	1.0496	1.3662	1.0583	1.3721	1.0806	1.2632	.7240
	2.1481	1.5625	.8637	1.5780	.5758	1.8878	.3008	-.1381
12	.9880	1.2090	1.1100	1.3800	1.0950	1.1520	1.2000	
	.9673	1.2024	1.0864	1.3720	1.0738	1.1404	1.1814	
	2.1400	.5489	2.1723	.5831	1.9743	1.0172	1.5744	
13	1.1160	1.0260	1.3510	1.1010	1.1520	1.2790	.7230	
	1.0980	1.0052	1.3444	1.0806	1.1404	1.2597	.7375	
	1.6393	2.0692	.4909	1.8878	1.0172	1.5321	-1.9661	
14	1.0770	1.1460	1.1490	1.2670	1.1990	.7220		
	1.0557	1.1425	1.1265	1.2632	1.1814	.7374		
	2.0176	.3063	1.9973	.3008	1.4898	-2.0884		
15	1.1460	1.1200	1.1290	.7230	SIMULATE-3			
	1.1291	1.0998	1.1124	.7241	PDQ			
	1.4968	1.8367	1.4923	-.1519	PERCENT ERROR			

5.0 APPLICATION

The SIMULATE-3P methodology is applicable to the following calculations.

A) Core Loading Pattern Determination

B) Input to Safety Analyses

C) Startup and Operational Data Predictions

D) Reactor Protection System and Core Operating Limits

Items A, B, and C involve a relatively straightforward substitution of the new models for the currently utilized models. References 2 and 3 specify that PDQ07 will be used to verify core loading patterns. This report substitutes SIMULATE-3P in place of PDQ07. Where References 2 and 3 specify that EPRI-NODE-P will be used for input to safety analyses, this report allows substitution of SIMULATE-3P for EPRI-NODE-P. Similarly, where References 2, 3, and 6 specify that EPRI-NODE-P will be used to generate startup and operational data, SIMULATE-3P may be substituted for EPRI-NODE-P. However, item D requires utilization of the ONRFs provided in Section 4 as described below.

When calculating peaking factors for generating RPS and operating limits the calculated peaks must be multiplied by the appropriate uncertainty factors. For total and radial peaks this uncertainty consists of contributions due to average biases between measured and calculated assembly data, random uncertainties between measured and calculated assembly data, and random uncertainties between measured and calculated local pin peaking. Each of these contributions may be derived from Section 4.0 as shown:

$$\text{Bias} = -\frac{\overline{D}}{\overline{M}} \quad \text{(From Table 4-5)}$$

$$\text{Assembly Uncertainty} = \frac{K \times S(D)}{\overline{M}} \quad \text{(From Table 4-5)}$$

Local Pin Peaking Uncertainty (U_{R-L}) = 0.02 (From Section 4.2(a))

For axial peaking factors the appropriate uncertainty does not include the local pin peaking uncertainty.

5.1 Total Peaking Uncertainty Factors

When using SIMULATE-3P as a stand-alone model for calculating total peaking factors (F_Q) the appropriate nuclear uncertainty is 5.6% for the B&W plants and 6.1% for the Westinghouse plants. When EPRI-NODE-P is used, as normalized to SIMULATE-3P, this value becomes 5.7% for the B&W plants and 5.7% for the Westinghouse plants. These values do not include the engineering uncertainty factor required to account for manufacturing tolerances as this is fuel vendor specific. The individual components of these total nuclear uncertainty factors are shown below.

Plant	Model	Bias	Assembly Total Uncertainty (U_{A-T})	Radial-Local Uncertainty (U_{R-L})	Total Nuclear Uncertainty (UCT)
B&W	SIMULATE-3P	0.013	0.038	0.020	1.056
B&W	EPRI-NODE-P	-0.009	0.063	0.020	1.057
Westinghouse	SIMULATE-3P	0.021	0.035	0.020	1.061
Westinghouse	EPRI-NODE-P	0.006	0.047	0.020	1.057

where,

$$UCT = 1 + Bias + \sqrt{U_{A-T}^2 + U_{R-L}^2}$$

The engineering uncertainty factor may be statistically combined and will be applied where appropriate per References 1 and 8.

5.2 Radial Peaking Uncertainty Factors

When using SIMULATE-3P as a stand-alone model for calculating radial peaking factors $(F_{\Delta H})$, the appropriate nuclear uncertainty is 3.4% for the B&W plants and 2.6% for the Westinghouse plants. When EPRI-NODE-P is used, as normalized by SIMULATE-3P, this value becomes 4.9% for the B&W plants and 3.8% for the Westinghouse plants. The individual components of these radial nuclear uncertainty factors are shown below.

Plant	Model	Bias	Assembly Radial Uncertainty (U_{A-R})	Radial-Local Uncertainty (U_{R-L})	Radial Nuclear Uncertainty (UCR)
B&W	SIMULATE-3P	0.000	0.027	0.020	1.034
B&W	EPRI-NODE-P	0.004	0.040	0.020	1.049
Westinghouse	SIMULATE-3P	0.000	0.016	0.020	1.026
Westinghouse	EPRI-NODE-P	0.004	0.028	0.020	1.038

where,

$$UCR = 1 + Bias + \sqrt{U_{A-R}^2 + U_{R-L}^2}$$

The additional engineering uncertainty factor may be statistically combined and will be applied where appropriate per References 1 and 8.

5.3 Axial Peaking Uncertainty Factors

When using SIMULATE-3P as a stand-alone model for calculating axial peaking factors (F_z) the appropriate uncertainty is 4.4% for the B&W plants and 5.3% for the Westinghouse plants. When EPRI-NODE-P is used, as normalized to SIMULATE-3P, this value becomes 3.0% for the B&W plants and 4.3% for the Westinghouse plants. The individual components of these axial nuclear uncertainty factors are shown below.

Plant	Model	Bias	Assembly Axial Uncertainty (U_{A-A})	Axial Nuclear Uncertainty (UCA)
B&W	SIMULATE-3P	0.010	0.034	1.044
B&W	EPRI-NODE-P	-0.013	0.043	1.030
Westinghouse	SIMULATE-3P	0.023	0.030	1.053
Westinghouse	EPRI-NODE-P	0.004	0.038	1.043

where,

$$UCA = 1 + Bias + U_{A-A}$$

The additional engineering uncertainty factor may be statistically combined and will be applied where appropriate per Reference 1 and 8.

6.0 CONCLUSION

This report justifies the use of CASMO-3 based SIMULATE-3P and EPRI-NODE-P models for reload design of B&W 177-assembly plants and Westinghouse 193-assembly plants. Since detailed descriptions and validation of the programs involved have already been provided to the NRC in References 4 and 5, this report demonstrates Duke Power Company's competence in their application. The methodology presented here supplements previous topical reports submitted by Duke Power Company (References 1,2,3,6, and 8) by providing alternative core models for performing reload design calculations.

Extensive benchmarking of this methodology to previous cycles of operation is presented. This includes comparisons of calculated and measured data from BOC, HZP startup testing, operating data, and near-EOC HFP ITC measurements. Derivation of peaking factor uncertainties for application of this methodology to calculation of core operating and Reactor Protection System limits is also provided. These uncertainties are derived in a manner consistent with the previously approved methods of References 1, 2, 3, and 8.

SIMULATE-3P calculations to determine pin power distributions are acceptable for replacing fine-mesh PDQ calculations. Core average physics parameters and three-dimensional nodal power distributions may also be calculated with SIMULATE-3P. The option to use EPRI-NODE-P for three-dimensional nodal power distributions is justified and retained for calculationally intense analyses such as determining operating and Reactor Protection System setpoints.

7.0 REFERENCES

1. "Duke Power Company Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002A, October 1985.

2. "Duke Power Company Oconee Nuclear Station Reload Design Methodology," NFS-1001A, April 1984.

3. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Nuclear Physics Methodology for Reload Design," DPC-NF-2010A, June 1985.

4. "Yankee Atomic Electric Company, CASMO-3G Validation," YAEC-1363, April 1988.

5. "Yankee Atomic Electric Company, SIMULATE-3 Validation and Verification," YAEC-1659, September 1988.

6. "Rod Swap Methodology Report for Startup Physics Testing," DPC-NE-1003A, Duke Power Company, December 1986.

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

8. "Duke Power Company, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," DPC-NE-2011P, April 1988.

9. "Safety Analysis Physics Parameters and Multidimensional Reactor Transients Methodology", DPC-NE-3001P, Duke Power Company, to be issued.

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DUKE POWER

January 26, 1990

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Oconee Nuclear Station
           Docket Nos. 50-269, -270, -287
           McGuire Nuclear Station
           Docket Nos. 50-369, -370
           Catawba Nuclear Station
           Docket Nos. 50-413, -414
           Nuclear Design Methodology Using
           CASMO-3/SIMULATE-3P, DPC-NE-1004

Gentlemen:

Please find enclosed for your review twenty-three copies of topical report
DPC-NE-1004, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P". This
report presents the CASMO-3/SIMULATE-3P code sequence as a reactor model for
use in reload design calculations. Descriptions of the programs utilized
benchmark comparisons of measured data to calculational results, code
specific reliability factors, and a description of the application of these
programs to reload design are provided. These programs are presented as an
alternative to those previously described in reports NFS-1001A, DPC-
NF-2010A, DPC-NE-1002A, DPC-NE-1003A, and DPC-NE-2011P.

Also included are copies of Duke Power Company's calculational results using
the CASMO-3/SIMULATE-3P code sequence to evaluate two standard benchmark
problems. The problem descriptions were provided to Duke Power Company, and
copies sent to representatives of the U.S. Nuclear Regulatory Commission,
via letters from J. F. Carew (Brookhaven National Laboratory) dated
April 18, 1989. These results are provided to aid evaluation of Duke Power
Company's competence in the use of the CASMO-3/SIMULATE-3P code sequence.

If there are any questions, or you require additional information, please
call Scott Gewehr at (704) 373-7581.

Very truly yours,

s/Hal B. Tucker
Hal B. Tucker

SAG204/lcs

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DUKE POWER

January 23, 1992

U. S. Nuclear Regulatory Commission
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Subject: McGuire Nuclear Station
          Docket Numbers 50-369 and -370
          Oconee Nuclear Station
          Docket Numbers 50-269,-270, and -287
          Catawba Nuclear Station
          Docket Numbers 50-413 and -414
          Topical Report DPC-NE-1004: Response to Request for
          Additional Information

By letter dated January 26, 1990, Duke Power submitted Topical
Report DPC-NE-1004, "Nuclear Design Using CASMO-3/SIMULATE-3P." By
letter dated December 6, 1991, the NRC responded with a set of
questions pertaining to the Topical Report.

Attached are the responses to those questions.

If there are any questions, or if we may be of further assistance
in your review, please call Scott Gewehr at (704) 373-7581.

Very truly yours,

H. B. Tucker

QA1004/sag

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U. S. Nuclear Regulatory Commission
January 23, 1992
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DPC-NE-1004
Response to Request for Additional Information

1.   Question: Identify the "data based on ENDF/B-V" used in the CASMO-3 calculation, as stated in Section 2.1(a), and the data based on ENDF/B-IV.

Answer: The majority of the data in the CASMO-3 library is based on ENDF/B-IV. The data selected from ENDF/B-V which is of primary importance includes the Xenon-135 yields and fission spectra data for U-235 and Pu-239. Additionally, the resonances for U-238 have been adjusted to agree with Hellstrand's resonance integral measurements.

2.   Question: When is the 70 energy group cross-section library used in the CASMO-3 calculation and when is the 40 group library used?

Answer: All of the calculations used as the basis for DPC-NE-1004 utilized the 40 energy group library. However, DPC would like to reserve the right to use the 70 group library. As noted in the response to question #11, CASMO-2 is currently used to generate ratios of Rhodium detector reaction rates to the power in the 8 surrounding fuel pins. This is done using the 69 group library for CASMO-2. It may, likewise, be necessary to use the 70 group library with CASMO-3 when generating these ratios. It would be expected that use of the 70 group library would yield as good or better results based on the additional detail of the calculations.

3.   Question: Identify the three-dimensional calculations which DPC will perform using the SIMULATE-3P sequence and the EPRI-NODE-P sequence. How many axial nodes will be used in either one?

Answer: DPC intends to use SIMULATE-3P as a substitute for PDQ and be supplemented by EPRI-NODE-P similarly to the descriptions provided in references 1, 2, 3, 6, 8, and 9 or to utilize SIMULATE-3P as a substitute for any EPRI-NODE-P as well as PDQ calculations. The applicable uncertainties for the particular code selected will then be applied as provided in section 5.0.
        The SIMULATE-3P models for all units utilize 12 axial fuel nodes plus one top and one bottom reflector node. The EPRI-NODE-P model for McGuire and Catawba utilizes 18 axial fuel nodes. For Oconee, the EPRI-NODE-P model utilizes 16 axial fuel nodes. The increased number of axial nodes in the EPRI-NODE-P models is required because the program does not model the intranodal peaking which SIMULATE-3P includes.

4.   Question: Are any adjustments made in the SIMULATE-3P model in order to improve comparisons with measurements?

Answer: There are no adjustments made to the SIMULATE-3P model to enhance the comparisons.

5.. Question: Have any adjustments been made in any of the SIMULATE-3P models (BOC, HZP, ARO) used in the determination of the critical boron concentration for the reactor cores listed in Table 3-1? If so, please explain.

Answer: No adjustments have been made to the SIMULATE-3P model for HZP conditions. The cross-section libraries utilized for HFP conditions also cover the range down to HZP.

6. Question: Have any cycle-to-cycle adjustments been made in the SIMULATE-3P models of the reactor cycles listed in Tables 3-2 through 3-6 and Figures 3-6 to 3-14, other than those changes due to operating conditions?

Answer: No.

7. Question: Explain the observed SIMULATE-3P HFP critical boron concentration underpredictions for Catawba Unit 2, Cycle 3, Oconee Unit 2, Cycles 9 and 10 and overpredictions for the McGuire Unit 2, Cycles 2 through 5.

Answer: A general explanation of possible causes for mispredictions of critical boron concentrations has been provided in response to question #12. The results for Catawba Units 1 and 2 are very consistent and demonstrate consistent trends with McGuire Unit 2, with the initial cycle having a more negative result than the reload cycles. The McGuire Unit 2 results appear to have a bias as compared to the Catawba units with the initial cycle having a small overprediction for McGuire Unit 2 and a small underprediction for Catawba Units 1 and 2. Because the same fuel types and similar models are used for the McGuire and Catawba units, this bias may be attributed to plant specifics such as as-built plant and fuel differences. The results for Oconee Units 2 and 3 show very consistent trends in their underpredictions. The difference in trends between the B&W versus Westinghouse units is indicative of differences due to plant design, plant procedures, fuel designs, fuel vendors, etc. In the later cycles modelled for Oconee Units 2 and 3 there were some difficulties encountered with fouling of the Venturi tubes which are used in measuring the power level. This may account for the small changes observed in the Oconee Unit 2, Cycle 9 and 10 results. However, because the error in the measured power levels is unknown, the quantitative effect of the fouling cannot be determined. Please note that there were no HFP boron comparisons provided for Catawba Unit 2, Cycle 3 as the cycle had not yet completed operation when the benchmark calculations were

performed. Subsequent discussions with the NRC representative identified that the question should have referred to Catawba Unit 2, Cycle 1 instead of Cycle 3.

8.   Question: How is the dependence of the generic ratios of power to detector reaction rates (used in the conversion of corrected signals) on parameters such as moderator temperature, exposure, and control accounted for?

    Answer: Refer to the answer for question #11.

9.   Question: Describe the method of selection of albedos and/or reflector constants used in the EPRI-NODE-P model. Are these constants changed from cycle to cycle?

    Answer: The axial albedos employed in EPRI-NODE-P were selected to obtain axial offsets and normalized axial peaks which are consistent with the SIMULATE-3P calculations. The radial albedos in EPRI-NODE-P were selected to obtain peripheral assembly powers consistent with SIMULATE-3P. The axial albedos remain the same from cycle to cycle and are only likely to change if fuel assembly/burnable poison designs are changed. The radial albedos are adjusted every cycle to match SIMULATE-3P.

10.   Question: Describe how the measured fission chamber signals are used in the derivation of the relative assembly powers. Describe all corrections/adjustments made to the measured signals to determine the local power.

    Answer: The movable incore fission chambers are intercalibrated for each power distribution measurement. The signals are therefore normalized based on this intercalibration. Sets of predicted assembly and peak pin powers versus predicted fluxes in the instrument location from PDQ are used in conjunction with detector cross sections to convert the normalized signals to relative powers. Sets of data(theoretical factors) are provided as a function of cycle burnup and control rod presence. The basic equation used is:

$$P(measured) = \frac{P(calculated)}{RR(calculated)} * RR(measured)$$

where: P(measured) = measured relative power
           P(calculated) = predicted power from PDQ
           RR(measured) = measured reaction rate
           RR(calculated) = predicted relative reaction rate

When PDQ is used to generate the theoretical factors, the predicted relative reaction rates are calculated using the fluxes from PDQ and a set of predetermined detector cross-sections. When SIMULATE-3P is used instead of PDQ, the relative reaction rates may be calculated directly by SIMULATE-3P.

11.   Question: Why are the rhodium detector signals corrected for a specific core configuration? How is this correction made? How are the signals corrected for rhodium depletion and background current?

       Answer: The corrected signals are obtained by first subtracting the background signal from the raw signal. The background signal is measured by a separate lead wire included in each detector string. The background-corrected signal is then corrected for depletion based on the amount of spent charge over the lifetime of the detector. This spent charge is converted to a depletion correction factor and the signal is corrected back to what would be obtained with a fresh detector based on calculated detector signals versus percent depletion.
       The corrected signals are converted to relative assembly powers by means of precalculated ratios of the power in the surrounding 8 fuel pins to detector reaction rates multiplied by predicted assembly to 8 fuel pin powers. The dependence of the flux distribution within an assembly on the moderator temperature, control rod presence, fuel enrichment, and exposure influences these ratios. Therefore, the ratios of reaction rates to power are generated for a variety of conditions and included in a database. The corrected signals are converted into relative power based on the ratio appropriate to the conditions of the measurement.
       Currently, the ratios of detector reaction rates to the power in the surrounding 8 fuel pins is calculated with CASMO-2 at predetermined burnups, moderator temperatures, enrichments, control rod conditions, etc. Likewise, the ratios of the power in the assemblies to the 8 pins surrounding the detector are calculated as a function of burnup with PDQ. DPC intends to utilize CASMO-3 and SIMULATE-3P for this purpose in the future.

12.   Question: Explain the trend with exposure of the boron difference for Catawba Unit 1, Cycles 1, 2, and 3, and Oconee Cycles 7 through 10. What is causing the significant boron overprediction in the early part of Catawba Unit 1, Cycle 3? Please explain.

       Answer: There are a variety of explanations available for the noted boron differences. Among these are the uncertainties associated with the T_{ave} programs of the individual plants, the heat balance procedure for determining thermal power, the "as-built" Uranium and burnable poison loadings, variability of soluble boron enrichment and B_{10} depletion, and non-

equilibrium reactor conditions. The approach used in performing these benchmarks was intended to reflect the method which would be used to perform reload design analysis. Many of the above factors are not known in advance. Therefore, nominal values were used to set up the models employed in the benchmarks. In practice, where consistent biases are observed through several cycles the biases would be applied to the calculated results. However, for the purpose of the benchmarks no biases have been applied.

13.   Question: There appears to be a systematic overprediction of the boron letdown in the early part of Cycles 3, 4, and 5 of McGuire Unit 2. Please explain.

Answer: One of the possible causes of these overpredictions lies in the specifications for the B_{10} content of the boric acid. This is ordered to a specification of 19.9 ± 0.3 a/o B_{10}. The content assumed in the model is 19.83. This variability could account for 15-20 PPMB of the anomaly shown at the beginning of the mentioned cycles. Other possible causes for the overpredictions have been listed in the response to question #12.

14.   Question: What is the cause of the large percent differences between the measured and SIMULATE-3P calculated BOC, HZP control rod worths of Catawba Unit 1, Cycle 4 (Table 3-7) and McGuire Unit 2, Cycles 2, 3, and 4 (Table 3-9)?

Answer: For the cycles identified, large percentage differences (>10%) are primarily associated with relatively small worths, and are actually small differences in the absolute sense. For those cases where large differences were identified only one case resulted in more than $\pm$ 100 pcm (Catawba 1 Cycle 4, Control Bank B). This case utilized the rod swap measurement technique and resulted in an error of 114 pcm. Based on past experience and known uncertainties with the rod swap test, the acceptance criteria as approved in reference 6 is that the absolute difference be $\leq$ 30% or $\leq$ 200 pcm, whichever is greater. All of the cases identified easily meet this criteria. A better indication of the model's accuracy may be obtained by evaluating the comparisons to measurement in the cases where boron dilution is used to measure the rod worths. This is the case for all of the Catawba 1 Cycle 1 and McGuire 2 Cycle 1 rod worths. This method was also used to measure the reference bank worth (the largest single bank worth) for all of the remaining Catawba and McGuire cycles. For those cases where the boron/dilution method is used, no differences > 10% were observed.

15.   Question: No comparisons are given in Table 3-14 of measured and calculated EOC HFP isothermal temperature coefficients for McGuire Unit 2,

Cycle 1, Catawba Unit 1, Cycle 4 and Catawba Unit 2, Cycle 3. Please discuss.

Answer: At the time the benchmarks were performed and the topical written, Catawba 1 Cycle 4 and Catawba 2 Cycle 3 were still in operation and the EOC ITC measurement had not yet been made. As stated in section 3.3 the McGuire 2 Cycle 1 measurement utilized a different technique than the other cycles which are included. The results from this technique were considered to be suspect and prompted the change in measurement technique.

16. Question: Provide the following design data for the fuel assemblies for which comparisons of calculated and measured relative radial assembly powers are presented in Figures 3-6 through 3-14: fuel pin layout, range of enrichment and burnable poison loading, and core average exposure. Discuss to what extent these fuel designs cover the intended DPC applications. Is there any correlation between the specific fuel designs and the observed large calculation-to-measurement differences?

Answer: For these specific figures the range of enrichment was from 3.10 to 3.40 w/o U$_{235}$ initial enrichment, the burnable poison loadings ranged from 0.20 to 1.10 w/o B4C for Oconee and for Catawba/McGuire used 4 or 8 finger BPs at 13.5 w/o B4C, and the core average exposures ranged from 11.726 - 25.715 GWD/MTU. The applicable fuel pin layouts are shown below:

Catawba/McGuire

```
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 3 1 1 3 1 1 3 1 1 1 1 1 1
1 1 1 3 1 1 1 1 1 1 1 3 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 3 1 1 3 1 1 3 1 1 3 1 1 3 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 3 1 1 3 1 1 2 1 1 3 1 1 3 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 3 1 1 3 1 1 3 1 1 3 1 1 3 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 3 1 1 1 1 1 1 1 3 1 1 1 1
1 1 1 1 3 1 1 3 1 1 3 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
```

Oconee

```
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 3 1 1 1 3 1 1 3 1 1 1 1 1
1 1 1 3 1 1 1 1 1 1 1 3 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 3 1 1 3 1 1 1 3 1 1 3 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 2 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 3 1 1 3 1 1 1 3 1 1 3 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 3 1 1 1 1 1 1 3 1 1 1
1 1 1 1 3 1 1 1 3 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
```

where: 1 = Fuel Pin Location
            2 = Instrument Tube Location
            3 = Guide Tube/Control Rod/BP Location

Note that, as stated in section 3.4(b), these figures are only representative comparisons from the most recently analyzed cycle at each station. The cycles used to generate the database for reliability factors includes enrichments from 1.6 to 3.6 w/o U_{235}, Oconee BPs from 0.0 to 1.1 w/o B4C, Westinghouse BPs with 0, 4, 6, 8, 9, 10, 12, 15, 16, and 20 fingers (with either 13.5 w/o B4C or 12.5 w/o B2O3 for WABA and Pyrex absorbers, respectively), and core average exposures from 0.0 to 25.714 GWD/MTU. DPC anticipates that this model would be applied up to enrichments of about 4 w/o (or greater if future fuel rack limits permit), BP loading of near 1.4 w/o B4C, BP finger arrangements of up to 24 fingers, and core average exposures of near 30 GWD/MTU. The wide range of data examined provides a high degree of confidence that the model will perform consistently under these conditions.

No correlation has been determined between specific fuel designs and the "observed large calculation-to-measurement differences". Note that the differences shown are given as percentages and that they demonstrate accuracies which are as good as or better than what are typically demonstrated by vendor or other utility comparisons.

17.   Question: Has DPC performed pin power distribution benchmark comparisons with CASMO-3 to justify the use of a pin wise power uncertainty of 2 percent. How is conservatism ensured in the selection of this value?

Answer: DPC has not performed benchmarking of CASMO-3 pin power distributions to critical experiments. The reference 5 submittal by Yankee Atomic Electric Co. did include comparisons of SIMULATE-3P pin power distributions to critical experiments. The ability of CASMO-3 to calculate these distributions is a part of the uncertainty in the SIMULATE-3P predictions. Since SIMULATE-3P was quantitatively benchmarked in reference 5, DPC did not consider it necessary to repeat this benchmarking.

DPC has performed comparisons to critical experiments using CASMO-2 as identified in reference 1. As part of the reference 3 submittal DPC also provided calculations supporting an uncertainty of 1% using the EPRI-CELL/PDQ07 code sequence. However, the NRC at that time indicated that a more conservative value of 2% should be utilized as a reliability factor. Since the reference 5 submittal demonstrated, "SIMULATE-3 to predict pin-by-pin distributions within 1%", it is expected that similar results would be obtained if DPC did a similar benchmarking. Therefore, an uncertainty of

2% is considered to be conservative and would be consistent with the minimum value which the NRC has found to be acceptable for DPC in the past.

18.   Question: Justify the assumption that the data from McGuire/Catawba and Oconee are sufficiently similar to allow a statistical combination to determine calculational uncertainty parameters.

   Answer: McGuire and Catawba are sister Westinghouse 4-loop plants which utilize the same fuel design and similar NSSS systems.   The data for McGuire/Catawba has not been combined with that from Oconee in generating the reliability factors. Separate reliability factors for the Westinghouse units and the B&W units have been provided in section 5.   Data was tabulated for all units combined(Westinghouse and B&W) and each individual cycle in section 3 for demonstrative purposes only.

19.   Question: In the McGuire/Catawba and Oconee data base, why is N for the radial NODE-P data smaller than its corresponding value for the SIMULATE-3P data? Have any NODE-P or SIMULATE-3P data been dropped from the data base? If so, justify this deletion.

   Answer: As stated in section 4.2(b), the reliability factors are based on comparisons of calculated and measured values only where both values are ≥ 1.0. This is based on the assumption that the primary importance of these factors is related to conservatively calculating the limiting peak powers in the core. Because this is a comparison of relative power distributions, all axial data will have values which are ≥ 1.0. However, radial comparisons were included in the database for cases where the measured value was ≥ 1.0 only if the calculated value for the particular model was also ≥ 1.0. This resulted in there being a different number of valid data points for the different models.

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DUKE POWER

March 30, 1992

U.S. Nuclear Regulatory Commission
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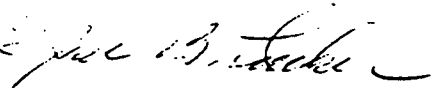
Subject: Oconee Nuclear Station
            Docket Nos. 50-269, -270, -287
            McGuire Nuclear Station
            Docket Nos. 50-369, -370
            Catawba Nuclear Station
            Docket Nos. 50-413, -414
            Nuclear Design Methodology Using CASMO-3/SIMULATE-3P,
            DPC-NE-1004; Response to Request for Information

The subject Topical Report was submitted for NRC review on January
26, 1990. On December 6, 1991, the NRC Staff transmitted a set of
questions regarding the Report; these questions were responded to
by Duke via letter dated January 23, 1992.

A subsequent telephone call between the NRC Staff, Duke, and the
contract reviewer identified one follow-up question.   That
question, and the response, is attached.

If we can be of further assistance in your review please call Scott
Gewehr at (704) 373-7581.

Very truly yours,

H. B. Tucker

QA1004/sag

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Question. Provide additional information to justify the 2.0% pin power uncertainty used in the calculation of uncertainty factors.

Answer. A series of three Babcock and Wilcox (B&W) critical experiments were evaluated to assess the capability of the CASMO-3/SIMULATE-3P code sequence to accurately calculate pin power distributions. The criticals modelled included core geometries comprised of 2.46 w/o and 4.02 w/o fuel, in addition to assembly geometries with both small and large water holes. The latter assembly geometry is based on Combustion Engineering's 16X16 fuel assembly design. Measured pin powers were obtained from reference 1 for cores 1, 12 and 18. Only the power distribution in the center assembly was measured by counting the fission product gammas produced from each fuel pin following irradiation. Each fuel pin was measured three times and the results averaged, and then normalized so that the average power in the center assembly would be unity. Also, all measurements were performed at the mid-plane of the core.

A three-dimensional SIMULATE-3P model was developed to evaluate the criticals. This model consisted of four nodes per assembly and seven axial nodes. CASMO-3 was used to generate cross sections and assembly discontinuity factors. Small compromises to the as-built core geometries had to be made in order to model the criticals with SIMULATE-3P. These compromises were necessary because of the inability of SIMULATE-3P to model partial fuel assemblies, but were restricted to the core periphery. Judging from the results of the evaluation, the compromises to the as-built core geometries were negligible.

Comparisons between the measured and SIMULATE-3P predicted pin power distribution for cores 1, 12 and 18 are presented in Figures 1 through 3, respectively. Table 1 contains a summary of the average absolute percent difference and the standard deviation of the percent difference for the cores evaluated. Agreement between SIMULATE-3P predicted and measured pin powers was excellent. The maximum pin error observed in any core location is less than 2.1%. From these results it can be concluded that the CASMO-3/SIMULATE-3P code sequence can accurately predict the pin-by-pin power distribution within an assembly.

A pin Observed Nuclear Reliability Factor (ONRF) was developed using all of the data from cores 1, 12 and 18, and was based on the methodology outlined in section 4.2 of this report. The normality of the data set was confirmed using the D' normality test. The pin ONRF that was calculated based on a 95% probability, and 95% confidence level was 1.0167. Therefore, the 2.0% pin uncertainty factor used in the calculation of the power distribution uncertainty factors in DPC-NE-1004 is conservative.

REFERENCES

1.   B&W Report, "Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark", prepared for U.S. Dept. of Energy Under Contract DE-AC02-78 ET 34212, BAW 1810, April 1984.

Table 1

**Difference Summary: Differences in the Predicted SIMULATE
Power Distribution for the B&W Criticals**

	Core 1 15X15 lattice 2.46 w/o fuel	Core 12 15X15 lattice 4.02 w/o fuel	Core 18 16X16 lattice 4.02 w/o fuel
Avg. Abs. + % Diff.	0.5186	0.6763	0.8561
Standard Dev. of % Diff	0.6460	0.8498	1.0395

+ Avg. Abs % Diff. = [|(C-M)|/M]*100

Figure 1
Core 1 - 2.46 w/o Fuel
15X15 Lattice

SPND	1.032	1.003	0.990	0.985	0.983	0.966	0.946
	1.018	1.011	0.987	0.981	0.997	0.966	0.945
	1.375	-0.791	0.304	0.408	-1.404	0.000	0.106
	1.026	1.064	1.014	1.011	1.049	0.985	0.950
	1.019	1.067	1.012	1.009	1.058	0.999	0.945
	0.687	-0.281	0.198	0.198	-0.851	-1.401	0.529
		WATER	1.080	1.080	**WATER**	1.038	0.954
			1.081	1.090		1.032	0.953
			-0.093	-0.917		0.581	0.105
			1.056	1.100	1.082	0.991	0.948
			1.054	1.104	1.086	0.989	0.945
			0.190	-0.362	-0.368	0.202	0.317
				WATER	1.054	0.965	0.939
					1.059	0.965	0.934
					-0.472	0.000	0.535
					0.984	0.946	0.928
					0.988	0.938	0.923
					-0.405	0.853	0.542
			Calculated (C)			0.931	0.919
			Measured (M)			0.925	0.914
			% Diff [(C-M)/M]*100			0.649	0.547
							0.910
							0.903
							0.775

Figure 2
Core 12 - 4.02 w/o Fuel
15X15 Lattice

SPND	1.088	1.030	1.011	0.998	0.987	0.958	0.926
	1.075	1.041	1.006	1.019	1.000	0.960	0.923
	1.209	-1.057	0.497	-2.061	-1.300	-0.208	0.325
	1.069	1.118	1.038	1.026	1.075	0.980	0.929
	1.067	1.125	1.044	1.034	1.075	0.987	0.927
	0.187	-0.622	-0.575	-0.774	0.000	-0.709	0.216
		WATER	1.121	1.114	**WATER**	1.050	0.930
			1.114	1.118		1.034	0.942
			0.628	-0.358		1.547	-1.274
			1.076	1.132	1.104	0.977	0.919
			1.083	1.137	1.102	0.979	0.908
			-0.646	-0.440	0.181	-0.204	1.211
				WATER	1.064	0.940	0.902
					1.071	0.939	0.895
					-0.654	0.106	0.782
					0.965	0.913	0.884
					0.958	0.900	0.883
					0.731	1.444	0.113
				Calculated (C)		0.888	0.865
				Measured (M)		0.884	0.856
				% Diff [(C-M)/M]*100		0.452	1.051
							0.845
							0.845
							0.000

Figure 3
Core 18 - 4.02 w/o Fuel
CE 16X16 Lattice

SPND	1.215 1.205 0.830	1.035 1.033 0.194	1.003 0.997 0.602	0.988 0.977 1.126	0.971 0.959 1.125	0.950 0.941 0.956	0.925 0.909 1.760
	1.082 1.076 0.558	1.033 1.021 1.175	1.027 1.012 1.482	1.013 1.010 0.297	0.983 0.982 0.102	0.952 0.946 0.634	0.924 0.912 1.316
		1.077 1.065 1.127	1.204 1.228 -1.954	1.190 1.203 -1.081	1.036 1.043 -0.671	0.959 0.957 0.209	0.920 0.928 -0.862
			WATER	**WATER**	1.163 1.183 -1.691	0.966 0.974 -0.821	0.915 0.924 -0.974
				WATER	1.149 1.170 -1.795	0.954 0.970 -1.649	0.904 0.909 -0.550
			Calculated (C)		0.999	0.924	0.886
			Measured (M)		0.995	0.924	0.886
			% Diff [(C-M)/M]*100		0.402	0.000	0.000
						0.894 0.893 0.112	0.866 0.866 0.000
							0.842 0.833 1.080

U. S. Nuclear Regulatory Commission
March 30, 1992
Page 3

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Section C

(Letter from M. S. Tuckman (Duke Power) to U. S. Nuclear Regulatory Commission, "Duke Power Topical Report DPC-NE-1004A; Minor Revision, December 12, 1995.)

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DUKE POWER

December 12, 1995

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Duke Power Topical Report DPC-NE-1004A; Minor Revision

Reference: Duke Power Company, Nuclear Design Methodology Using
CASMO-3/SIMULATE-3P, DPC-NE-1004A, November 1992

As part of Duke Power Company's continuing effort to improve its reload design
methods, the CASMO-3/SIMULATE-3P power distribution uncertainty factors have been
re-evaluated using measured data from recent Catawba and McGuire fuel cycles. In
addition, the benchmarking included an increase in the number of SIMULATE-3P axial
nodes from the 12 axial nodes used in the above Reference to 24 axial nodes. The
increased axial nodalization will allow explicit modeling of axial blanket fuel segments.

The results of the new benchmarking analysis are shown below in the Table. The
Observed Nuclear Reliability Factors (ONRFs) for the assembly radial, axial, and total
peaking are compared with the ONRFs documented in the Reference. This comparison
shows that the axial and total peaking uncertainty factors decreased for the new
benchmarking and the assembly radial power peaking increased slightly from 1.017 to
1.020. The statistical treatment used in this analysis is identical to that described in the
above referenced Topical Report and the calculations are documented in a safety related
calculation.

Observed Nuclear Reliability Factors

Type	Original Factors DPC-NE-1004A	24 Level/ Recent Cycles
Radial	1.017	1.020
Axial	1.053	1.031
Total	1.057	1.037
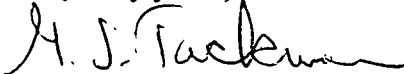
U. S. Nuclear Regulatory Commission
December 12, 1995
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The more recent cycles include cores that are longer in cycle length, have higher fuel enrichments, and contain more burnable poisons than earlier cores. One cycle also includes fuel with natural UO_2 axial blankets. The cores used in this benchmarking were McGuire 1 Cycle 9, McGuire 2 Cycle 9, Catawba 2 Cycle 6, Catawba 1 Cycle 7, and Catawba 2 Cycle 7.

Duke Power is planning to use the ONRF values listed in the second column of the above table with 24 axial level SIMULATE-3P models for McGuire and Catawba. This model would first be used on the Catawba 1 Cycle 10 fuel cycle design.

If additional information is needed before this minor change is implemented, please call Scott Gewehr at (704) 382-7581.

Very truly yours,

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December 12, 1995
Page 3

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Section D

[Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Revision 1. (Includes NRC Request for Additional Information, Acceptance Letter and Safety Evaluation Report)]

Nuclear Design Methodology Using CASMO-3/SIMULATE-3P
DPC-NE-1004A
Revision 1

Abstract

On December 12, 1995, and as supplemented on April 9, 1996, Duke Power submitted revision 1 to DPC-NE-1004A. This submittal requested a review of new reliability and uncertainty factors applicable to Westinghouse reactors based on a 24 axial level SIMULATE-3P model, versus the 12 axial level model used in the original analysis. This re-evaluation of reliability and uncertainty factors was performed to capture the effects of current reload design strategies versus the strategies employed in the original uncertainty factor determination. An SER was issued by the Nuclear Regulatory Commission on April 26, 1996 accepting the revision.

The results of the statistical analysis and the calculation of observed nuclear reliability factors (ONRFs) and uncertainty factors are included in this Section. The NRC's request for additional information, including Duke Power Company's responses, and the NRC SER are also included this Section.

1.0 Introduction

The development of new observed nuclear reliability and uncertainty factors applicable to Westinghouse reactors using a 24 axial level CASMO-3/SIMULATE-3P model was performed for several reasons. These reasons are:

- To include the effects of longer cycle lengths, higher enriched fuel, higher BPRA loadings, higher burnup and axial blankets in the development of SIMULATE-3 uncertainty factors,

- To better account for axial heterogeneities characteristic of axial blanket fuel in predicted and measured power distributions (i.e. axial dependency of power to reaction rate ratio) and

- To increase the accuracy of the calculation by decreasing the axial node size from 12 to 6 inches.

The transition from 12 to 24 axial levels was performed for the following reasons:

- SIMULATE-3 has a coding limitation on the number of axial regions which can be modelled within one node. For axial blanket fuel with burnable poisons rods, the nodal length must be decreased to less than 12 inches in order to satisfy this criteria. Axial node segments of 6, or 8 inches satisfy this requirement.

- For a 24 axial levels model, the nodal boundaries match up exactly with the transition from the blanketed fuel region to the non-blanketed fuel region. Modelling these boundaries exactly increases calculational accuracy in this region of the core since cross sections of two different enrichments are not averaged. In addition, an increase in axial resolution results from decreasing the axial node size from 12 to 6 inches, which is also a desired effect.

- The anticipated future transition to the Westinghouse IFBA fuel product will require the use of a 24 axial level nodal model.

The results of the statistical analysis and the calculation of Observed Nuclear Reliability Factors (ONRFs) and uncertainty factors are discussed in the remainder of this Section. The December 12, 1995 letter requesting NRC review, the NRC's request for additional information, including Duke Power Company's responses, and the NRC SER are included in this Section.

2.0 Benchmark Cycles

Characteristics of the fuel cycles analyzed to develop 24 level nuclear reliability factors are summarized in Table 1. These fuel cycles are expected to be representative of future generation core designs.

2.0 Benchmark Cycles Continued

Table 1
Fuel Cycle Characteristics

Cycle	No. of Feed and Enrichment	Design EFPD	Cycle Characteristics
M1C09	64 feed at 3.45 w/o	340	Low number of feed and BP loading
M2C09	76 feed at 3.65 w/o	395	High Enr. and BP loading
C1C07	72 feed at 3.45 w/o	350	Typical
C2C06	76 feed at 3.75 w/o	380	First Transition cycle to Mk-BW
C2C07	40 feed at 4.0 w/o + 8 feed at 3.60 w/o+ 40 feed at 3.50 w/o	430	Axial Blankets and High No. of feed

+ Axial Blanket fuel with 6 inch natural uranium blankets

3.0 Uncertainty Analysis (ONRF)

The statistical analysis performed to develop Observed Nuclear Reliability Factors (ONRF's) for FΔH, Fq and Fz was based on the analysis of 75 measured power distributions. SIMULATE-3 predicted data was generated at the explicit conditions (the core power level, rod position and burnup) at which the measurement was performed. Depletion history effects were modelled by averaging the power level and rod positions between measurements. A quarter core model was used for all predictions.

Measured power distributions were obtained by processing measured reaction rate data using the computer code DETECTOR (formerly SNA-CORE). The methodology used to expand instrumented location data to un-instrumented locations is described in Section 3.4 and in the response to Question 9 in revision 0 to DPC-NE-1004, and also in reference 1.

The processing of measured reaction rates to power requires the use of predicted data. The development of this data, which consists of predicted powers and detector reaction rates (theoretical factors) for each instrumented core location, is identical to that described in Section 3.4 (rev. 0) of this report with the exception that powers and reaction rates from SIMULATE-3 are collapsed for discrete axial segments of the reactor core (3-D theoretical factors) instead of averaged over the entire core height (2-D theoretical factors). The motivation for going to multiple axial segments was to capture the axial dependency of the power to reaction rate ratio. This is especially important for fuel designs which are not axial homogeneous such as axial blanket fuel and Westinghouse's Integral Fuel Rod Burnable Absorber (IFBA) fuel designs. The

3.0 Uncertainty Analysis (ONRF) Continued

methodology employed in SIMULATE-3 to calculate 3-D theoretical factors is identical to that used to calculate 2-D theoretical factors. The only difference is that the reactor core is divided into axial regions, where each axial region is represented by a theoretical factors set. The application of the 3-D theoretical factor sets to process measured reaction rates is unchanged from the application of 2-D theoretical factors sets.

The statistical analysis performed to develop uncertainty factors is based on comparison of predicted to measured assembly data for FΔH, Fq and Fz. Only predicted and measured data pairs with values greater than 1.0 are included in the statistical data base. The uncertainties developed are based on normal distribution theory with a 95% probability and 95% confidence level. The results of the normality test are shown in Table 2. Table 3 contains the FΔH, Fq and Fz ONRFs for the 24 axial level model, along with previously calculated 12 axial level ONRF's.

Table 2
Westinghouse 24 Axial Level
Normality Test Results

ONRF Type	N	D'(P=.025)	D'	D'(P=.975)	Remarks	k
FΔH	2516	35448	34858	35743	nearly normal	1.6973
Fq	3162	49,966	49,719	50,338	nearly normal	1.6911
Fz	4200	76,530	76,284	77,024	nearly normal	1.6854

Table 3
Westinghouse 24 Axial Level
ONRF Results

ONRF Type	N	$\overline{M}$	$\overline{D}$	k	S(D)	24 Level ONRF	12 Level ONRF
FΔH	2516	1.168	0.0034	1.6973	0.016	1.020	1.017
Fq	3162	1.281	-0.0055	1.6911	0.025	1.037	1.057
Fz	4200	1.138	-0.0103	1.6854	0.015	1.031	1.053

The derivation of the statistical model used to develop ONRF's is described in Section 4.2(b) in revision 0 of this report, and in more detail in Supplement 2, Section 5.1 of reference 2.

4.0 Uncertainty Factors for Safety Related Analyses

The uncertainty factors used in safety related analyses include the bias between predicted and measured assembly powers, and assembly and local pin uncertainties between predicted and measured powers. The assembly and local uncertainties are combined using square root sum of the squares.

The equations used to develop FΔH, Fq and Fz uncertainties are defined in Section 5 in revision 0 of this report and are included below for completeness.

$$UCR(F\Delta H) = 1 + Bias + \sqrt{U_{A-R}^2 + U_{R-L}^2}$$

$$UCT(Fq) = 1 + Bias + \sqrt{U_{A-T}^2 + U_{R-L}^2}$$

$$UCA(Fz) = 1 + Bias + U_{A-A}$$

$$BIAS = -\overline{D} / \overline{M}$$

$$\text{Assembly Uncertainty} = (k \times S(D)) / \overline{M}$$

Where,

$\overline{D} =$ Mean of the difference between calculated and measured values (C - M)

$\overline{M} =$ Mean of the measured values

$k =$ One-sided 95/95 tolerance factor

$UCR\ (F\Delta H) =$ Radial uncertainty factor - FΔH

$UCT\ (Fq) =$ Total uncertainty factor - Fq

$UCA(Fz) =$ Axial uncertainty factor - Fz

$Bias =$ Bias between measured and predicted assembly data

$S(D) =$ Standard deviation of the difference between calculated and measured

$U_{A-R} =$ Radial (FΔH) Uncertainty (i.e. ONRF without bias term)

$U_{R-L} =$ Local Pin Uncertainty = 0.02

$U_{A-T} =$ Total (Fq) Uncertainty (i.e. ONRF without bias term)

$U_{A-A} =$ Axial Peak (Fz) Uncertainty (i.e. ONRF without bias term)

4.0 Uncertainty Factors for Safety Related Analyses

The 24 axial level uncertainty factors which can be used in Westinghouse reactor core safety related analyses are shown in Table 4 and do not include the engineering hot channel factor. The engineering hot channel factor may be statistically combined and applied as appropriate per references 3 and 4. The local pin peaking uncertainty was developed in revision 0 of DPC-NE-1004A and has a value of 2.0%.

Table 4
24 Axial Level Uncertainties
Without Engineering Hot Channel Factor

Parameter	Bias	Assembly Uncertainty	Pin Uncertainty	Total Uncertainties
FΔH	-0.0029	0.0234	0.02	1.0279
Fq	0.0043	0.0329	0.02	1.0428
Fz	0.0091	0.0216	----	1.0307

5.0 Conclusion

The ONRF's and uncertainty factors developed in Tables 3 and 4 are applicable for use in Westinghouse reactor core safety related analyses performed using a 24 axial level SIMULATE-3 model.

6.0 References

1. Duke Power Company McGuire Nuclear Station, Catawba Nuclear Station Nuclear Design Methodology for Reload Design", DPC-NE-2010PA, June 1985.

2. "Duke Power Company Oconee Nuclear Station Reload Design Methodology", NFS-1001A, April 1984.

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

4. "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P, April 1988.

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - NUCLEAR DESIGN METHODOLOGY
USING CASMO-3/SIMULATE-3P, DPC-NE-1004A, CATAWBA NUCLEAR STATION,
UNITS 1 AND 2 (TAC NOS. M94403 AND M94404)

Dear Mr. Tuckman:

On December 12, 1995, you submitted a proposed revision to the approved
Topical Report, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P,"
DPC-NE-1004A, November 1992. Your letter indicated that the number of axial
nodes in SIMULATE-3P had been increased and that your analysis supported a
proposed change in the observed nuclear reliability (peaking uncertainty)
factors.

The information provided in your letter is insufficient for us to determine
the acceptability of the proposed changes. We have identified a need for
additional information as set forth in the Enclosure.

This requirement affects nine or fewer respondents and, therefore, it is not
subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: Request For Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

NUCLEAR DESIGN METHODOLOGY, DPE-NE-1004A

1. What is the reason for the increase from 12 to 24 axial nodes? DPC's letter of December 12, 1995, mentions several factors potentially involved with the change but the reason for the change is not explicitly discussed.

2. Discuss all of the inputs involved in the changes shown in the Observed Nuclear Reliability Factors (ONRF). Are the changes the result of an analysis of actual fuel cycle data versus previously assumed values? If so, which is more conservative? Or, are the changes the result of the axial noding change?

3. Please identify which transients and accidents are affected by each change. Provide supporting analysis results for the limiting transients and accidents demonstrating how their acceptance criteria (DNB, kw/ft, pressure, etc.) are met.

Response to NRC Questions

1. What is the reason for the increase from 12 to 24 axial nodes? DPC's letter of December 12, 1995 mentions several factors potentially involved with the change, but the reason for the change is not explicitly discussed.

    Answer: The choice of a 12 axial node SIMULATE-3 model was originally chosen as a compromise between the desire for more axial nodes and that of computer run time. Since initially submitting DPC-NE-1004, the increase in computational efficiency has allowed Duke to increase the number of nodes without a substantial increase in computer cost. Additional reasons for wanting to increase the number of axial nodes in SIMULATE-3 from 12 to 24 are discussed below.

    a. SIMULATE-3 has a coding limitation on the number of axial regions which can be modelled within one node. For axial blanket fuel with burnable poisons rods, the nodal length must be decreased to less than 12 inches in order to satisfy this criteria. Axial node segments of 6 or 8 inches satisfy this requirement. A 6 inch axial node segment was chosen because this causes the nodal boundaries to match up exactly with the transition from the blanketed fuel region to the non-blanketed fuel region. Modelling these boundaries exactly increases calculational accuracy in this region of the core since cross sections of two different enrichments are not averaged.

    b. To better account for the axial dependence in predicted and measured power distributions of current generation vendor fuel designs which are not axially homogeneous (eg. axial blanket fuel).

2. Discuss all of the input involved in the changes shown in the Observed Nuclear Reliability Factors (ONRF). Are the changes the result of an analysis of actual fuel cycle data versus previously assumed values? If so, which is more conservative? Or, are the changes the result of the axial noding change.

Answer: The ONRF's for both 12 and 24 axial level SIMULATE-3 models are the result of analysis of actual fuel cycles. The changes in the calculated ONRF's are the result of using power distributions from more recent fuel cycles, increasing the number of axial nodes from 12 to 24 and accounting for the axial dependency of the power to reaction rate ratio in the measured power distribution. The original ONRF's for Westinghouse plants were developed in DPC-NE-1004 based on the analysis of the McGuire 2 Cycle 4, McGuire 2 Cycle 5, Catawba 1 Cycle 3 and Catawba 2 Cycle 2 core designs. The 24 axial level ONRF's were developed based on current generation core designs in order to reflect the more aggressive reload design strategies reflected in current core designs. Specifically, the power distribution database used to develop the 24 axial level ONRF's included the effects of higher enriched fuel, longer cycle lengths, higher burnup, higher BPRA loadings, and axial blankets. Characteristics of the fuel cycles analyzed are shown in Table 1.

Table 1
Fuel Cycle Characteristics

Cycle	No. of Feed and Enrichment	Design EFPD	Cycle Characteristics
M1C09	64 feed at 3.45 w/o	340	Low number of feed and BP loading
M2C09	76 feed at 3.65 w/o	395	High Enr. and BP loading
C1C07	72 feed at 3.45 w/o	350	Typical
C2C06	76 feed at 3.75 w/o	380	First Transition cycle to Mk-BW
C2C07	40 feed at 4.0 w/o + 8 feed at 3.60 w/o+ 40 feed at 3.50 w/o	430	Axial Blankets and High No. of feed

+ Axial Blanket fuel with 6 inch natural uranium blankets

The 24 axial level Fq and Fz ONRF's decreased by 2.0 and 2.2%, respectively, relative to DPC-NE-1004 values. However, the 24 axial level model ONRF for FΔH increased by 0.3% over the DPC-NE-1004 value. This increase is considered statistically insignificant and is attributed to the selection of more challenging, and radially heterogeneous core designs for the creation of the 24 axial level statistical data base relative to the core designs which were available when the DPC-NE-1004 FΔH ONRF was developed. The significance of the 0.3% increase in the FΔH ONRF is addressed in the answer to question 3.

Answer to Question 2 (Continued)

The decrease in the Fq and Fz uncertainties is attributed to the increase in axial resolution of the core model resulting from the axial node size reduction and from the use of axially dependent power to reaction rate ratios (which better characterize the spectral dependency and axial geometry of the fuel) to process measured reaction rates. These factors result in the reduction in the bias term included in the ONRF derivation. Note also that the variability of the statistical population as measured by the standard deviation of the Fq and Fz populations remain similar to previously calculated values. Therefore, the reduction in the Fq and Fz uncertainty factors is almost entirely due to the reduction in the predicted to measured bias.

The statistical data used to develop the 12 and 24 axial level ONRF's is provided in Table 2. The equations used to develop the ONRF's are contained in Section 4.2 of DPC-NE-1004.

<div align="center">

Table 2
12 and 24 Axial Level
Observed Nuclear Reliability Factors

</div>

12 Axial Level ONRF's:

Parameter	N	$\overline{M}$	$\overline{D}$	k	S(D)	ONRF
FΔH	1455	1.145	0.000	1.713	0.011	1.017
Fq	1998	1.257	-0.027	1.703	0.026	1.057
Fz	2520	1.148	-0.027	1.697	0.020	1.053

24 Axial Level ONRF's:

Parameter	N	$\overline{M}$	$\overline{D}$	k	S(D)	ONRF
FΔH	2516	1.168	0.0034	1.6973	0.016	1.020
Fq	3162	1.281	-0.0055	1.6911	0.025	1.037
Fz	4200	1.138	-0.0103	1.6854	0.015	1.031

3. Please identify which transients and accidents are affected by each change. Provide supporting analysis results for the limiting transients and accidents demonstrating how their acceptance criteria (DNB, kw/ft, pressure, etc.) are met.

Answer: Future safety analyses will use FΔH, Fq and Fz uncertainties with values greater than or equal to the values of the uncertainty factors shown in Table 3. The uncertainties calculated in Table 3 were developed using the same statistical data used in the development of ONRF's and are based on the FΔH, Fq, and Fz uncertainty factor equations developed in Section 5.1 through 5.3 of DPC-NE-1004. The difference between the ONRF's shown in Table 2 and the uncertainties shown in Table 3 is the statistical combination of the pin uncertainty with the assembly uncertainty.

Table 3
DPC-NE-1004 and 24 Axial Level Uncertainties
Without Engineering Hot Channel Factor

Parameter	Bias	Assembly Uncertainty	Pin Uncertainty	Total Uncertainties	DPC-NE-1004 Uncertainties +
FΔH	-0.0029	0.0234	0.02	1.028	1.026
Fq	0.0043	0.0329	0.02	1.043	1.061
Fz	0.0091	0.0216	----	1.031	1.053

+ 12 axial level uncertainty

The Fq and Fz uncertainty factors used in current and previous accident analyses bound values calculated for the 24 axial level model. Therefore, there is no impact to past, present or future safety analyses in which only Fq and Fz uncertainties are used. The increase in the FΔH uncertainty over the topical value is of no safety concern for future analyses because a value greater than or equal to the 24 axial level FΔH uncertainty will be used in these safety analyses.

Since the increase in the FΔH uncertainty factor may be a result of the analysis of more complex reactor cores and not a result of transitioning to a 24 axial level model, the impact of this increase was assessed by performing a review of FSAR Chapter 15 accidents and their appropriate acceptance criteria which could be affected by an increase in radial (FΔH) uncertainty factor. The following calculations are affected:

- Pin Pressure
- Creep collapse
- DNB

The calculation of peak fuel enthalpy, Linear Heat Rate to Melt (LHRTM) kw/ft limits and primary and secondary peak pressures are not affected by the increase in radial uncertainty for the following reasons.

Answer to Question 3 Continued

Significant margin exists to the peak fuel enthalpy limit of 280 cal/gm, such that this parameter is not limiting. For the calculation of LHRTM kw/ft limits, a total (Fq) uncertainty and not a radial uncertainty is applied since this is local phenomenon. Primary and secondary system peak pressure response calculations are based on a balance between energy removed by the steam generators, and energy added from the reactor core. Since the pressure response is dependent on the rate of energy deposited from the reactor core, independent of the peaking within the core, local peaking uncertainties are not important. Therefore, the calculation of accident acceptance criteria and confirmation of limits for peak fuel enthalpy, LHRTM and primary and secondary side peak pressure are unaffected.

Peak pin pressure and creep collapse calculations assume a bounding radial uncertainty of 1.036. Since this uncertainty bounds the 24 axial level value of 1.028, it can be concluded that the current and past analyses are unaffected.

For the FSAR Chapter 15 accidents in which DNB is a concern, thermal analyses are performed to ensure that fuel clad integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit. Two types of DNBR analyses are performed. Thermal analyses which are based on the Statistical Core Design (SCD) methodology described in reference 1, and thermal analyses which are not based on this methodology (non-SCD DNBR analyses). The FSAR Chapter 15 accidents which are based on the SCD methodology are unaffected by the change in uncertainty factors. This is because radial and axial uncertainty factors of 1.04 and 1.053 are assumed in these accident analyses, which bound the 24 axial level uncertainty factors.

The FSAR Chapter 15 accidents which are based on the non-SCD methodology, along with the radial and axial uncertainty factors assumed in the analysis of each accident, are summarized in Table 4.

Table 4
Non-SCD DNB Uncertainty Factors

Accident	Radial Uncertainty	Axial Uncertainty
Startup of an Inactive RC Pump at an Incorrect Temp.	N/A	N/A
Steam Line Break	1.036	1.053
Locked Rotor	1.026	1.053
Rod Ejection	1.026	1.053

From the data in Table 4 it is not immediately evident that past DNBR analyses performed based on non-SCD methodology were conservative. Therefore, each of these analyses are discussed below.

Answer to Question 3 Continued

The startup of an inactive coolant pump at an incorrect temperature transient is a non-limiting transient which is bounded by the analysis of other FSAR transients. The FΔH uncertainty assumed in the steam line break accident bounds both the 12 and 24 axial level uncertainties. For the Locked Rotor and Rod Ejection accidents, the FΔH uncertainty assumed in the accident analyses does not bound the FΔH uncertainty calculated for the 24 axial level model. However, since DNB is a function of both the radial and axial power distribution, the decrease in the 24 level axial uncertainty factor more than offsets the slight increase in radial uncertainty, resulting in an increase in DNB margin. Therefore, it can be concluded that previous accident analyses performed are conservative and the consequences of FSAR accidents previously evaluated and the margin to safety as defined in Technical Specifications is not reduced for the Locked Rotor and Rod Ejection accidents. In addition, it should be noted that margin retained between the 95/95 correlation and design DNBR limit used in accident analyses, (which is retained to account for unanticipated non-conservatisms) could also have been used to account for the slight increase in radial uncertainty.

In summary, the 24 axial level calculational uncertainty factors will be used in all safety related analyses in which a 24 axial level SIMULATE-3 model will be used. The slight increase in the FΔH uncertainty factor relative to the topical value does not increase the consequences or reduce the margin to safety of accidents previously evaluated. This is because either the accident was non-limiting, the accident analysis employed conservative FΔH uncertainty factors, or the tradeoff between the decrease in Fz uncertainty more than compensated for the increase in FΔH uncertainty. The calculation of accident acceptance criteria for LHRTM, peak fuel enthalpy and peak pressure are also unaffected by the increase in radial uncertainty factor. Therefore, it can be concluded that the consequences of FSAR accidents previously evaluated and margin to safety as defined in the bases to Technical Specification is not decreased.

References

1. "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology", DPC-NE-2005P-A, February 1995.

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SUBJECT: SAFETY EVALUATION - NUCLEAR DESIGN METHODOLOGY USING CASMO-
3/SIMULATE-3P, DPC-NE-1004A, CATAWBA NUCLEAR STATION, UNITS 1 AND 2
MCGUIRE NUCLEAR STATION, UNITS 1 AND 2,(TAC NOS. M94403 AND M94404)

Dear Mr. Tuckman:

On December 12, 1995, and as supplemented on April 9, 1996, you submitted a
proposed revision to the approved Topical Report, "Nuclear Design Methodology
Using CASMO-3/SIMULATE-3P," DPC-NE-1004A, November 1992. Specifically, Duke
Power Company requested review of an effort to improve its reload design
methods. The CASMO-3/SIMULATE-3P power distribution uncertainty factors have
been re-evaluated using measured data from recent Catawba and McGuire fuel
cycles. The revision includes increasing the axial nodes in SIMULATE-3P from
12 axial nodes to 24 axial nodes. This change in the number of nodes
structure will allow explicit modeling of axial blanket fuel segments.

We have completed our evaluation and find this change to be acceptable. Our
evaluation is provided in the enclosure.

Sincerely,

Robert Martin

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Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414
            50-369 and 50-370

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MINOR REVISION TO TOPICAL REPORT DPC-NE-1004A

DUKE POWER COMPANY

DOCKET NOS. 50-413, 50-414,50-369 and 50-370

1.0 INTRODUCTION

By letter dated December 12, 1995, as supplemented on April 9, 1996,
(Reference 1), Duke Power Company, (DPC or licensee), requested review of a
revision to topical report DPC-NE-1004A, (Reference 2). Specifically, DPC
requested review of an effort to improve its reload design methods. The
CASMO-3/SIMULATE-3P power distribution uncertainty factors have been re-
evaluated by DPC using measured data from recent Catawba and McGuire fuel
cycles. The revision includes increasing the axial nodes in SIMULATE-3P from
12 axial nodes to 24 axial nodes. This change in the number of nodes
structure will allow explicit modeling of axial blanket fuel segments.

Benchmarking calculations were performed and presented in tabular form in the
DPC submittal.

2.0 EVALUATION

The results (Observed Nuclear Reliability Factors, ONRFs) of the calculations
for the assembly radial, axial, and total peaking factors (using 24 nodes),
were compared to those results obtained in the NRC approved topical,
DPC-NE-1004A. The comparison showed that the axial and total peaking
uncertainty factors decreased for the new (24 nodes) benchmarking and the
assembly radial power peaking increased slightly from 1.017 to 1.020. The
statistical analysis used in this analysis is the same as that used in the
approved topical, DPC-NE-1004A. The benchmarking performed by the licensee
included recent cycle data such as, longer cycle length, higher fuel
enrichment, and consequences of additional burnable poisons. The cores used
in the benchmarking were McGuire 1, McGuire 2, cycle 9, Catawba 1 cycle 7, and
Catawba 2, cycles 6 and 7.

2.1 A Change in the Number of Nodes: from 12 Nodes to 24 Nodes.

The increase from 12 nodes to 24 nodes will remove the coding limitation
(number of axial regions which can be modeled in one node) in SIMULATE-3P. In
order to take advantage of axial blanket fuel with burnable poisons rod, a
6 inch axial node was chosen, because nodal boundaries match exactly with the
transition from the blanketed fuel region to the non-blanketed fuel region.
Boundary matching enhances calculational accuracies and simplifies cross-
section assignments. More importantly, the increase in the number of nodes
will provide a more accurate prediction of measured power distributions of
current generation vendor fuel designs which are typically not axially
homogeneous.

The original ONRFs for the Westinghouse plants were developed in the approved
topical DPC-NE-1004 based on the analysis of the McGuire 2 cycles 4 and 5,

Catawba 1, cycle 3 and Catawba 2 cycle 2 core designs. The 24 nodes ONRFs were developed based on current core designs, reflecting the more aggressive reload design strategies reflected in current core designs. Included in current core design databases is such information as higher enrichment, higher burnup, longer cycle lengths, and axial loading.

The analysis conducted by the licensee indicated that the F_q and F_z ONRF's decreased by 2.0 and 2.2%, respectively, relative to ONRF's stated values in DPC-NE-1004A. The minor statistical increase in the radial axial peaking factor, F_{AH}, is due mainly to the increase in the number of nodes which contributes to a more radially heterogenous core design.

The decrease in the F_q and F_z uncertainties is due to the increase in the accuracy of presentation of the axial spectral dependency, as a result of the reduction of axial node size.

These reductions (decreases in the uncertainties) result in a reduction in the bias term included in the ONRF.

The licensee analyzed the impact of the increase in the F_{AH} uncertainty and found that the increase in the F_{AH} was mainly due to the increase in the complexity of the reactor cores and very little of it was due to the increase in the number of nodes. Consequently, the licensee considered the impact of this increase on the FSAR Chapter 15 accidents. The Chapter 15 calculations that were effected are pin pressure, creep collapse, and DNB. Peak fuel enthalpy, linear heat rate to melt limits, and primary and secondary peak pressures are not affected by the increase in radial uncertainty factors, because the licensee showed that in each case, either significant margin exists, (and thus the increase in the radial uncertainty factor is not limiting), or the actual pin peak pressure and creep calculations assume a bounding uncertainty value higher than that for the 24 axial node value.

The licensee conducted Chapter 15 DNB analysis to ensure that fuel integrity is maintained and that the minimum DNBR remains above the 95/95 DNBR limit. They conducted two kinds of DNBR analyses: 1) A thermal analysis which is based on the Statistical Core Design (SCD) methodology described in (Reference 3), and 2) a thermal analysis which is not based on an SCD method.

For the first method, the pertinent Chapter 15 accidents were not affected because bounding radial and axial uncertainty factors which are greater than those of the 24 node analysis are assumed in the final analyses. For the second method, Chapter 15 analyses including Startup of Inactive RC pump, Steam Line Break, Locked Rotor, and Rod Ejection, were analyzed separately.

For the startup of the inactive coolant pump, and of the steam line break, the F_{AH} assumed bounds both the 12 and 24 axial level uncertainties. For the rod ejection accident, analysis showed that the present F_{AH} uncertainty did not bound the F_{AH} uncertainty calculated for the 24 axial level model. However, analyses conducted by the licensee showed that, since DNB is a function of both the radial and axial power distributions, the decrease in the 24 axial level uncertainty factor more than offsets the slight increase in the radial uncertainty, resulting in a net increase in the DNB margin. Consequently, it can be concluded that analyses regarding Locked Rotor and Steam Line Break are conservative, and that the margin of safety as defined in the Technical

Specifications are maintained. The staff agrees with these conclusions.

3.0 CONCLUSION

The NRC staff has reviewed the revision, dated December 12, 1995 and April 9, 1996, to Topical Report DPC-NE-1004A, submitted by the Licensee for the operation of the Catawba and McGuire Nuclear Stations. Based on this review, the staff concludes that the requested minor change to the above mentioned topical is acceptable.

Principal Contrubutor: A. Attard

4.0 REFERENCES

1. Duke Power Company Letter, M. S. Tuckman, dated December 12, 1995, to U.S. Nuclear Regulatory Commission, Document Control Desk, Washington DC 20555.

2. Duke Power Company Letter, M. S. Tuckman, Dated April 9, 1996, Responses to Request for Additional Information.

3. DPC-NE-2005P-A, February 1995, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology."

Section E

Duke Power Company 10 CFR50.59 Evaluations

1.0 Introduction

This Section contains 10 CFR50.59 Evaluation Summaries which are applicable to the CASMO-3/SIMULATE-3P Methodology described in DPC-NE-1004A (ref. 1). The following 10 CFR50.59 evaluations have been performed.

a. SIMULATE-3 Version 4 50.59 Evaluation for Westinghouse Reactors (Ref. 5) - This evaluation evaluated minor changes to core modeling as described in DPC-NE-1004A. Specifically, this evaluation addressed increasing the number of axial levels from 12 to 18 levels.

b. SIMULATE-3 Version 4 10 CFR50.59 Evaluation for B&W Reactors (Ref. 6) - This evaluation evaluated minor changes to core modeling as described in DPC-NE-1004A. Specifically, this evaluation addressed increasing the number of axial levels from 12 to 23 and the transition to an enrichment dependent cross section library with shutdown cooling.

A more detailed discussion of each of the 10 CFR50.59 evaluations performed is provided below.

2.0 10CFR50.59 Evaluation Summaries

A. SIMULATE-3 Version 4 10 CFR50.59 Evaluation for Westinghouse Reactors

This 50.59 evaluated the impact of changing from SIMULATE-3 version 3 to version 4, including an axial level change from 12 to 18 levels. Note that the change in code versions as designated in the 10 CFR50.59 evaluation does not represent a fundamental change in the solution technique, or analytical models used within SIMULATE-3, but does represent code modifications performed to correct code errors, add core modeling options and enhance output file content. The change in axial nodalization was performed in order to more accurately model the axial direction of the reactor core and account for the introduction of axial blankets. The benchmark cycles which were used to develop Observed Nuclear Reliability Factors (ONRF's) and Uncertainty Factors in revision 0 of DPC-NE-1004A were repeated using SIMULATE-3 version 4. These calculations were performed in references 2 and 3 and included the effects SIMULATE-3's grid model. Comparisons between predicted and measured peaking factors, mainly, $F\Delta H$, F_q and F_z, were performed using measured data generated using SIMULATE-3 generated theoretical factors (power to reaction rate ratios).

The uncertainty analysis was performed in accordance with the methodology outlined in DPC-NE-1004A. All distributions were normal, or nearly normal, and resulted in the 18 axial level ONRF's and Uncertainty Factors shown on the next page.

18 Axial Level Uncertainty Factors*:

Parameter	N	$\bar{M}$	$\bar{D}$	k	S(D)	ONRF	Uncertainty Factor
FΔH	1399	1.1491	-0.0015	1.7150	0.0102	1.017	1.026
Fz	2464	1.1478	-0.0198	1.6928	0.0236	1.052	1.052
Fq	1973	1.2549	-0.0201	1.7035	0.0277	1.054	1.059

*   All data is from Reference 4

In conclusion, all ONRF's and Uncertainty Factors were calculated to be less that their corresponding values in reference 1, DPC-NE-1004A. Therefore, it was concluded that original ONRF's and Uncertainty Factors calculated in DPC-NE-1004A are applicable for use in safety related analyses for either a 12 or 18 level SIMULATE-3 model.

B. SIMULATE-3 Version 4 10 CFR50.59 Evaluation for B&W Reactors

This 10 CFR50.59 evaluation evaluated the impact of changing from SIMULATE-3 version 1 to version 4, an axial noding change from 12 to 23 levels and the introduction of an enrichment dependent cross section library with shutdown cooling. Note that the change in code versions as designated in the 10 CFR50.59 evaluation does not represent a fundamental change in the solution technique, or analytical models used within SIMULATE-3, but does represent code modifications performed to correct code errors, add core modeling options and enhance output file content. The increase from 12 to 23 axial levels was made to accurately account for the geometry of the axial blankets. The transition to an enrichment dependent cross section library with shutdown cooling was pursued to realistically model the decay of fission products during a shutdown. Benchmark calculations performed in reference 7 demonstrated that the use of the enrichment dependent cross section library produced equal or better results when calculating Zero Power Physic Test parameters (i.e. critical boron concentrations, temperature coefficients, rod worths, etc). In addition, power distribution benchmarks were performed in reference 8 to assess the impact on Observed Nuclear Reliability Factors (ONRF's) and Uncertainty Factors. These calculations compared predicted and measured peaking factors; FΔH, Fq and Fz. The uncertainty analysis performed in reference 8 was performed in accordance with the methodology outlined in DPC-NE-1004A. All distributions were normal, or nearly normal, and resulted in the 23 axial level ONRF's and Uncertainty Factors shown below.

23 Axial Level Uncertainty Factors*:

Parameter	N	$\bar{M}$	$\bar{D}$	k	S(D)	ONRF	Uncertainty Factor
FΔH	1108	1.206	0.001	1.724	0.019	1.026	1.033
Fz	1719	1.098	-0.001	1.708	0.019	1.030	1.037
Fq	1237	1.306	-0.007	1.720	0.030	1.045	1.050

*   All data is from Reference 8

In conclusion, all ONRF's and Uncertainty Factors were calculated to be less that their corresponding values in reference 1, DPC-NE-1004A. Therefore, it was concluded that original ONRF's and Uncertainty Factors calculated in DPC-NE-1004A are applicable for use in safety related analyses for either a 12 or 23 level SIMULATE-3 model using an enrichment dependent cross section library with shutdown cooling.

3.0 References

1. Duke Power Company, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P", DPC-NE-1004A, November 1992.

2. Calculation File, "SIMULATE-3P Benchmark Analysis for McGuire Units 1 and 2", Revision 2, MCC-1553.05-00-0051, July 1993.

3. Calculation File, "SIMULATE-3P Benchmark Analysis for Catawba Units 1 and 2", Revision 2, CNC-1553.05-00-0066, October 1993.

4. Calculation File, "SIMULATE-3P Uncertainty Factor Analysis for McGuire/Catawba", Revision 1, DPC-1553.05-00-0055, September 1993.

5. Calculation File, "SIMULATE Version 4 50.59 Evaluation", Revision 0, DPC-1553.05-00-0106, May 1994.

6. Calculation File, "Modifications to Oconee Reload Design Methodologies" (10 CFR50.59), Revision 0, OSC-5772, June 1994.

7. Calculation File, "Benchmark of SIMULATE Using Enrichment Dependent Cross Section Library', Revision 1, OSC-5759, May 1994.

8. Calculation File, "Generation of Observed Nuclear Reliability Factors for SIMULATE-3P", Revision 2, OSC-3761, August 1994.