ATTACHMENT 8a

Detailed Listing of Changes to DPC-NE-2011-P-A

Attachment 8a - Detailed Listing of Changes to DPC-NE-2011PA

This attachment provides a detailed list of proposed changes to the topical report DPC-NE-2011P. Changes are listed according to the location in DPC-NE-2011PA. Cited references are listed at the end of this attachment.

1. Cover, Revision History, Table of Contents, List of Figures

   Description: Editorial changes to correspond to changes made throughout this report.

2. Section 1.1, First Paragraph

   Description: Editorial change to add the acronym "RPS" for reactor protection system.

3. Section 1.1, Last Paragraph

   Description: Removed the word "current" in the first sentence and deleted the last sentence.

   Justification: The methodology of this report is consistent with current Technical Specifications (Reference 3).

4. Section 1.2, First Paragraph

   Description: Editorial change to add the acronyms "LOCA" and "LOFA".

5. Section 1.2, Second Paragraph

   Description: Editorial change adding the words "departure from nucleate boiling" to the acronym DNB.

6. Section 1.2, Third Paragraph

   Description: Updated to include a description of the SIMULATE-3P methodology.

   Justification: These changes clarify that NODE is augmented by RADLOC factors, make the description consistent with current NRC approved methods (Reference 1), and clarify the description of applying appropriate uncertainty factors.

7. Section 1.3, First Paragraph

   Description: Editorial change to add the word "level" in the last sentence.

8. Section 1.3, Second Paragraph

   Description: Changed the last sentence to include a reference to SIMULATE and clarified that non-NRC approved codes are post-processing codes.

   Justification: This change makes the description consistent with current NRC approved methods (Reference 1).

9. Section 1.4

   Description: For clarity, added the acronym "RADLOC" to the definition of "Radial Local Factors" and added definitions for the terms "QPTR", "MATP", and "MARP".

10. Figure 1

    Description: Editorial change to add CFM to the list of computed monitor factors. Added a flow diagram for a SIMULATE based maneuvering analysis to makes the report consistent with current NRC approved methods (Reference 1).

11. Section 2.1

    Description: Added a paragraph to describe the generation of power distribution data using SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

12. Section 2.3, Third Paragraph, Third sentence

    Description: Changed the sentence to read "The transient is initiated with some combination of instantaneous changes in power level, control rod positions, and soluble boron concentration." instead of "The transient is initiated with an instantaneous change in power level, control rod position and soluble boron concentration."

    Justification: For some xenon transient simulations, the power level can be held constant while the control rods are moved to induce a larger AFD to span the AFD range of interest. This change is made to avoid difficulties with the literal interpretation of the original description.

13. Section 2.4, Last Paragraph

    Description: Editorial change to remove the word "step" to clarify that discrete points in time are used.

14. Section 2.5

    Description: Clarified that the generation of radial local factors applies to NODE methods. Added a paragraph to describe the generation of power distribution data using SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

15. Tables 1 and 2

    Description: Changed the description of the transient conditions for modeling xenon transients and the description of control rod positions used to model transient power distributions to reflect current use.

    Justification: The indicated values are provided for illustration.

16. Section 3.1

    Description: Revised the description of power peaking uncertainties.

    Justification: The revised wording clarifies the original description and makes the discussion consistent with previously NRC approved methods (References 1 and 2). Also, specific values are removed to avoid confusion (the values are contained in References 1 and 2).

17. Section 3.3

    Description: The description of the F_Z ONRF calculation is moved to proposed Revision 1 of Reference 2.

    Justification: The calculation of the axial ONRF is more appropriately placed in DPC-NF-2010 (Reference 2). The application of this uncertainty is covered in the revised Section 3.1.

18. Section 4.1, Third Paragraph

    Description: Changed the first sentence to reflect that the calculated peak may be obtained directly from SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

19. Section 4.2

    Description: Added a footnote to the definitions of LHR and NP describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

    Justification: Three dimensional local peak pin values are calculated directly in SIMULATE.

20. Section 4.2

    Description: Clarified the development and application of UCT (see revised Section 3.1).

21. Section 4.2

    Description: Removed the numerical value of the power level uncertainty in the definition of FP and in the last paragraph.

    Justification: This value is an assumption in the UFSAR Chapter 15 Safety Analyses and is subject to change.

22. Section 4.3, First Paragraph

    Description: The description of the set of MATP curves is changed to indicate the limits are based on NRC approved thermal-hydraulic methods.

23. Section 4.3

    Description: Clarified the definitions for UCA and UCR.

24. Section 4.3

    Description: Added a footnote to the definitions of RPP and RNP describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

    Justification: Two dimensional peak pin values are calculated directly in SIMULATE.

25. Section 4.3

Description: Corrected the misspelling of the word "quadrant" in the definition of TILT.

26. Sections 4.4 and 4.5

Description: Clarified that the DNB calculations and CFM limits "validate" the RPS limits.

27. Section 4.5, First Paragraph

Description: Added the word "level" in the last sentence for clarity.

28. Section 4.5

Description: Added a footnote to the definitions of LHR and NP describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

Justification: Three dimensional local peak pin values are calculated directly in SIMULATE.

29. Section 4.5

Description: Changed the reference of "SC" to "UCT" in the definition of LHR, deleted the definitions of SC and RBOW, and updated the definition of UCT to clarify the development and application of UCT (see revised Section 3.1).

30. Section 4.5

Description: Removed the numerical value of the power level uncertainty in the definition of FP and in the last paragraph.

Justification: This value is an assumption in the UFSAR Chapter 15 Safety Analyses and is subject to change.

31. Section 4.5

Description: Deleted the last sentence of this section for clarity.

Justification: The application of uncertainties is covered in the revised Section 3.1.

32. Section 4.6, First Paragraph

Description: Clarified the description of the AFD – power level limits calculation in the next to last sentence to reflect that limits are set to preclude operation with negative peaking margin.

Justification: This change is consistent with the basis of the operating limit and avoids difficulties with the literal interpretation of the original description.

33. Section 4.7, First Paragraph

    Description: Removed the rod ejection analysis reference to avoid confusion.

    Justification: The reference was provided as a general reference, and the rod ejection accident methods are contained in other NRC approved topical reports (References 4, 6).

34. Section 6.

    Description: Moved general/redundant information from Sections 6.1 and 6.2 to this Section.

35. Section 6.1

    Description: Updated the LOCA F_Q Technical Specification limit equation and definition of terms to make the report consistent with the Technical Specifications (Reference 3).

36. Section 6.1

    Description: Clarified the definitions for UMT and MT to make the report consistent with the Technical Specifications (Reference 3).

    Justification: The value of the manufacturing tolerance factor is subject to change with the fuel design.

37. Section 6.1

    Description: Corrected the misspelling of the word "quadrant" in the definition of TILT.

38. Section 6.1

    Description: Added a footnote to the definition of $F_Q^D(x,y,z)$ describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

    Justification: Three dimensional local peak pin values are calculated directly in SIMULATE.

39. Section 6.1

    Description: Added "LOCA" in the definition of $M_Q(x,y,z)$ for clarification..

40. Section 6.1, Paragraph following the definition for $M_Q(x,y,z)$

    Description: Updated the description of $F_Q^D(x,y,z)$ to indicate that the design data is generated as close to operating conditions as possible to be consistent with the flux measurement.

41. Section 6.1, Paragraph following the definition for $M_Q(x,y,z)$

    Description: Editorial change to improve the clarity of the third sentence.

42. Section 6.1

    Description: Updated the equation for $M_Q(x,y,z)$ to make the report consistent with the Technical Specifications (Reference 3).

43. Section 6.1

    Description: Added a footnote to the definitions of $F_Q^T(x,y,z)$ describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

    Justification: Three dimensional local peak pin values are calculated directly in SIMULATE.

44. Section 6.1

    Description: Changed F_Q^{Max} to $F_Q^L(x,y,z)^{OP}$ to make the report consistent with the Technical Specification terminology (Reference 3).

45. Section 6.1

    Description: Removed the extra left parenthesis before $M_Q(x,y,z)$ in the equation for $F_Q^{Max}(x,y,z)$.

46. Section 6.1

    Description: For completeness and to make the report consistent with Technical Specifications (Reference 3), added a paragraph at the end of this section referencing the method for accounting for possible peaking increases over the 31 EFPD surveillance period.

47. New Section

    Description: For completeness and to make the report consistent with Technical Specifications (Reference 3), added a section (designated as Section 6.2) containing a description of the methods used to perform surveillance monitoring of the core against CFM limits.

48. Section 6.2 (Original section numbering is used),

    Description: Renumbered this section to Section 6.3, because of adding a section for CFM monitoring.

49. Section 6.2

    Description: Remove the first sentence, since it is covered in the introduction in the revised Section 6.

50. Section 6.2

    Description: Revised the first paragraph to include the Technical Specification $F_{\Delta H}$ limit, added a paragraph to describe the terms used in this limit, and updated the discussion of the maximum allowed peaks to make the report consistent with the Technical Specifications (Reference 3).

51. Section 6.2

    Description: Clarified the definition for UMR to make the report consistent with the Technical Specifications (Reference 3).

52. Section 6.2

    Description: Added a footnote to the definitions of $F^{D}_{\Delta H}(x,y)$ describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

    Justification: Two dimensional peak pin values are calculated directly in SIMULATE.

53. Section 6.2, Paragraph following the equation for $F^{D}_{\Delta H}(x,y)$

    Description: Clarified the first three sentences and the next to last sentence to reflect that power distributions within the operating limits are used to determine $M_{\Delta H}$

    Justification: Operating limits do not consist of only the AFD – power level limit, but also include the rod insertion limit. This change avoids difficulties with the literal interpretation of the original description. This wording change is also consistent with the revised description for $M_Q(x,y,z)$.

54. Section 6.2

    Description: Changed the equation for $M_{\Delta H}$ to include "MARP" instead of "MATP" to make the report consistent with the Technical Specifications (Reference 3).

55. Section 6.2

    Description: Added a footnote to the definitions of $F^{T}_{\Delta H}(x,y)$ describing that RADLOCs are not required with the use of SIMULATE to make the report consistent with current NRC approved methods (Reference 1).

    Justification: Two dimensional peak pin values are calculated directly in SIMULATE.

56. Section 6.2

    Description: Changed $F_{\Delta H}$ and $F^{Max}_{\Delta H}$ to $F^{L}_{\Delta H}(x,y,z)^{SURV}$ to make the report consistent with the Technical Specifications (Reference 3).

57. Section 6.2

    Description: For completeness and to make the report consistent with Technical Specifications (Reference 3), added a paragraph at the end of this section referencing the method for accounting for possible peaking increases over the 31 EFPD surveillance period.

58. Section 6.3 (Original section numbering is used),

    Description: Renumbered this section to Section 6.4, because of adding a section for CFM monitoring

59. Section 6.3,

    Description: Added subsection numbers for AFD – Power Level Limits, Control rod insertion limits, Heat flux hot channel factor – $F_Q(x,y,z)$, and Nuclear enthalpy rise hot channel factor – $F_{\Delta H}(x,y)$ for clarity.

60. Section 6.3, 'Heat flux hot channel factor' Section, First Paragraph

    Description: Changed the wording at the end of the first sentence to state that $F_Q^M(x,y,z)$ will always be within "applicable limits" instead of "limits specified by the LOCA analysis".

    Justification: This change makes the description of the requirements for $F_Q^M(x,y,z)$ consistent with the Technical Specifications (Reference 3).

61. Section 6.3, 'Heat flux hot channel factor' Section, First Paragraph

    Description: Editorial change to remove the $F_Q^M(x,y,z)$ LOCA limit equation and replace with a reference to Section 6.1. This equation is fully described in Section 6.1.

62. Section 6.3, 'Heat flux hot channel factor' Section, Third Paragraph

    Description: Changed the wording to state that $F_Q^M(x,y,z)$ "meets applicable limits for LOCA and CFM" instead of "is met at the extremes of the AFD – power level operating limits", removed the words "maneuvering analysis", and updated the nomenclature of the $F_Q^M(x,y,z)$ limit terms.

    Justification: These changes make the description of the requirements for $F_Q^M(x,y,z)$ consistent with the Technical Specifications (Reference 3).

63. Section 6.3, 'Heat flux hot channel factor' Section, Fourth Paragraph

    Description: Clarified the wording of the first three sentences to remove ambiguity and to make the report consistent with the Technical Specifications (Reference 3).

64. Section 6.3, 'Heat flux hot channel factor' Section, Fourth Paragraph

    Description: Split this paragraph into two paragraphs, and clarified the requirements of exceeding a monitoring limit for F_Q to make the report consistent with Technical Specifications (Reference 3).

65. Section 6.3, 'Nuclear enthalpy rise hot channel factor' Section, First Paragraph

    Description: Reworded the second sentence to state "maximum allowed values" instead of "Maximum Allowed Total Peak curves" to make the report consistent with the Technical Specifications (Reference 3).

66. Section 6.3, 'Nuclear enthalpy rise hot channel factor' Section, Second Paragraph

    Description: Changed the wording to state that $F_{\Delta H}^{M}(x,y)$ "meets applicable limits for LOFA" instead of "is met at the extremes of the AFD – power level operating limits", removed the words "maneuvering analysis", and updated the nomenclature of the $F_{\Delta H}^{M}(x,y)$ limit terms.

    Justification: These changes make the description of the requirements for $F_{\Delta H}^{M}(x,y)$ consistent with the Technical Specifications (Reference 3).

67. Section 6.3, 'Nuclear enthalpy rise hot channel factor' Section, Last Paragraph

    Description: Clarified the second sentence to make the report consistent with Technical Specifications (Reference 3).

68. Section 6.3, 'Quadrant power tilt' Section

    Description: Clarified the maximum limit term for F_Q CFM and other terms to make the report consistent with Technical Specifications (Reference 3).

69. Section 7.0,

    Description: Added References 15 and 16, and updated Reference 14.

    Justification: These additional references make the description consistent with current NRC approved methods (References 1, 4, and 5).

70. Appendix A, MARGINS

    Description: Changed the second sentence to state "MARGINS requires three dimensional power distribution data for input" instead of "MARGINS requires the radial local factors from PDQEDIT and the three dimensional nodal power distributions from NODE for input".

    Justification: This change clarifies the required power distribution input format.

71. Appendix A

    Description: For completeness and to make the report consistent with NRC approved methods (Reference 1), added discussion for the computer code SIMULTE-3P.

72. New Appendix B

    Description: Placed all NRC requests for additional information and DPC responses in a new appendix.

73. New Appendix C

    Description: Placed a copy of the NRC SER giving approval of the methods DPC-NE-2011P, Revision 0 in a new appendix.

References:

1. "Duke Power Company, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P", DPC-NE-1004A, Revision 1, SER dated April 26, 1996.
2. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Nuclear Physics Methodology for Reload Design", DPC-NF-2010A, June 1985.
3. Technical Specifications for McGuire Nuclear Station Units Nos. 1 and 2 (Docket Nos. 50-369/370). Technical Specifications for Catawba Nuclear Station Units Nos. 1 and 2 (Docket Nos. 50-413/414).
4. "Duke Power Company Westinghouse Fuel Transition Report", DPC-NE-2009P-A, December 1999.
5. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Core Thermal-Hydraulic Methodology using VIPRE-01", DPC-NE-2004P-A, Revision 1, SER dated February 20, 1997 (DPC Proprietary).
6. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology", DPC-NE-3001PA, November 1991 (DPC Proprietary).

ATTACHMENT 8c

DPC-NE-2011, Revision 1

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ATTACHMENT 8c

DPC-NE-2011, Revision 1

DUKE POWER COMPANY

Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors

DPC-NE-2011-NP
Revision 1

August 2001

Nuclear Generation Department
Nuclear Engineering

NON-PROPRIETARY

Revision History

Revision	Description
DPC-NE-2011P, Original Issue	Originally submitted to the NRC for approval in April 1988. An additional submittal was made to the NRC supplying reponses to a request for additional information.
DPC-NE-2011PA, Original Issue	NRC approved version issued in January 1990.
DPC-NE-2011P, Revision 1	Submitted to the NRC for approval in August 2001. This revsion updates the report for completeness (1) to indicate the use of methods approved by the NRC subsequent to the implementation of the original issue and (2) to expand the description of Fq monitoring to include the centerline fuel melt criterion. This revison also reflects changes in the descriptions of the xenon transient conditions. Finally, various editorial changes are made, including updating the Table of Contents and revising the page numbering format. Changes associated with this revision are denoted by revision bars, except for format changes.

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1. INTRODUCTION

1.1. Purpose

This report describes the methodology for performing a maneuvering analysis for four-loop, 193 fuel assembly Westinghouse reactors, such as McGuire and Catawba Nuclear Stations. Duke Power Company has developed this methodology as an alternative to the existing Relaxed Axial Offset Control (RAOC) Methodology (1). This maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of Duke Power Company's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower ΔT (OPΔT) and overtemperature ΔT (OTΔT) reactor protection system (RPS) trip functions.

Specifically, these limits are the axial flux difference (AFD) - power level operating space, the rod insertion limits and the $f(\Delta I)$ function of either the OPΔT or the OTΔT trip functions of the RPS.

These limits are monitored via Technical Specifications.

1.2. Summary of the Methods

The operating limits define the AFD - power level space and rod insertion limits which provide assurance that the peak local power in the core is not greater than that assumed in the analysis of design basis accidents or transients (loss of coolant accident (LOCA) or loss of flow accident(LOFA)). Operating the reactor within the allowed AFD - power level window and rod insertion limits satisfies the power peaking assumptions of the LOCA and LOFA analyses.

The RPS limits, among other functions, provide protection against fuel failure due to fuel melting (CFM) or departure from nucleate boiling (DNB) during anticipated transients. The relevant limits are set such that the RPS will trip the reactor before fuel damage occurs.

The maneuvering analysis uses a three dimensional nodal reactor model to calculate a set of power distributions at several points in core life. These power distributions are based on a set of abnormal xenon distributions to insure predicted power distributions are conservative with respect to those expected to occur. In the EPRI-NODE-P (NODE) model, the three dimensional nodal power distribution is augmented by pin to assembly factors for the maximum pin power in each assembly. These pin to assembly factors are derived from a two dimensional fine mesh (pin by pin) model of the core. In the SIMULATE-3P (SIMULATE) model, the three dimensional local peak pin power distributions are explicitly calculated. Appropriate uncertainty factors are applied to the calculated power distributions which are then evaluated against the various thermal limits. The operating limits and the f(ΔI) function of either the OPΔT or the OTΔT RPS trip functions are then set to exclude the power distributions that exceed the respective thermal limits. Figures 1A and 1B show representative flow charts of the data as it goes through a NODE and a SIMULATE based maneuvering analysis.

1.3. Applicability of the Method

The maneuvering analysis presented in this report applies to Westinghouse four loop, 193 assembly reactors. This method is intended to be used to set or validate the AFD - power level operating limits, the control rod insertion limits, and the RPS trip limits.

A system of computer programs is used to implement this method. A description of the computer programs currently in use is contained in Appendix A. This list includes both the major design codes approved by the NRC (4, 15) and minor codes that are used for post-processing data.

1.4. Definition of Terms

AFD

Axial Flux Difference is the percent power in the top of the core minus the percent power in the bottom of the core.

Radial Local Factors

A Radial Local Factor (RADLOC) is the peak rod power in an assembly divided by the average rod power in the same assembly.

F_Q

F_Q is the local heat flux on a fuel rod surface divided by the core average fuel rod heat flux.

$F_{\Delta H}$

$F_{\Delta H}$ is the integral of linear power along a particular fuel rod divided by the average integral of all of the fuel rods.

QPTR

Quadrant Power Tilt Ratio is the normalized radial power distribution in each quadrant of the core as measured by excore nuclear detectors.

MATP

Maximum Allowed Total Peak values derived from core thermal-hydraulic analysis.

MARP

Maximum Allowed Radial Peak values derived from MATP values by dividing the MATP by the axial peak.

Figure 1A
Flow of Data Through a Maneuvering Analysis - EPRI-NODE-P

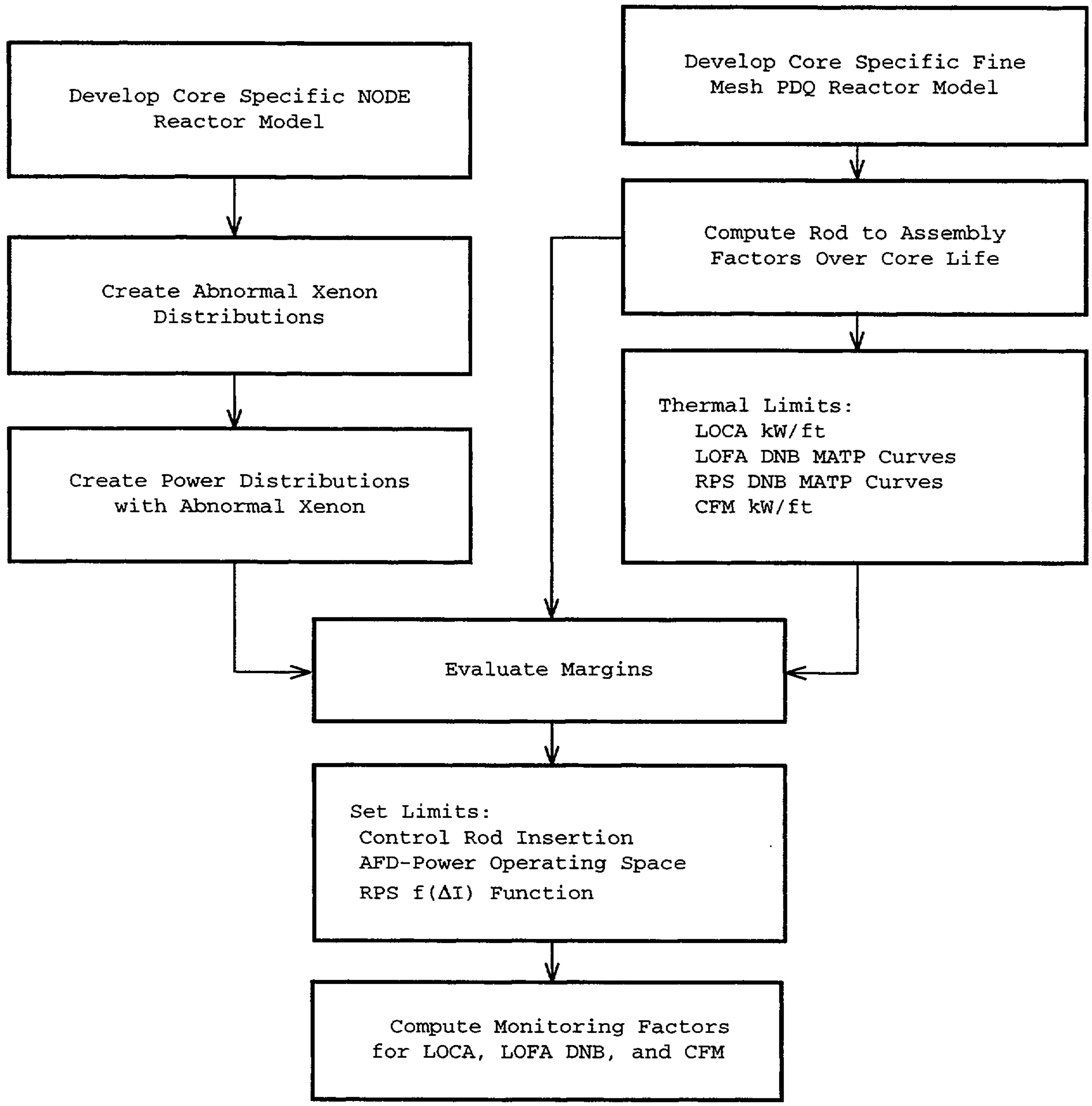
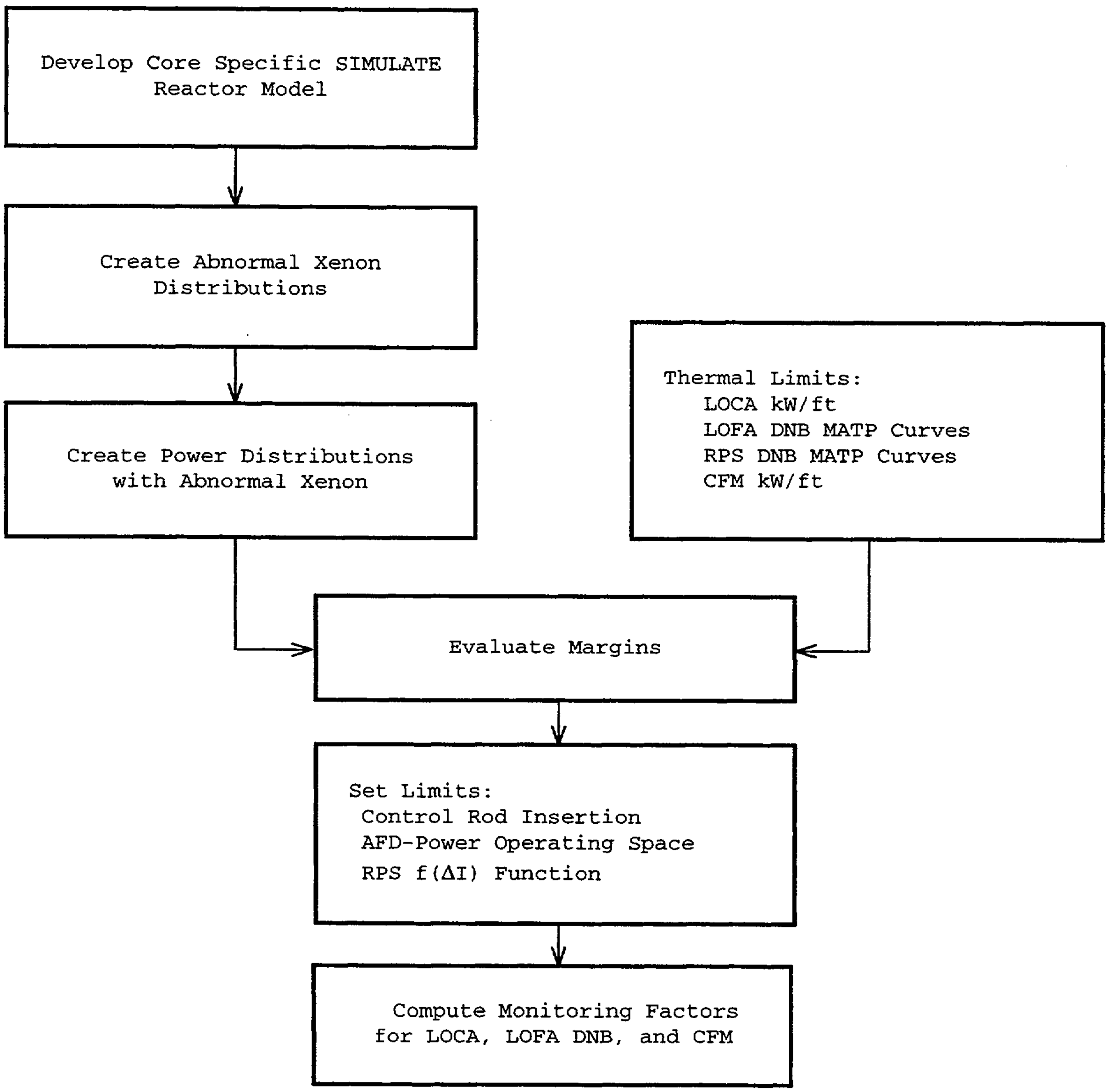


Figure 1B
Flow of Data Through a Maneuvering Analysis - SIMULATE-3P



2. GENERATION OF POWER DISTRIBUTIONS

2.1. Description of the Models Used

The three dimensional nodal power and xenon distributions are generated by a DPC version of EPRI-NODE-P (NODE). NODE has an explicit xenon and iodine model that allows power and time dependent xenon transients. NODE has a closed channel thermal hydraulic feedback model to generate fuel and moderator temperature distributions that are used in the neutronics model. The neutronics model accounts for fuel and moderator temperature, coolant flow, soluble boron concentration, lumped burnable absorbers, control rods, fuel burnup, and xenon and iodine distributions. The NODE model was approved by the NRC for use in reload design in Reference 5.

The radial local factors are extracted from a quarter core, one pin per mesh PDQ07 model of the core. PDQ calculations are run in two dimensions (X-Y) with a two dimensional thermal hydraulic feedback model. The PDQ model was approved for use in reload core design in Reference 5.

SIMULATE-3P (SIMULATE) can be used to generate three dimensional local peak pin power distributions. The SIMULATE model was approved for use in reload core design analyses in Reference 15.

2.2. Times in Core Life

The maneuvering analysis is typically performed at three times in core life:
[

]

2.3. Generation of Abnormal Xenon Distributions

The abnormal xenon distributions are generated with a set of limiting xenon transients at each point in core life that is to be analyzed. [

]

Table 1 shows the initial and transient conditions of the reactor for each of the transients. [

]

To add to the conservatism, these transients are modeled conservatively in several respects: [

] Because of these factors, the xenon transients in the reactor model will be more severe than could be reasonably expected to occur.

Each of the xenon transients start with xenon in equilibrium with the core at the initial conditions. The initial conditions are different for each transient. [

]

The control rod positions for the xenon transients were chosen to be at or near the expected rod insertion limits. The final control rod insertion limits may be different from the positions used in the xenon transients and the analysis will still be valid. This is because the xenon transients are so severe that the maneuvering analysis results are not sensitive to the control rod motions that drive the xenon transients.

The xenon transients proceed until [ ] Depending on the transient power level, this usually takes about [ ] hours. Figures 2 through 5 show graphs of AFD, xenon offset, xenon concentration, and soluble boron concentration plotted against time for a typical set of beginning of cycle xenon transients.

2.4. Generation of Power Distributions

Using the abnormal xenon distributions from the xenon transients, three dimensional power distributions are generated so that the operating and the RPS limits can be determined. As shown on Table 2, power distributions are generated with

] The operating limits are pre-conditions that would prevent exceeding the peak local power in the core assumed in the loss of coolant accident (LOCA) analysis or the loss of flow accident (LOFA, or a primary coolant pump trip) analysis. Because this is the normal operating mode of the reactor, control rod motion will be constrained by the power dependent rod insertion limits. [

] Power distributions for the operating limits are generated with these abnormal xenon distributions with the reactor at nominal conditions.

The RPS limits protect the fuel against damage from DNB or fuel melting even if the reactor should go through any one of several anticipated transients: [

]

The limit of the control rod motion for [

]

During an [

]

The abnormal xenon distributions from the xenon transients are chosen so that [

] Table 2 shows the reactor conditions and range of control rod positions. Criticality in the reactor model is maintained by instantaneous changes in soluble boron concentrations.

2.5. Generation of Radial Local Factors

The radial local factor is the ratio of the maximum rod power in an assembly to the average rod power of the assembly. Radial local factors are assembly and burnup dependent.

In the NODE methodology, the radial local factors are extracted from a core specific fine mesh PDQ model that has been depleted over the life of the cycle. The assembly average burnup, used as the independent variable to interpolate the radial local factors, is also extracted from the PDQ model. The PDQ model has two neutron energy groups and one spatial mesh point per fuel pin. Cross sections are taken from the EPRI-CELL (6) system and the CASMO (7) system. The PDQ model is described more fully in Reference 4.

SIMULATE (15) directly calculates local peak pin power distributions.

Table 1

Typical Reactor Conditions During Xenon Transients

	Initial Conditions		Transient Conditions	
Transient Name	% Power	Control Rods	% Power	Control Rods

Table 2

Typical Power Levels and Control Rod Bank Positions

for Generating Power Distributions

Figure 2
Sample Xenon Transient at Beginning of Life
AFD vs Transient Time

Figure 3
Sample Xenon Transient at Beginning of Life
Xenon Concentration vs Transient Time

Figure 4
Sample Xenon Transient at Beginning of Life
Xenon Offset vs Transient Time

Figure 5
Sample Xenon Transient at Beginning of Life
Soluble Boron Concentration vs Transient Time

3. UNCERTAINTY FACTORS

3.1. Power Distribution

The power peaks calculated in the Maneuvering Analysis are adjusted to account for calculation uncertainty and other applicable factors that may affect the power peaking in the core.

References 4, 5, and 15 present calculation peaking uncertainties based on the benchmarking analysis of measured to predicted power distribution. The peaking uncertainty factor is calculated as described below.

$$\text{Peaking Uncertainty Factor} = 1 + BIAS + \sqrt{(UC^2 + Ux1^2 + Ux2^2 + \ldots)}$$

Where:

Peaking Uncertainty Factor - Defined as UCT, UCR, UCA in this report

UC - Calculation Uncertainty

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

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

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

UT - Total Peaking Uncertainty

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

UR - Assembly Radial Power Peaking Uncertainty

UA - Assembly Axial Power Peaking Uncertainty

Uxi - Additional Uncertainties, e.g. engineering HCF, rod bow, etc.

BIAS - Calculation Bias

When additional, independent, peaking augmentation factors (shown as Uxi above) such as the engineering hot channel factor and/or rod bow factor are required, the corresponding uncertainty values are statistically combined with the pin and assembly power calculation uncertainty values to obtain the total uncertainty factor. The application of specific parameters is discussed in Section 4.

3.2. <u>Quadrant Tilt</u>

The excore detector system is used to monitor gross changes in the core power distribution. The primary purpose of the excore detectors with respect to quadrant power tilts is to detect changes in tilt from the previous calibration. Since the Technical Specifications (2, 3) allow reactor operations with excore quadrant power tilts up to 2%, the relationship between excore quadrant power tilt and a penalty to apply to the thermal limits calculations had to be determined.

This relationship was determined by evaluating various tilt causing mechanisms for several reactor cores. This analysis was performed with full core NODE models. The results showed that a [ ] power peaking penalty is required to account for the allowed 2% excore quadrant power tilt. This penalty will be applied as TILT to the LOCA, DNB and centerline fuel melt margin calculations in Section 4.

4. LCO AND RPS LIMITS

4.1. General Methodology

The power distributions are divided into two categories for the thermal limits calculations. The operating limits use power distributions that were calculated with nominal inlet temperature, with control rod positions that bound expected insertion limits, and with power less than or equal to 100% power. Control rod positions will bound insertion limits in order to set the insertion limits. The RPS limits use power distributions with the power level up to and including 118% power, no administrative restriction on the control rod insertions and either nominal or low inlet temperature.

The margin to the various limits is calculated in the following fashion:

MARGIN % = (ALLOWED PEAK - CALCULATED PEAK)*100 / ALLOWED PEAK

The calculated peak is obtained directly from SIMULATE or is a synthesis of the three dimensional nodal power distribution from NODE and the radial local factors from the fine mesh two dimensional PDQ calculations. Depending on the limit type, this equation may be in terms of a peaking factor or a linear heat rate. Either the calculated peak or the allowed peak would contain sufficient factors to account for the various uncertainties and tolerances. AFD and control rod insertion limits for each limit type are set to exclude all power distributions with negative margins of the same limit type.

4.2. LOCA Margin Calculations

Since the LOCA limits are used to define the operating limits of the core, the operating limits power distributions, as described in Section 2.4, are used in this calculation. The LOCA margin is calculated for each node in the core, but only the most limiting value is used in the determination of the AFD - power level limits. The equations below show how the LOCA margin, LOCAM, is calculated.

LOCAM = Min {(LOCAMX(z) - LHR(x,y,z)) * 100 / LOCAMX(z)}

Where:

LOCAMX(z) = Axially dependent maximum allowable linear heat rate in kw/ft.

LHR(x,y,z)[1] = NP(x,y,z) * FP * AVGLHR * RADLOC(x,y,e) * UCT * TILT * RPF * AMF

NP(x,y,z)[1] = Nodal power from the power distribution calculation.

FP = Fraction of core power level, including power level uncertainty.

AVGLHR = Total core power divided by the total length of fuel rods in the core, kw/ft, accounting for fuel densification and thermal expansion.

RADLOC(x,y,e) = Burnup (e) dependent maximum rod assembly power factor.

UCT = Uncertainty factor on the pin total peak, including engineering hot channel factor and rod bow if not included in the LOCA analysis (see Section 3.1).

TILT = Factor to account for a peaking increase due to an allowed quadrant tilt (see Section 3.2).

RPF = Factor to account for the power deposited in the fuel rod.

AMF = Additional Margin Factor, optionally used to incorporate additional design margin.

The values for LOCAMX(z) are derived from the Technical Specification limits on F_Q. Typical limiting values are shown in Figure 6.

The uncertainty on power level and the factor to account for power deposited in the fuel will be used only if these factors were not accounted for in the limits on F_Q.

[1] For SIMULATE, LHR does not include the RADLOC factor, since NP is the SIMULATE three dimensional local peak pin value.

4.3. LOFA DNB Margin Calculations

The LOFA DNB limits are also used to define the operating limits, so the operating limits power distributions, as described in Section 2.4, are used in this calculation. The DNB margin calculation is based on a set of Maximum Allowed Total Peak (MATP) curves that are calculated with a NRC approved thermal-hydraulic method (e.g., Reference 14). The MATP curves are determined for several power levels (e.g., 100, 75 and 50% power). The input power distributions are selected to match the power level of each set of MATP curves. Sample MATP curves for LOFA DNB are shown in Figure 10. The DNB margin is computed for each assembly in the core, but only the minimum margin for each power distribution is used in the determination of the AFD - power limits. DNB margin, DNBM, is calculated as:

$$DNBM = Min\left\{\frac{MARP(x,y) - RPP(x,y)}{MARP(x,y)}\right\} * 100$$

Where:

MARP(x,y)	= MATP(z,AP(x,y))/(AP(x,y)*UCA)
AP(x,y)	= Axial peak in an assembly, on an assembly normalized basis.
UCA	= Assembly axial peak uncertainty factor (see Section 3.1).
MATP(z)	= Maximum allowed total peak, at the axial plane of the axial peak.
RPP(x,y)[2]	= RNP(x,y) * RADLOC(x,y,e) * AMF * TILT * UCR
RNP(x,y)[2]	= Normalized assembly power from the power distribution calculation.
RADLOC(x,y,e)	= Burnup (e) dependent maximum rod to assembly power factor.
UCR	= Uncertainty factor on the pin radial peak, including engineering hot channel factor and rod bow if not included in the DNB analysis (see Section 3.1).
AMF	= Additional margin Factor, optionally used to incorporate additional design margin.

[2] For SIMULATE, RPP does not include the RADLOC factor, since RNP is the SIMULATE two dimensional peak pin value.

TILT = Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

The axial uncertainty factor will be included only if it has not been accounted for in the MATP curves.

4.4. RPS DNB Margin Calculations

The rest of the DNB margin calculations are used to validate the RPS limits, so the operating limits restrictions on power distributions are not applied. The methodology for computing RPS DNB margin is the same as in Section 4.3, however the MATP curves are different. Table 3 lists the conditions at which the RPS MATP curves were generated and the conditions of the power distributions that will be used for each set of MATP curves.

4.5. Centerline Fuel Melt Margin Calculations

The centerline fuel melt limit is also used to validate the RPS limits, so the operating limits restrictions on power distributions are not applied in the calculation. Since there usually is a positive margin for centerline fuel melting, only the power distributions at 118% power are used for the centerline fuel melt margin calculations. A positive margin at 118% power will preclude negative margins at lower power levels. If the 118% power level results show negative margins, lower power levels will be analyzed to fully define the AFD - power level limit. The equations below show how the margin for centerline fuel melt is calculated. Note that the linear heat rate is calculated similarly to the LOCA margin calculation. Each node in the core model is analyzed, but only the minimum margin for a power distribution is used to determine the AFD - power level limits.

$$CFMM = Min\left\{\frac{MAXLHR - LHR(x,y,z)}{MAXLHR}\right\}*100$$

Where:

MAXLHR	= Maximum allowable linear heat rate in kw/ft.
LHR(x,y,z)[3]	= NP(x,y,z) * FP * AVGLHR * RADLOC(x,y,e) * TILT * RPF * UCT * AMF
NP(x,y,z)[3]	= Nodal power from the power distribution calculation.
FP	= Fraction of core power level, including power level uncertainty.
AVGLHR	= Total core power divided by the total length of fuel rods in the core, kw/ft, accounting for fuel densification and thermal expansion.
RADLOC(x,y,e)	= Burnup(e) dependent maximum rod to assembly power factor.
UCT	= Uncertainty factor on the pin total peak, including engineering hot channel factor and rod bow if not included in the fuel mechanical analysis (see Section 3.1).
TILT	= Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).
RPF	= Factor to account for the power generated in the fuel rod.
AMF	= Additional Margin Factor, optionally used to incorporate additional design margin.

The uncertainty on power level and the factor to account for power deposited in the fuel will be used only if these factors were not accounted for in the limiting heat generation rate.

[3] For SIMULATE, LHR does not include the RADLOC factor, since NP is the SIMULATE three dimensional local peak pin value.

4.6. Determining the AFD - Power Level Limits

The individual values of margin for each power distribution and margin calculation are collected into a database. For each power level and margin calculation, the margin data is plotted against AFD. The data points are connected by drawing lines between points with an equal independent parameter. Control rod position is usually chosen as this independent parameter, which means that different points along these lines represent different xenon time steps. The limit is set to preclude operation with negative peaking margin. At lower power levels, core conditions may not produce an AFD at the desired AFD limit. For this case, the AFD limit from the upper power level is extrapolated to the lower power level and the core conditions are verified to yield non-negative margins. Figures 7 and 8 shows an example plot of LOCA and LOFA DNB margin plotted against AFD, connected by equal rod position lines.

The operating AFD limits are determined by selecting the limiting of either the LOCA margin results or the LOFA DNB margin results at the various power levels analyzed. The AFD limits may be interpolated between rod position if the rod position chosen for the rod insertion limit was not explicitly modeled when the power distributions were generated. The bounding AFD envelope is adjusted to account for measurement system (two segment power-range excore nuclear detectors) uncertainties. The uncertainties account for the excore detector calibration error and drift between calibrations.

The DNB margin calculations performed for the RPS OTΔT AFD Trip penalty, f(ΔI), provide AFD limits [

] The power - AFD penalty is determined by selecting the limiting breakpoints and slopes defined by the [

] The uncertainty associated with the f(ΔI) function is combined with the uncertainties of the other OTΔT function input parameters in determining the adjusted K_1 constant in the setpoint equation (References 2, 3), or the f(ΔI) function is adjusted to account for the AFD uncertainties.

The centerline fuel melt protection criterion is associated with the OPΔT Trip f(ΔI) penalty function. Since the OPΔT f(ΔI) function is usually zero, the check performed at 118% power is adequate to verify that the penalty is not required. Should the centerline fuel melt margin calculations result in an AFD limit at 118% power, lower power levels would be analyzed in order to define the power - AFD penalty. The penalty could then be incorporated into the OPΔT trip function or the required protection could be provided by the OTΔT function.

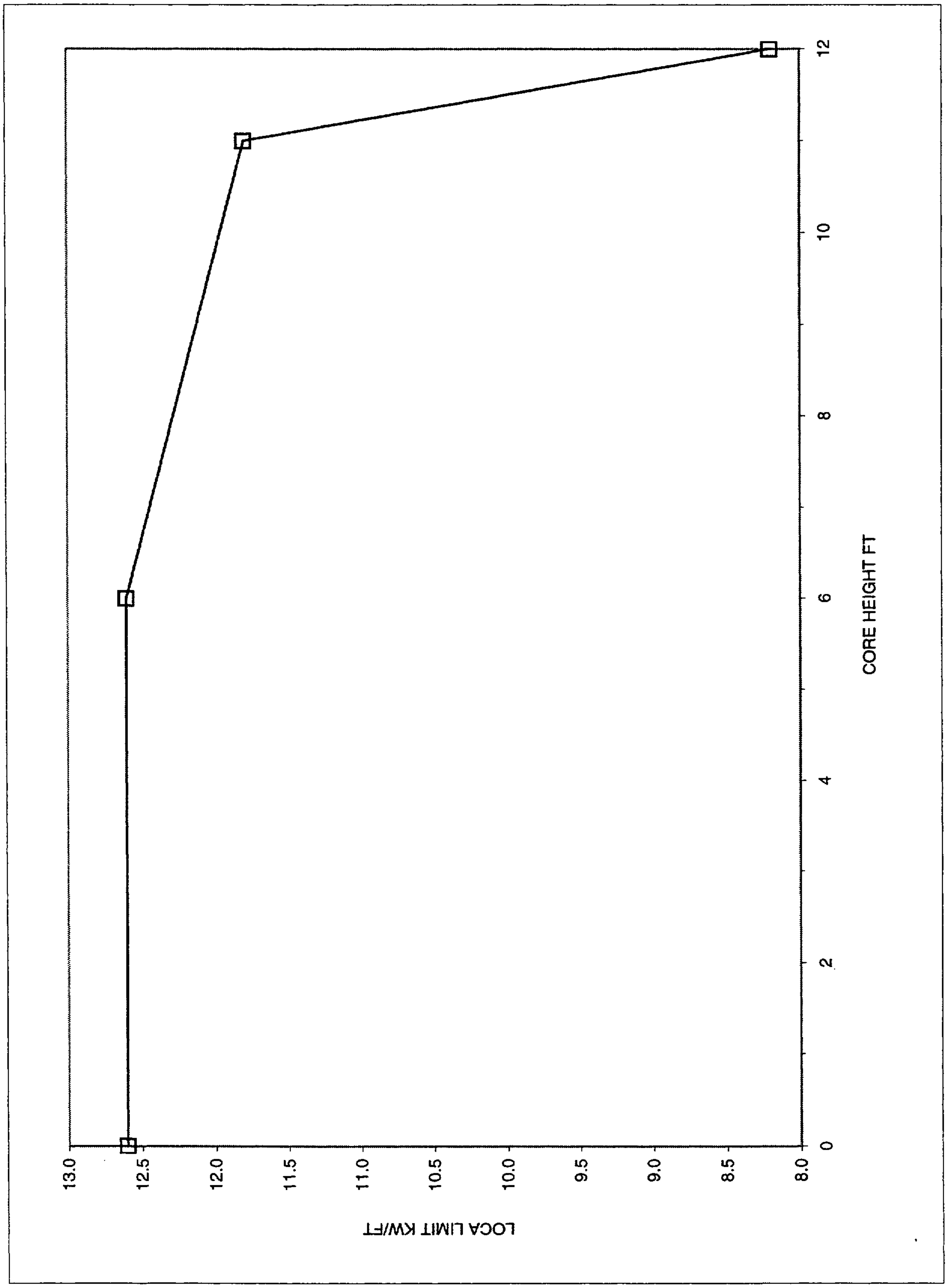
4.7. Control Rod Insertion Limits

The rod insertion limits are assumed when the operational AFD - power level limits are set. However, further iteration on the limits may be necessary depending on the results of the shutdown margin and ejected rod analyses. Adjustments are made to the rod insertion limits and AFD - power level limits as necessary.

Table 3

Typical RPS MATP Curve Conditions
and Conditions of the Power Distributions
used for each set of MATP Curves

Figure 6
Typical LOCA Linear Heat Rate Limits vs Core Height



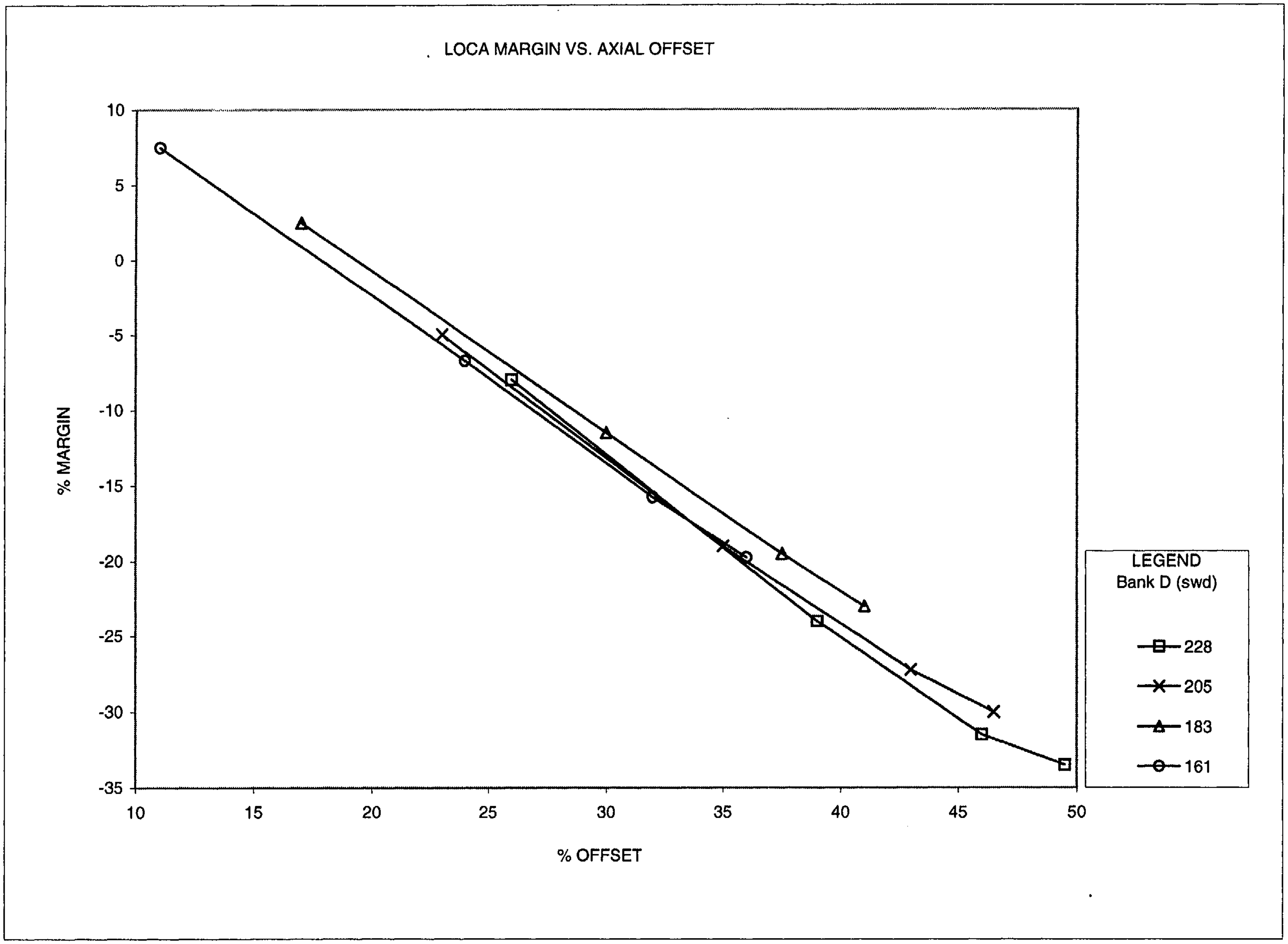


Figure 7
Sample LOCA Margin Plot

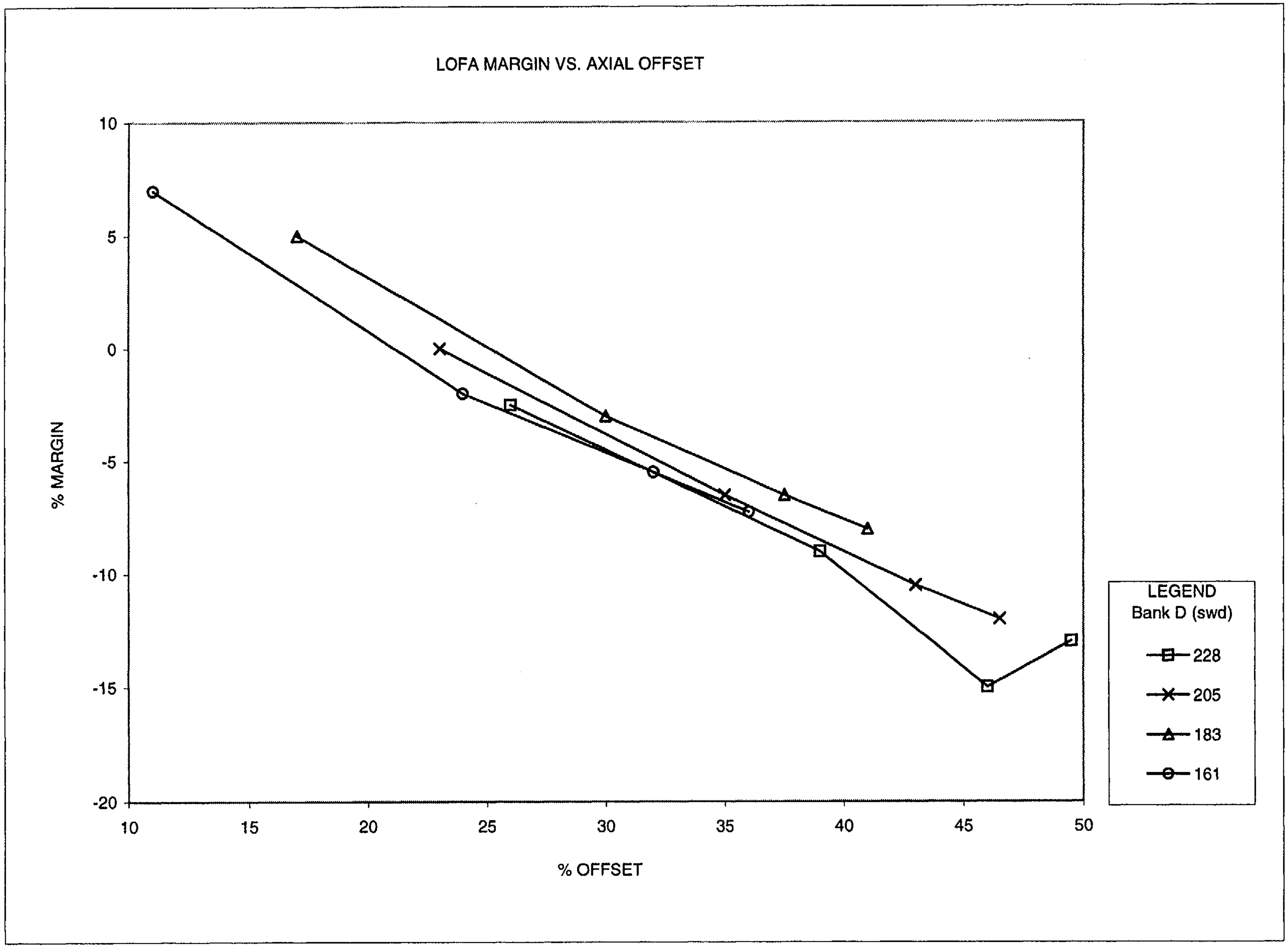


Figure 8
Sample LOFA Margin Plot

5. BASE LOAD LCO LIMITS

If the operational limits for a particular fuel cycle are too restrictive for normal operation, then a set of base load limits can be defined that may allow power operation at 100% power. Base load is defined as operating the reactor within a relatively narrow AFD band about a plant measured AFD target and within a limited power range. By limiting the allowed AFD - power level space, extra margin can be gained in the power distribution monitoring factors (see Section 6).

Base load limits and monitoring factors are computed the same as the operational limits, only the xenon transients will be re-defined so that they will be restricted to the base load operating band about a predicted AFD target. The power level at which the plant will be allowed to enter base load will be greater than or equal to the power level of the xenon transients.

6. POWER DISTRIBUTION SURVEILLANCE

The AFD - power level limits are set to preserve the power peaking assumptions in the LOCA analysis and to protect the fuel from damage during a LOFA when the power distribution is skewed in the axial direction. Similary, $f(\Delta I)$ limits are set to preclude RPS limits from being exceeded during Condition II transients. Because only steady state power distributions can be measured with reasonable accuracy, the limits on the measured power distribution are reduced by pre-calculated factors that account for perturbations from steady state conditions to applicable limits.

6.1. LOCA F_Q Surveillance Methodology

The Technical Specification (2, 3) LOCA F_Q limit that must be satisfied within the AFD - power level operating limits is:

$$F_Q^M(x,y,z) < \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q^M(x,y,z) < \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P < 0.5$$

Where: P = relative thermal power.

$K(Z)$ = normalized F_Q as a function of core height (see Figure 9).

F_Q^{RTP} = the LOCA limit at rated thermal power (RTP).

This criterion is a Technical Specification (2, 3) limiting condition for operation (LCO).

Using definitions from Section 4.2, the reduced limits for the measured F_Q are specified as:

$$F_Q^M(x,y,z)*UMT*MT*TILT \leq [\qquad\qquad]$$

Where:

$F_Q^M(x,y,z)$ = The measured total peak in location x,y,z.

UMT = The measurement uncertainty factor on the total peak, provided in the Technical Specifications (2, 3).

MT = Manufacturing tolerance factor (or engineering hot channel factor), provided in the Technical Specifications (2, 3).

TILT = Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

$F_Q^D(x,y,z)$[4] = $NP^D(x,y,z)$ * RADLOC(x,y,e) * UCT, design power distribution for F_Q.

$M_Q(x,y,z)$ = The LOCA margin remaining in location x,y,z in the calculated transient power distributions.

[ ]

6.2. <u>CFM F_Q Surveillance Methodology</u>

Using definitions from Section 4.5, the measured F_Q CFM surveillance limit is:

$F_Q^M(x,y,z)*UMT*MT*TILT \leq$ [ ]

Where the parameters in the above equation are defined in Section 6.1, except:

[ ]

[ ]

6.3. LOFA DNB $F_{\Delta H}$ Surveillance Methodology

The Technical Specification (2, 3) $F_{\Delta H}$ limit that must be satisfied within AFD - power level operating limits is:

$$F^{M}_{\Delta H}(x,y) \le MARP(x,y)*\left[1.0+\frac{1}{RRH}*(1.0-P)\right]$$

Where P is the relative thermal power. MARP(x,y) is the Maximum Allowed Radial Peak which is derived from the MATP curves (see Figure 10) by dividing the MATP by the axial peak term. This criterion is a Technical Specification (2, 3) LCO.

The limits for $F_{\Delta H}$ must be reduced for the same reason as the F_Q limits are reduced (see Section 6.1). Using definitions from Section 4.3, the reduced limit for monitoring $F_{\Delta H}$ is given in the following relationship:

$$F^{M}_{\Delta H}(x,y)*UMR*TILT \le [\qquad]$$

Where:

$F^{M}_{\Delta H}(x,y)$ = Measured value of $F_{\Delta H}$

[ ]

UMR = Uncertainty factor on the measured radial peaks, provided in the Technical Specifications (2, 3).

TILT = Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

[ ]

6.4. <u>Monitoring of Plant Measured Parameters</u>

During power operations, the power distribution is continuously monitored by the ex-core nuclear instrumentation. The parameters of interest to power distribution monitoring are the core power level, the AFD and the quadrant power tilt. Limitations are imposed on these three parameters by the maneuvering analysis. The maneuvering analysis also imposes limits on control rod positions during power operations. The power distribution is also measured periodically by the in-core instrumentation system. The results of these measurements are used to verify that the core is behaving as predicted by the maneuvering analysis or to adjust the AFD - power level limits if it is not. The surveillance of these parameters is described below.

6.4.1. <u>AFD - Power Level Limits</u>

During normal operations, the combination of AFD and power level must be maintained within the operating limits that are provided by the maneuvering analysis. Example AFD - power level limits are shown in Figure 11. Since the operating limits are a Limiting Condition of Operation (instead of a Limiting Safety System Setting), the plant would be allowed to operate outside of the operating AFD - power level limits for short periods of time if necessary. This allowance is meant to be used to increase the plant availability during transient situations and is not meant to be used for normal operation.

If the power distribution is unusually limiting (because of severe power peaking, for example), then base load operation may be used if it provided for by the maneuvering analysis. During base load operation, the measured AFD must be within a relatively small AFD band about a plant measured target AFD. The size of the AFD band is specified by the maneuvering analysis. Note that this target may or may not be within the AFD - power level operating limits. Base load may not be entered unless the plant has been relatively stable in AFD and power level for a period of time. The power level must be above the Allowed Power Level (APL - a value supplied by the maneuvering analysis) and the AFD must be within the AFD - power level operating limits. The power level may then be increased to a maximum of 100% rated thermal power or the Maximum Base Load Power (MBLP - a value described below).

6.4.2. Control Rod Insertion Limits

The control rods must be maintained within the insertion limits that were determined by the maneuvering analysis. Example limits are shown in Figure 12. These limits are a Limiting Condition of Operation, so operation outside of these limits is allowed for short periods of time.

6.4.3. Heat Flux Hot Channel Factor - $F_Q(x,y,z)$

The in-core instrumentation system is used periodically to measure $F_Q^M(x,y,z)$, which must always be within applicable limits. The LOCA limit is specified in the Technical Specifications (2, 3) and is shown in Section 6.1.

This limit on $F_Q^M(x,y,z)$ is a Limiting Condition of Operation, so operation outside of the limit is allowed for a short period of time to allow the operator to bring the reactor back within the limits without a reactor trip.

$F_Q^M(x,y,z)$ is usually measured at or near nominal conditions. To ensure that $F_Q^M(x,y,z)$ meets applicable limits for LOCA and CFM, the following limits are imposed at nominal conditions:

For nominal operation: $F_Q^M(x,y,z) \leq F_Q^L(x,y,z)^{OP}$ and
$F_Q^M(x,y,z) \leq F_Q^L(x,y,z)^{RPS}$, or

For base load operation: $F_Q^M(x,y,z) \leq F_Q^{Max\ BL}(x,y,z)$

$F_Q^L(x,y,z)^{OP}$ and $F_Q^L(x,y,z)^{RPS}$ are generated in the maneuvering analysis. These limits are specified in the Technical Specifications (2, 3) and are not imposed on the top or bottom 15% of the core. The limits on $F_Q^M(x,y,z)$ account for an appropriate measurement uncertainty, which is provided in the Technical Specifications (2, 3).

If $F_Q^M(x,y,z)$ exceeds $F_Q^L(x,y,z)^{OP}$ (LOCA limits), the AFD - power level limits must be adjusted by reducing the allowed AFD span (move the negative and positive AFD limits closer to the zero AFD point), so that positive margin would be maintained at the extremes of the AFD - power level operating limits. If $F_Q^M(x,y,z)$ exceeds $F_Q^L(x,y,z)^{RPS}$ (CFM limits), then a reduction is made to the OTΔT trip setpoints.

For base load operation, reactor power must be reduced until the above limit on $F_Q^{Max\ BL}(x,y,z)$ is satisfied. For base load operation, reactor thermal power may not exceed the Maximum Base Load Power (MBLP), which is defined as:

$$\text{MBLP} = \min_{\substack{\text{over} \\ (x,y,z)}} \frac{F_Q^{Max\ BL}(x,y,z)*100\%}{F_Q^M(x,y,z)}$$

Note that this is equivalent to saying that $F_Q^M(x,y,z)$ may not exceed $F_Q^{Max\ BL}(x,y,z)$ for base load operation.

6.4.4. Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(x,y)$

$F_{\Delta H}^M(x,y)$ is measured at the same time that $F_Q^M(x,y,z)$ is measured with the in-core instrumentation system. $F_{\Delta H}^M(x,y)$ must be within the maximum allowed values used in the maneuvering analysis (see Figure 10 for sample MATP curves). This limit is a Limiting Condition of Operation, so operation outside of this limit is permitted for a period of time to allow the operator to bring the reactor back within the limit without a reactor trip.

$F_{\Delta H}^{M}(x,y)$ is usually measured at or near nominal conditions. To ensure that $F_{\Delta H}^{M}(x,y)$ meets applicable limits for LOFA, the following limits are imposed at nominal conditions:

For nominal operation: $F_{\Delta H}^{M}(x,y) \leq F_{\Delta H}^{L}(x,y)^{SURV}$

For base load operation: $F_{\Delta H}^{M}(x,y) \leq F_{\Delta H}^{Max\ BL}(x,y)$

If the appropriate relationship is not satisfied, then the reactor power will be reduced until it is satisfied. The limits on $F_{\Delta H}^{M}(x,y)$ account for an appropriate measurement uncertainty, which is provided in the Technical Specifications (2, 3).

6.4.5. Quadrant Power Tilt

An allowance for a 2% quadrant power tilt was made in the AFD - power level operating limits and in the values of $F_{Q}^{L}(x,y,z)^{OP}$, $F_{Q}^{L}(x,y,z)^{RPS}$, $F_{Q}^{Max\ BL}(x,y,z)$, $F_{\Delta H}^{L}(x,y)^{SURV}$, and $F_{\Delta H}^{Max\ BL}(x,y)$. Thus, no action is required for an indicated quadrant power tilt of up to 2%. A quadrant power tilt larger than 2% is a Limiting Condition of Operation, so operation of the plant is allowed to continue for a period of time while the operator attempts to correct the condition.

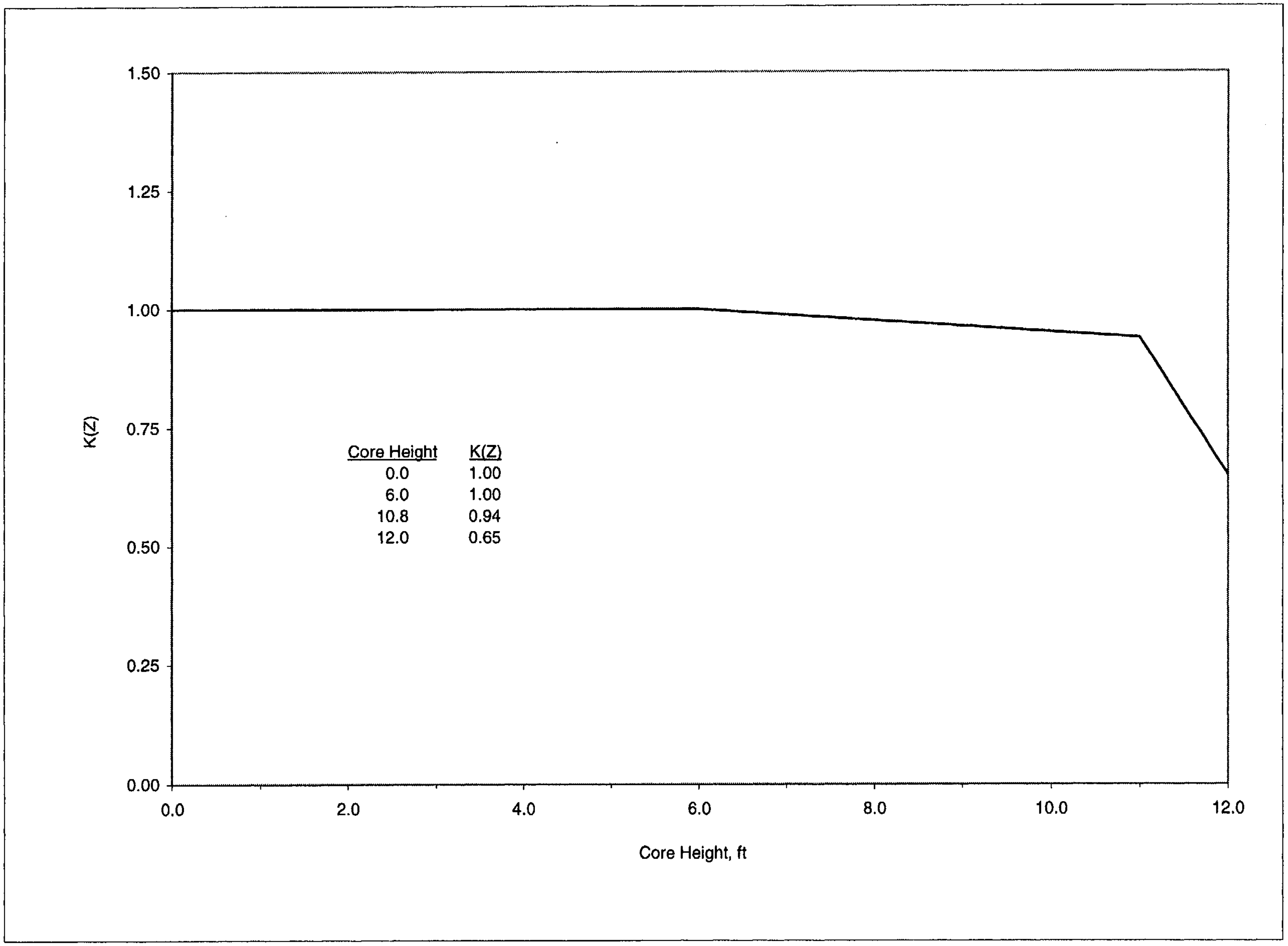


Figure 9

K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

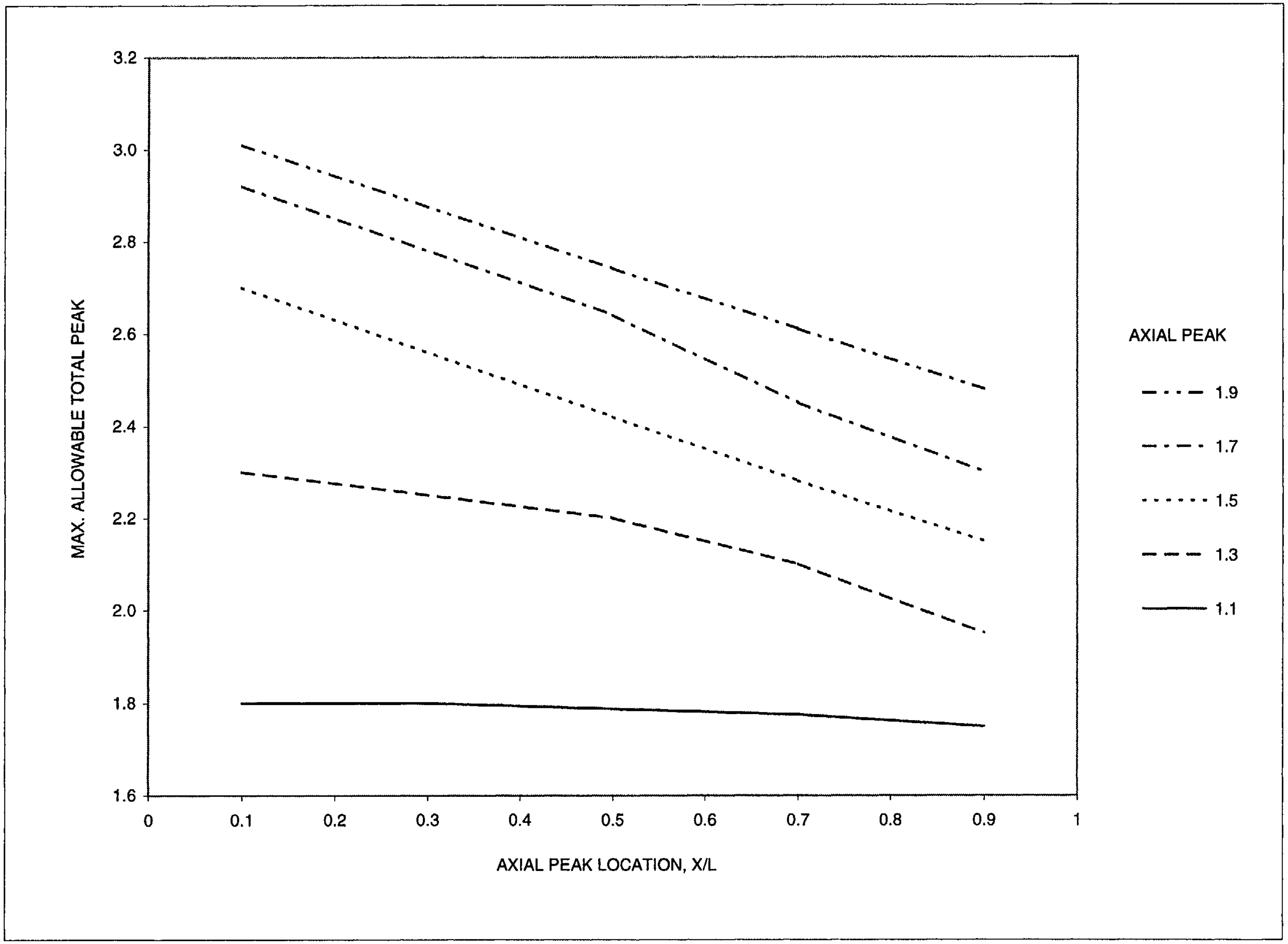


Figure 10
Sample LOFA DNB MATP Curves for 100% Power

Figure 11
Sample AFD - Power Level Operating Space

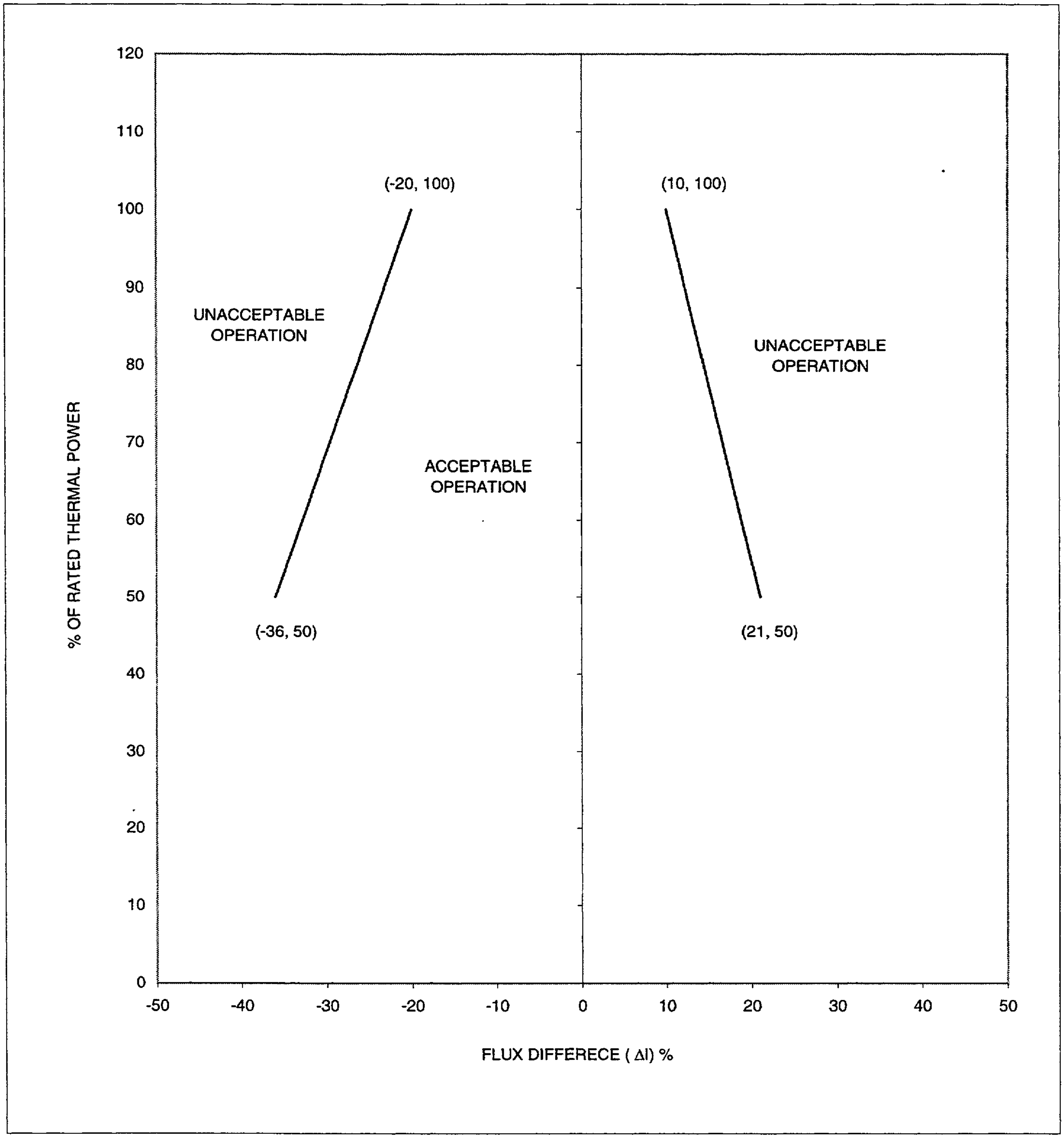
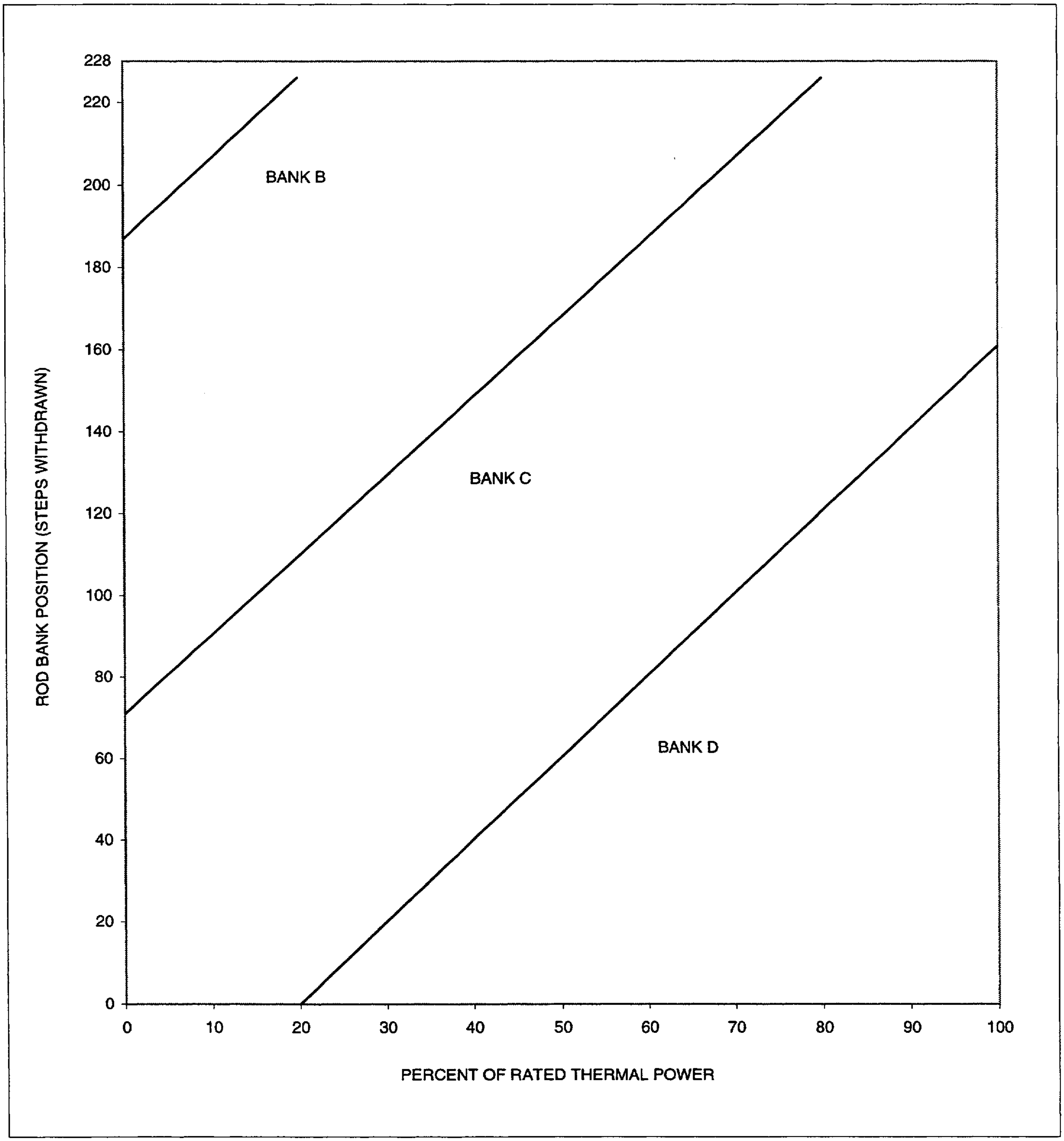


Figure 12
Control Rod Insertion Limits vs. Thermal Power



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12. "Computer Code Certification for MARGINS," DPC internal document.

13. "Computer Code Certification for MARGINPLOT," DPC internal document.

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APPENDIX A

Computer Program Description

EPRI-NODE-P

NODE (8) is a three dimensional nodal program that is derived from FLARE (9). NODE computes a three dimensional power distribution with thermal hydraulic feedback, the core multiplication factor, the fuel burnup distribution and maintains a reactivity inventory. The physics models within NODE account for the presence of control rods, fuel and moderator temperatures, fixed burnable poisons, soluble boron, fuel depletion, and time dependent xenon and iodine. The input to NODE is generated either from CASMO-2E (7) data or from EPRI-CELL (6) color set PDQ data.

PDQ07

PDQ07 (10) is an industry accepted multi-group, multi-dimensional, neutron diffusion depletion program. The Combustion Engineering version of PDQ that is used by DPC has been modified with a two dimensional thermal hydraulic feedback model to account for fuel and moderator temperature distributions. PDQ uses cross sections from either CASMO-2E or EPRI-CELL.

PDQEDIT

PDQEDIT (11) is a utility program that reads the PDQ system files. The program has several abilities, one of which is to produce radial local power factors from the mesh average power file.

MARGINS

MARGINS (12) is a program written by DPC that computes the margin to thermal limits for LOCA F_Q, DNB and centerline fuel melt. MARGINS requires three dimensional power distribution data for input. The output of MARGINS is a file that contains one entry per power distribution; the entry contains the case and limit type identifiers, the core axial offset and the core margin to the thermal limit evaluated.

MARGINPLOT

MARGINPLOT (13) is a program written by DPC that plots the MARGINS data and computes the zero margin intercepts for the thermal limits data.

SIMULATE-3

SIMULATE (15) is an advanced two-group nodal code written by Studsvik based on the QPANDA neutronics model. SIMULATE computes three dimensional nodal and pin power distributions accounting for fuel and moderator temperature, fuel burnup, xenon distributions, control rods, burnable absorbers, and soluble boron. Cross-section input to SIMULATE is provided from CASMO.

APPENDIX B

NRC/DPC Correspondence Regarding NRC Request for Additional Information

5 A5

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

MAR 3 1989

Mr. Hal B. Tucker, Vice President
Nuclear Production
Duke Power Company
P. O. Box 33189
Charlotte, NC 28242

Dear Mr. Tucker:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE NUCLEAR DESIGN METHODOLOGY FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS, TOPICAL REPORT DPC-NE-2011P

The Reactor Systems Branch has reviewed the subject topical report and has concluded that additional information is required for us to complete this review.

Please submit the responses to the questions in the enclosure within 45 days of the receipt of this letter to enable the staff to complete its review. If you need any clarification, please contact Lambros Lois of my staff at 301-492-0890.

Sincerely,

M. Wayne Hodges

M. Wayne Hodges, Chief
Reactor Systems Branch
Division of Engineering & Systems Technology

Enclosure:
As stated

REQUEST FOR ADDITIONAL INFORMATION
DPC-NE-2011P

1. How do operating limits obtained via this methodology compare to limits based on the use of the present RAOC methodology?

2. Is the potential increase in the available margin associated with the subject methodology due solely to the use of three-dimensional analyses/monitoring, or do other aspects contribute?

3. There is no indication in that the methodology employed in generating and using the LOFA DNB MATP curves has been reviewed and accepted by the NRC.

4. The procedure for generating power distributions appears to involve running two xenon transients at each of three times in a cycle, followed by using xenon distributions from each transient/time-in-life to calculate instantaneous power distributions associated with various combinations of power level, inlet temperature and control rod bank position, as well as those occurring during the course of several anticipated transients.

   It appears that only four xenon distributions from each transient at each time in life are used along with the statepoint configurations given in Table 2. Please clarify/elaborate as to how many power level/inlet temperature/control rod/xenon statepoints are evaluated at each time in life.

5. Is there demonstrated assurance that the power distributions resulting from the above analyses are indeed conservative with respect to those that might occur, and that they sufficiently span the AFD/rod insertion power level operating spaces to permit an accurate determination of operating limits?

6. What is the basis for the 15 minute limit assumed in the analysis of the boron dilution accident?

7. Radial local factors appear to be obtained from a nominal all-rods-out depletion calculation for the cycle and are, therefore, only functions of assembly type and burnup. However, local peaking should also be affected by transient xenon, control presence, etc. What is the basis for not accounting for these effects?

8. What are the other components of UCT in addition to those specifically mentioned in 3.1?

9. How are the axial peaking due to grid spacers and densification spike effects accounted for in the margin calculations?

10. Please explain the basis for the use of SC in the CFMM calculation, and its form.

11. Please explain why the uncertainties considered in the linear heat rate equation for the CFMM calculation are different from those used in obtaining LOCAM given that they refer to the same basic quantity.

12. The definition of the TILT factor varies while its value appears to be constant. Please explain/elaborate.

13. Since the maneuvering analysis involves two xenon transients at three times in core life there are six F_Q^D and six $F_{\Delta H}^D$ design distributions available for comparisons to measurements. Have the errors introduced by the subsequent interpolation on cycle burnup and power level been quantified and included in the analysis? How are mismatches between the measured and design data associated with AFD and control rod position differences accounted for?

14. Are M_Q and $M_{\Delta H}$ minimum values over the cycle?

15. How are possible increases in peaking between measurements due to mechanisms other than tilt (e.g. burnup) accounted for in the F_Q and $F_{\Delta H}$ surveillance?

16. What are the similarities/differneces between base load operation and CAOC?

17. Under what conditions would the AFD target and operating band for base load operation not fall within the normal AFD-power level operating limits?

18. Why are the uncertainties associated with $F^D_{\Delta H}$ and $F^T_{\Delta H}$ in 6.2 different?

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

March 28, 1989

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

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Topical Report DPC-NE-2011P,
"Nuclear Design Methodology for Core
Operating Limits of Westinghouse Reactors";
Response to Request for Additional Information

Attached are responses to questions regarding the subject topical report, which were transmitted by letter dated March 3, 1989.

Please note that the proprietary nature of the original topical report, as identified in my April 27, 1988 transmittal letter and accompanying affidavit, is maintained in the responses to these questions. Therefore, they should be withheld from public disclosure.

Very truly yours,

Hal B. Tucker

H. B. Tucker

SAG154/lcs

xc: Mr. Darl S. Hood, Project Manager
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R. M. Glover - CNS
File: MC, CN-801.01

Q1 How do operating limits obtained via this methodology compare to limits based on the use of the present RAOC methodology?

A1 The operating space AFD limits from this method are expected to be a few percent wider than the current RAOC limits.

Q2 Is the potential increase in the available margin associated with the subject methodology due solely to the use of three-dimensional analyses/monitoring, or do other aspects contribute?

A2 The margin increase is due primarily to analysis of three-dimensional power distributions, as opposed to the 1D/2D synthesized power distribution that the RAOC limits are based on.

Q3 There is no indication in that the methodology employed in generating and using the LOFA DNB MATP curves has been reviewed and accepted by the NRC.

A3 The general methodology for generating DNB MATP curves has previously been approved by the NRC as applied to Oconee Nuclear Station in the SER for the topical report, "Duke Power Company, Oconee Nuclear Station, Reload Design Methodology," NFS-1001A, April 1984. A topical report describing the codes and methods used by Duke Power for generating DNB MATP limits specifically for Westinghouse reactors was submitted to the NRC in January 1989 under the title, "Duke Power Company, McGuire and Catawba Nuclear Stations, Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004. When approved, this methodology will be used to generate DNB MATP curves for setting the core limits.

Q4 The procedure for generating power distributions appears to involve running two xenon transients at each of three times in a cycle, followed by using xenon distributions from each transient/time-in-life to calculate instantaneous power distributions associated with various combinations of power level, inlet temperature and control rod bank position, as well as those occurring during the course of several anticipated transients.

It appears that only four xenon distributions from each transient at each time in life are used along with the statepoint configurations given in Table 2. Please clarify/elaborate as to how many power level/inlet temperature/control rod/xenon statepoints are evaluated at each time in life.

A4 The matrix of statepoints shown below will be used as an initial guide and may be modified as experience is accumulated. A power distribution will be analyzed for each statepoint in the matrix below for[ ] That is, a set of[ ]three-dimensional power distributions will be analyzed to set limits.

List of State Points

[ ]

Q5 Is there demonstrated assurance that the power distributions resulting from the above analyses are indeed conservative with respect to those that might occur, and that they sufficiently span the AFD/rod insertion power level operating spaces to permit an accurate determination of operating limits?

A5 Yes. [ ]

As shown in the response to question 4, the statepoint conditions will span the allowable rod insertion limits and the accident condition rod insertions as described in section 2.4 of the report. The power distributions will generally span the AFD space, although some extrapolation on AFD may be required at times. Therefore, the AFD/rod insertion space will be sufficiently analyzed to accurately determine the operating limits.

Q6 What is the basis for the 15 minute limit assumed in the analysis of the boron dilution accident?

A6 The 15 minute limit is based on the operator action time acceptance criteria of the Standard Review Plan, section 15.4.6-II.

Q7 Radial local factors appear to be obtained from a nominal all-rods-out depletion calculation for the cycle and are, therefore, only functions of assembly type and burnup. However, local peaking should also be affected by transient xenon, control presence, etc. What is the basis for not accounting for these effects?

A7 Duke Power has examined the effects of control rods and transient xenon on local peaking factors using both two-dimensional and three-dimensional models. In general, it has been observed that the limiting nodes in a specific case are located away from the inserted control rods. That is, the peak nodal power occurs in an unrodded plane and/or an assembly removed from the rodded assemblies by several assembly pitches. Therefore, the intra-assembly flux distribution of the limiting node is relatively unaffected by the flux gradients induced locally near the rodded assembly. Similarly, the transient xenon distributions, while significantly skewed globally, do not cause significant changes in local power distributions.

Q8 What are the other components of UCT in addition to those specifically mentioned in 3.1?

A8 UCT is defined in Reference 4 of the report to be

$$1 + (.031/1.375) + \sqrt{(.03)^2 + (.035)^2 + (.02)^2} = 1.073.$$

The term (.031/1.375) accounts for a small bias in the calculated power distributions.

Q9 How are the axial peaking due to grid spacers and densification spike effects accounted for in the margin calculations?

A9 In the development of the observed reliability factors the calculated peaks did not include any grid effects while the measured data did. Therefore, the effects of the grid on peaking are inherently included in the observed reliability factors which are applied to the calculated values.

Current fuel designs used by Duke Power specify fuel pellet density greater than or equal to 95% of theoretical density. Results of hot cell and gamma scan measurements on fuel rods containing pellets of these densities have not shown any significant gap formation. Thus, no power peaking penalty will be taken for densification power spikes.

Q10 Please explain the basis for the use of SC in the CFMM calculation, and its form.

A10 A rod bow penalty is applied to the calculated peak when computing CFMM. However, since rod bow is considered to be independent of the calculational uncertainty, it is statistically combined with the engineering and power distribution factors in the equation for UCT found in Reference 4 of the report. The algebraic derivation is shown below:

$$UCT = 1 + .031/1.375 + \sqrt{(.03)^2 + (.035)^2 + (.02)^2}$$

$$SC = 1 + .031/1.375 + \sqrt{(.03)^2 + (.035)^2 + (.02)^2 + (RBOW-1)^2}$$

$$\sqrt{(.03)^2 + (.035)^2 + (.02)^2} = UCT - 1 - .031/1.375$$

$$SC = 1 + .035/1.375 + \sqrt{(UCT - 1 - .031/1.375)^2 + (RBOW-1)^2} .$$

Q11 Please explain why the uncertainties considered in the linear heat rate equation for the CFMM calculation are different from those used in obtaining LOCAM given that they refer to the same basic quantity.

A11 The only difference is that the rod bow penalty is not applied to the LOCA limits, since any increase in peaking will be compensated for by the increased coolant flow.

Q12 The definition of the TILT factor varies while its value appears to be constant. Please explain/elaborate.

A12 The magnitude of the tilt factor is the same in all sections and the correct definition in all sections is "peaking increase due to allowable quadrant tilt."

Q13 Since the maneuvering analysis involves two xenon transients at three times in core life there are six F_Q^D and six $F_{\Delta H}^D$ design distributions available for comparisons to measurements. Have the errors introduced by the subsequent interpolation on cycle burnup and power level been quantified and included in the analysis? How are mismatches between the measured and design data associated with AFD and control rod position differences accounted for?

A13 The values of F_Q^D and $F_{\Delta H}^D$ from the design power distributions are not the values that are compared to measurements. [

]

This is very similar to the current monitoring methods which apply burnup-dependent W(Z) transient peaking factors to the measured peaks.

[ ]

The impact on peaking of differences between the measured and design data for AFD are inherently included in the uncertainty factors which are applied to the predicted peaks. The uncertainty factor used is an observed nuclear reliability factor developed by matching reactor power and rod positions between predicted and measured statepoints. The calculated AFD was allowed to vary from the measured value in these calculations, although these differences are generally within 2%. The impact of control rod position differences between measured and design data is considered negligible since power distribution maps are usually taken at nearly all-rods-out conditions.

Q14 Are M_Q and $M_{\Delta H}$ minimum values over the cycle?

A14 [ ]

Q15 How are possible increases in peaking between measurements due to mechanisms other than tilt (e.g., burnup) accounted for in the F_Q and $F_{\Delta H}$ surveillance?

A15 If F_Q^M is greater than F_Q^{Max}, then the AFD power level space is reduced by an appropriate amount such that F_Q^T, at the new AFD limit, will be within the LOCA limits.

If $F_{\Delta H}^M$ is greater than $F_{\Delta H}^{Max}$, then power level will be reduced until the limit is met.

If the margins to the limits are found to be decreasing over successive measurements, then either the measurement frequency will be increased or the margins will be reevaluated with an additional penalty to account for the expected peaking increase to the next measurement.

Q16 What are the similarities/differences between base load operation and CAOC?

A16 The only significant difference is the power level at which the mode of operation may be entered. Base load operation is typically entered at 80% power after stabilizing the plant at the target AFD. CAOC is used for the full range of power operation.

Q17 Under what conditions would the AFD target and operating band for base load operation not fall within the normal AFD-power level operating limits?

A17 This condition is not expected to occur since the AFD-power level limits will be set each cycle with a cycle specific three-dimensional core model. However, operating for a significant period of time at reduced power may cause the AFD target to be outside of the operating AFD space. If this condition should occur, the surveillance of the measured peaking will ensure that the allowable limits are not exceeded and tighter AFD limits would be used to minimize potential transient peaking.

Q18 Why are the uncertainties associated with $F^D_{\Delta H}$ and $F^T_{\Delta H}$ in 6.2 different?

A18 A typographical error was made in the equation for $F^T_{\Delta H}$. The equation that was intended is:

[ ]

However, in further research it was discovered that a rod bow penalty does not need to be applied to a limit that is related to DNB. This approach has previously been approved by the NRC in the SER to "Duke Power Company Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002A, October 1985. Thus, the uncertainties in $F^D_{\Delta H}$ and $F^T_{\Delta H}$ should be the same. The correct equation for $F^T_{\Delta H}$ is:

[ ]

Also, the rod bow penalty should be removed from the calculation of DNBM. In section 4.3 of the report, the equation for RPP(x,y) should be:

[ ]

APPENDIX C

Original Issue NRC SER

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

January 24, 1990

RECEIVED
JAN 29 1990
DUKE POWER CO.
REGULATORY COMPLIANCE

Mr. H. B. Tucker, Vice President
Nuclear Production
Duke Power Company
P. O. Box 33189
Charlotte, NC 28242

Dear Mr. Tucker:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT DPC-NE-2011P, "DUKE POWER COMPANY NUCLEAR DESIGN METHODOLOGY FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS"

The staff has completed its review of the Topical Report DPC-NE-2011P, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors" submitted for NRC review by the Duke Power Company by letter dated April 27, 1988. Additional information was submitted on March 28, 1989. This topical report (DPC-NE-2011P) provides information and justification for the operating limits on power distribution, control rod insertion and power distribution inputs to the overpower-delta-T and overtemperature-delta-T reactor protection system trip functions. These limits are the axial flux difference for a given power level, the rod insertion limits and the f(delta-I) function of the overpower- and overtemperature-delta-T. These operating limits provide assurance that the peak local power is not greater than that assumed in the design basis transient and accident analyses. The limits are set such that the RPS will trip the reactor before fuel damage occurs. A three-dimensional reactor model power distribution is employed for the maneuvering analyses in several points in the core life. These power distributions are based on a set of conservative xenon distributions to ensure that the predicted power distributions are conservative with respect to those expected to occur. These power distributions are augmented by appropriate uncertainty factors.

We find the application of DPC-NE-2011P to be acceptable for referencing in license applications to the extent specified, and under the limitations delineated, in DPC-NE-2011P and the associated NRC technical evaluation. The evaluation defines the basis for acceptance of this topical report.

We do not intend to repeat our review of the matters found acceptable as described in DPC-NE-2011P when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the application of DPC-NE-2011P.

In accordance with procedures established in NUREG-0390, it is requested that the Duke Power Company publish accepted versions of this topical report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, Duke Power Company and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Scott Newberry

for Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
DPC-NE-2011P Evaluation

ENCLOSURE

SAFETY EVALUATION FOR THE TOPICAL REPORT DPC-NE-2011P "DUKE POWER COMPANY, NUCLEAR DESIGN METHODOLOGY FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS"

1.0 INTRODUCTION

By letter dated April 27, 1988, the Duke Power Company submitted the Topical Report DPC-NE-2011P for NRC review (Ref. 1). Additional information was submitted on March 28, 1989 (Ref. 2). This topical report provides information and justification for the operating limits on power distribution, control rod insertion and power distribution inputs to the overpower-delta-T and overtemperature-delta-T reactor protection system trip functions. These limits are the axial flux difference for a given power level, the rod insertion limits and the f(delta-I) function of the overpower- and overtemperature-delta-T. These operating limits provide assurance that the peak local power is not greater than that assumed in the design basis transient and accident analyses. The limits are set such that the RPS will trip the reactor before fuel damage occurs. A three-dimensional reactor model power distribution is employed for the maneuvering analyses in several points in the core life. These power distributions are based on a set of conservative xenon distributions to ensure that the predicted power distributions are conservative with respect to those expected to occur. These power distributions are augmented by appropriate uncertainty factors.

The following evaluation incorporates our consultant's, BNL, contribution to this review. Restrictions to be observed in the application of this topical report are listed in Section 3.5.

2.0 SUMMARY OF THE TOPICAL REPORT

At first the report describes the three-dimensional nodal power and xenon distribution generation method which is based on an NRC approved version of the EPRI-NODE-P code (Ref. 3). The local radial factors are estimated using a pin-by-pin PDQ-07 model. Power distributions are generated for different times in the cycle. Limiting xenon distributions are generated to assure conservatism. The power distribution is augmented by uncertainty factors which account for the (X-Y) power distribution calculation uncertainty, quadrant tilt and axial power distribution.

The general methodology for the limiting condition of operation and the reactor protection system limits is followed by the calculation of the LOCA margin and the estimation of the loss of flow DNB limits. In addition the reactor protection system margin, the centerline fuel melt margin, the axial flux difference power level limits and the control rod insertion limits are calculated.

The power distribution surveillance and their relation to the operation and transient limits are then estimated for the LOCA F_Q limits, the loss of flow DNB, $F_{\Delta H}$, axial flux difference power level limits, control rod insertion limits, the heat flux hot channel factor, the nuclear enthalpy rise hot channel factor and the quadrant power tilt.

Appendix A in the report gives a brief description of the computer codes used in the above calculations.

3.0 EVALUATION

The proposed methodology employs a three-dimensional reactor and cycle specific model in conjunction with xenon distributions obtained from a maneuvering analysis which simulates severe xenon transients. Bounding power distributions are then generated based on these severe xenon distributions, and various combinations of rod positions, inlet temperature, power level and cycle burnup. These power distributions are compared to operating and safety thermal limits to define or validate the axial flux difference (AFD) power level operating space, the rod insertion limits and the f(delta-I) penalty function employed in the OPΔT and/or the OTΔT trip functions of the Reactor Protection System (RPS) such that power distributions that might exceed the respective thermal limits are prohibited. In addition to the xenon transient based power distributions, a number of anticipated transients (e.g., boron dilution, rod withdrawal, etc.) are analyzed in setting the RPS limits. A core monitoring/surveillance procedure which assures safe operation within the applicable limits is an integral part of the proposed methodology. This approach is an alternative to the Relaxed Axial Offset Control (RAOC) methodology (Ref. 4) currently in use at Duke Power Company's (DPC) McGuire and Catawba Nuclear Stations.

The present review considered the information provided in the topical report along with additional information provided by DPC in response to a request for additional information (RAI) (Ref. 5).

The computer codes and associated methodologies employed in the power distribution and peaking calculations have been previously reviewed by the NRC and found to be acceptable (Refs. 6 and 7). The shutdown margin and ejected rod analyses that enter into the setting of control rod insertion limits have also been approved by the NRC. A topical report describing the codes and methods to be used by DPC to generate the core thermal hydraulics (including hot rod) for Westinghouse (W) reactors is presently under review (Ref. 9).

In view of the above, and noting that the DPC methods for determining maximum allowable LOCA peaking and loss of flow accident (LOFA) DNB based operating limits and maximum allowable DNB and linear heat rate based RPS limits have been approved by the NRC, the acceptability of the proposed methodology hinges on the following major issues.

3.1 Operating Space AFD Limits

Since the proposed methodology represents a departure from currently accepted practice, any changes in limits relative to those obtained with the presently employed and approved RAOC methodology that represent a reduction in conservatism must be justified.

DPC has indicated that the proposed methodology will yield operating space AFD limits that are a few percent wider (less conservative) than the current RAOC limits; this is due primarily to the use of explicit three-dimensional (3-D) power distributions as opposed to the synthesized 3-D power distributions on which RAOC is based. The increase in the available margin, and consequently the AFD operating space limits, is consistent with previous experience that supports a reduction in peaking when explicit 3-D power distributions are used as compared to synthesizing 3-D distributions from 1-D and 2-D calculations.

Under the proposed DPC methodology, if operating limits are too restrictive for normal operation, a set of limits can be defined that may still allow operation at full power. The resulting "base load" operation is typically used above 80 percent power and is similar to the widely used and accepted constant axial offset control (CAOC) approach. The xenon distributions used in setting the limits in this case are restricted to a relatively narrow operating band about a predicted AFD target.

It is therefore concluded that the DPC approach is acceptable with respect to AFD limits.

3.2 Conservatism of Power Distributions

In order to have confidence in the operating and RPS limits obtained by the proposed methodology, there must be demonstrated assurance that the power distributions resulting from the DPC approach are conservative with respect to those that might be reasonably expected to occur, and that they sufficiently span the AFD/rod-insertion power-level operating spaces to permit an accurate determination of limits.

DPC has determined through sensitivity studies that the power distributions employed in setting the operating and RPS limits are conservative. This is due in part to the severity of the xenon transients employed in the maneuvering analyses and conservative modelling assumptions. In addition, since the limits are based on the analyses of almost 3000 three-dimensional power distributions (resulting from a matrix of power level/rod position/inlet temperature/burnup and xenon distribution statepoints), DPC is confident that the operating limits can be determined accurately, and any extrapolation would be minimal. A review of the statepoints (combinations of power level, rod insertion, etc.) and anticipated transients considered by DPC in generating bounding power distributions supports the conclusion that there is assurance that the power distributions assumed in the analyses of thermal limits are indeed conservative relative to the expected distributions, and this aspect of the DPC methodology is acceptable. It should be noted that the matrix of statepoints currently considered in the analysis may be modified as experience is accumulated. However, any reductions in the number of statepoints considered should be implemented only if there are no concomitant adverse effects (e.g., excessive interpolations required to set limits).

3.3 Uncertainties and Parameters in Margin and Monitoring Algorithms

The DPC methodology requires the determination of margins to linear heat rate and DNB thermal limits and the monitoring of the measured state to assure that operation is consistent with the DPC analyses performed to ensure that these limits will not be violated. Two linear heat rate related margins are determined - an operating limit based on LOCA considerations and an RPS limit that protects against centerline fuel melt. Similarly, two DNB related margins are also determined - an operating limit based on LOFA considerations and an RPS limit. In the core surveillance, precalculated factors based on the maneuvering analyses and the available margins are used to define an F_Q^{Max} and $F_{\Delta H}^{Max}$ which are then compared to measured values to determine whether the core is behaving as expected.

The equations used in the determination of the margins, including the uncertainties, were reviewed and found to be acceptable. The components of the margin equations used in the determination of linear heat rate and DNB are justified, and the values of the uncertainties applied have been previously reviewed and approved by the NRC.

Since only steady-state power distributions can be measured with reasonable accuracy, changes in the margins to limits accompanying deviations from steady-state conditions must be determined on the basis of calculations. The measured values of F_Q^{Max} and $F_{\Delta H}^{Max}$ are therefore compared to maximum allowable values that account for the minimum margins determined in the maneuvering analysis to ensure that the limits on the measured values will be met at the extremes of the AFD-power level operating limits. If the measured values of F_Q^{Max} or $F_{\Delta H}^{Max}$ exceed their respective limits, then the AFD-power level limits and the f(delta-I) function in the OPΔT trip function are adjusted and/or the power level is reduced. The trends in the margins to the limits are monitored from measurement-to-measurement, and the measurement frequency is increased or an additional penalty is included in the margins if increased peaking is expected. Monitoring in the case of base load operation is similar. This monitoring philosophy is similar to that currently employed in connection with RAOC. The factors and uncertainties (and related methodologies) applied in the comparisons to measurements are justified, and the DPC methodology is acceptable.

3.4 Evaluation Summary

Based on the review of the topical report and the additional information provided, and recognizing that the NRC has reviewed and approved the computer codes and some components of the proposed methodology (e.g., the generation and use of DNB MATP curves), it is concluded that the DPC analysis represents an acceptable approach for determining and monitoring core operating and RPS limits for the McGuire and Catawba Nuclear Stations. The proposed methodology, however, should be confirmed by continued calculation-to-measurement comparisons, and monitoring of trends or any loss of conservatism. While the application of the methodology to other four-loop, 193-assembly W PWRs is acceptable, the appropriate, plant specific reactor systems aspects must be considered and justified.

3.5 Restrictions

The following restrictions are imposed on the use of the Nuclear Design Methodology described in DPC-NE-2011:

(1) Application of this methodology is to be limited to the McGuire and Catawba nuclear power stations,

(2) Application to other Westinghouse 193-assembly plants would be acceptable provided that plant-specific differences be considered and justified,

(3) Application of this methodology is contingent upon NRC approval of the Reload Design Thermal-Hydraulic Methodology DPC-NE-2004 (presently under NRC review) using the VIPRE-01 code, and

(4) Calculation of power and xenon distributions are limited to the use of the EPRI-NODE-P and the PDQ-07 codes.

4.0 REFERENCES

1. Letter from H. B. Tucker Duke Power Company to USNRC, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," dated April 27, 1988.

2. Letter from H. B. Tucker Duke Power Company to USNRC, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors - Response to Request for Additional Information," dated March 28, 1989.

3. Letter from C. O. Thomas NRC, to H. B. Tucker Duke Power Company, dated March 13, 1985.

4. WCAP-10216-PA, "Relaxation of Constant Axial Offset Control, F(q) Surveillance Technical Specification," June 1983.

5. Letter from H. B. Tucker (DPC) to NRC, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors - Response to Request for Additional Information," March 28, 1989.

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

7. Letter from C. O. Thomas (NRC) to H. B. Tucker (DPC), March 13, 1985.

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

9. DPC-NE-2004, "Duke Power Company, McGuire and Catawba Nuclear Stations, Core Thermal-Hydarulic Methodology Using VIPRE-01," January 1989.

ATTACHMENT 9a

Detailed Listing of Changes to DPC-NE-1003-A

Attachment 9a - Detailed Listing of Changes to DPC-NE-1003A

This attachment provides a detailed list of proposed changes to the topical report DPC-NE-1003. Changes are listed according to the location in DPC-NE-1003A. Cited references are listed at the end of this attachment.

1. Cover, Table of Contents, List of Figures

   Description: Editorial changes and additions to correspond to changes associated with this revision.

2. Section 1

   Description: Revised this section to clarify the application of the rod swap process and to make the report consistent with current NRC approved methods (Reference 1).

3. Section 3, First Paragraph

   Description: Revised the third sentence to make the report consistent with current procedures.

4. Section 4, Item 4

   Description: Clarified the process for modeling the critical height.

   Justification: The original statement was applicable to NODE. This change is made to make the report consistent other methods approved by the NRC (Reference 1). SIMULATE is capable of performing an automated search to determine critical height.

5. Section 6, Item (b)

   Description: Clarified the acceptance criteria on the total rod worth to make the report consistent with the NRC SER (for the original version of this report) dated May 22, 1987.

6. Data Tables

   Description: Clarified terms and more coherently numbered the tables.

7. Section 7

   Description: Added Reference 3.

References:

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

ATTACHMENT 9b

DPC-NE-1003, Revision 1

Duke Power SM

A Duke Energy Company

ATTACHMENT 9b

DPC-NE-1003, Revision 1

McGuire Nuclear Station
Catawba Nuclear Station

Rod Swap Methodology Report for Startup Physics Testing

DPC-NE-1003
Revision 1

August 2001

Duke Power Company
Nuclear Generation Department
Nuclear Engineering

Revision History

Revsion	Description
DPC-NE-1003, Original Issue	Originally submitted to the NRC for approval in October 1986. Additional submittals were made to the NRC supplying additional data and reponses to requests for additional information.
DPC-NE-1003A, Original Issue	NRC approved version issued in May 1987.
DPC-NE-1003, Revision 1	Submitted to the NRC for approval in August 2001. This revsion updates the report for completeness to indicate the use of NRC approved methods approved subsequent to the implementation of the original issue including the use of CASMO-3/SIMLUATE-3 reactor physics methods. This revison also reflects a refinement in the rod swap to make use of two test banks. This revision also reflects the ability of the SIMULATE-3P computer code to iterate numerical results in the determination of the reference bank critical height. Finally, various editorial changes are made, such as reformatting tables and adding a Table of Contents, a List of Tables, and page numbers. Changes associated with this revision are denoted by revision bars, except for the editorial changes mentioned above.

Table of Contents

List of Tables

1. Introduction

This report describes the calculational procedure used to develop the rod swap constants and describes the measurement procedure used to determine the inferred bank worths. This paper also presents a comparison between the calculated and inferred bank worths for McGuire 1 Cycles 2, 3 and 4, and McGuire 2 Cycles 2 and 3.

In order to perform the "Control Rod Worth Measurement - Rod Swap Test Procedure" (2), the following information must be provided to the station. This information shall include the bank worths, critical heights and α's. The critical heights and α's are used to calculate the inferred bank worth of each control and shutdown bank, as reduced from information following the iso-reactivity interchange with the reference bank.

This report presents the calculated procedures used to derive these parameters. The calculations as performed in this procedure utilize the approved physics codes and methodologies described in References 1 and 3.

The rod swap procedure is one of the methods available for determining total rod worth and individual bank worths during zero power physics testing.

2. Definitions

The following is a list of the constants needed by the plant, to perform the rod swap procedure. These include:

- W^{P}_{x} - Predicted reactivity worth of each control and shutdown bank, when inserted individually into an otherwise unrodded core.

- h^{P}_{x} - Predicted critical position of the reference bank after interchange with bank x, starting with the reference bank at 0 steps and bank x fully withdrawn.

- α_{x} - A correction factor which accounts for the effect of bank x on the partial integral worth of the reference bank, equal to the ratio of the integral worth of the reference bank from h^{P}_{x} to the fully withdrawn position with and without x in the core.

In addition, included is a list of constants and their definitions as used in this report.

- W^{I}_{x} - Measured rod bank worth of bank x from rod exchange

- W^{m}_{Ref} - Measured rod bank worth of reference bank

- $(\Delta\rho)_{x}$ - The measured integral worth of the reference bank from the measured critical position (h^{m}_{x}) to the fully withdrawn position.

- h^{m}_{x} - The measured critical position of the reference bank after interchange with bank x.

3. Measurement Procedure

With an initial configuration of all rods out, hot zero power, the integral worth of the reference bank is measured using the standard boration/dilution technique. The reference bank is the bank that is predicted to have the highest integral worth. All other banks are then exchanged with the reference bank or other test banks at constant boron conditions until the measured bank is fully inserted.

The worth of each bank is then the amount of reactivity change caused by the withdrawal of the reference bank to its new critical height.

The rod bank worth is inferred from the measured reference bank worth and the measured reference bank height using the following equation:

$$W^I_X = W^M_{ref} - \alpha_x \ (\Delta\rho)_x$$

where the above terms are defined in Section 2.0 of this report.

4. Calculational Procedure

This calculation is performed using EPRI-NODE-P or SIMULATE-3P to model core conditions during the rod swap procedure. The following procedure describes the method of data generation:

1. Calculate the integral bank worth at HZP, ARO critical boron. Insert one bank at a time with no overlap and calculate the bank worth as the difference between ARO and the bank fully inserted condition. (The calculated highest worth bank will be considered the reference bank.)

2. With the reference bank fully inserted, calculate the critical boron concentration. (The reference bank in boron concentration is used in predicting the predicted rod worth - $W^P{}_x$).

3. Using the above calculated critical boron concentration for the reference bank, the new integral bank worths at HZP are determined. These values correspond to the predicted worth for each bank ($W^P{}_x$).

   The reference bank should be inserted in approximately six (6) step increments such that a plot of the integral worth of the reference bank can be obtained. (As should be noted, the K_{eff} with the reference bank inserted, is referred to as the base K_{eff}).

4. In order to calculate the critical height, the core is modeled with the measured bank fully inserted. The critical height ($h^P{}_x$) of the reference bank is then determined by adjusting the reference bank position until the K_{eff} matches the base K_{eff}.

5. In order to calculate α for each bank position, the following expression is used:

$$\alpha = \frac{\text{Integral Worth of the reference bank from } h^P{}_x \text{ to the fully withdrawn position } \underline{\text{with}} \text{ bank x inserted in the core}}{\text{Integral worth of the reference bank from } h^P{}_x \text{ to the fully withdrawn position } \underline{\text{without}} \text{ bank x inserted in the core}}$$

5. Results

Tables 1 and 2 present a comparison between Duke's predicted and inferred bank worths. A review of the available data from McGuire 1 Cycles 2, 3, and 4, and McGuire 2 Cycles 2 and 3, identifies a mean difference of 5.27 pcm or 0.66% between Duke's predicted and inferred bank worths.

Tables 3 and 4 identify a comparison between measured and predicted total critical heights. The standard deviation of the differences between the measured critical heights and Duke's calculated critical heights is 12.63.

Table 5 presents some typical α values as calculated for McGuire 1, Cycle 3.

Additional benchmarking of predicted and measured rod worth data using SIMULATE-3P can be found in Section 3.2 of Reference 3.

6. Conclusion

Reference to the Rod Swap Test Procedure (2) identifies the specific acceptance criteria. In order to satisfy this procedure the following conditions must be met:

(a) The absolute value of the percent difference between the measured and predicted integral worth for the reference bank is $\leq$ 15%.

(b) The sum of the measured/inferred worth of all the rods must be > 90% of the predicted rod worth.

(c) For all RCC banks other than the reference bank, either:

(i) the percent difference between the inferred and predicted worth for each individual bank is $\leq$ 30%

<u>or</u>

(ii) $|W^I_x - W^P_x| \leq$ 200 pcm for each bank,

whichever is greater.

These criteria were found acceptable using Duke's predicted values.

Based on the predicted and measured data presented in this report the rod swap method described has been verified to be accurate for use in startup physics testing.

Table 1

Duke Predicted and Inferred Bank Worth

Unit/Cycle	Bank	Duke Predicted Worth (PCM)	Duke Inferred Worth (PCM)	Difference (PCM)	Difference (%)
1/2	CA	289	301	-12	-4.0
	CB	557	606	-49	-8.1
	CC	786	788	-2	0.3
	CD	616	566	50	8.8
	SA	473	546	-73	-13.4
	SB	443	479	-36	-7.5
	SC	370	354	16	4.5
	SD	362	374	-12	-3.2
	SE	223	237	-14	-5.9
Total		4119	4251	-132	-3.1
Mean		-	-	-14.67	-3.17
Standard Deviation		-	-	35.94	6.80

Difference (PCM) = Predicted - Inferred

$$\text{Difference (\%)} = \frac{W^P - W^I}{W^I} \times 100$$

Table 1 (Cont.)

Duke Predicted and Inferred Bank Worth

Unit/Cycle	Bank	Duke Predicted Worth (PCM)	Duke Inferred Worth (PCM)	Difference (PCM)	Difference (%)
1/3	CA	311	305	6	2.0
	CB	657	609	48	7.9
	CC	789	745	44	5.9
	CD	488	466	22	4.7
	SA	269	303	-34	-11.2
	SB	856	779	77	9.9
	SC	394	373	21	5.6
	SD	395	383	12	3.1
	SE	429	392	37	9.4
Total		4588	4355	233	5.4
Mean		-	-	25.89	4.14
Standard Deviation		-	-	31.16	6.34

Difference (PCM) = Predicted - Inferred

$$\text{Difference (\%)} = \frac{W^P - W^I}{W^I} \times 100$$

Table 1 (Cont.)

Duke Predicted and Inferred Bank Worth

Unit/Cycle	Bank	Duke Predicted Worth (PCM)	Duke Inferred Worth (PCM)	Difference (PCM)	Difference (%)
1/4	CA	301	313	-12	-3.8
	CB	656	677	-21	-3.1
	CC	775	778	-3	-0.4
	CD	581	556	25	4.5
	SA	293	307	-14	-4.6
	SB	746	750	-4	-0.5
	SC	381	377	4	1.1
	SD	382	314	68	21.7
	SE	473	471	2	0.4
Total		4588	4543	45	1.0
Mean		-	-	5	1.7
Standard Deviation		-	-	27.04	8.0

Difference (PCM) = Predicted - Inferred

$$\text{Difference (\%)} = \frac{W^P - W^I}{W^I} \times 100$$

Table 1 (Cont.)

Duke Predicted and Inferred Bank Worth

Unit/Cycle	Bank	Duke Predicted Worth (PCM)	Duke Inferred Worth (PCM)	Difference (PCM)	Difference (%)
2/2	CA	437	459	-22	-4.8
	CB	413	452	-39	-8.6
	CC	858	871	-13	-1.5
	CD	654	664	-10	-1.5
	SA	327	430	-103	-24.0
	SB	425	480	-55	-11.5
	SC	354	375	-21	-5.6
	SD	355	374	-19	-5.1
	SE	270	292	-22	-7.5
Total		4093	4397	-304	-6.9
Mean		-	-	-33.78	-7.79
Standard Deviation		-	-	29.42	6.87

Difference (PCM) = Predicted - Inferred

$$\text{Difference (\%)} = \frac{W^P - W^I}{W^I} \times 100$$

Table 1 (Cont.)

Duke Predicted and Inferred Bank Worth

Unit/Cycle	Bank	Duke Predicted Worth (PCM)	Duke Inferred Worth (PCM)	Difference (PCM)	Difference (%)
2/3	CA	344	314	30	9.6
	CB	698	668	30	4.5
***	CC	869	787	82	10.4
	CD	591	530	61	11.5
	SA	381	404	-23	-5.7
	SB	906	842	64	7.6
	SC	438	378	60	15.9
	SD	440	406	34	8.4
	SE	481	424	57	13.4
Total		5148	4753	395	8.3
Mean		-	-	43.89	8.40
Standard Deviation		-	-	30.70	6.23

*** This was the reference bank used because vendor supplied data was used for the official rod swap calculation.

Difference (PCM) = Predicted - Inferred

$$\text{Difference (\%)} = \frac{W^P - W^I}{W^I} \times 100$$

Table 2

Summary of Duke Predicted and Inferred Bank Worth

	Duke Calculated Difference (PCM)	Difference (%)
Mean	5.27	.66
Standard Deviation	40.72	8.69

Difference (PCM) = Predicted - Inferred

$$\text{Difference (\%)} = \frac{W^P - W^I}{W^I} \times 100$$

Table 3

Duke Predicted and Measured Critical Heights

Unit/Cycle	Bank	Critical Height (Steps) Measured	Predicted	Difference (Steps)
1/2	CA	83	88	-5
	CB	197	195	2
	CD	183	196	-13
	SA	191	187	4
	SB	156	157	-1
	SC	144	158	-14
	SD	147	156	-9
	SE	86	92	-6
Σ		-	-	-42
Σ of Absolute Value		-	-	54
Standard Deviation		-	-	6.63

Difference (Steps) = Measured - Predicted

Table 3 (Cont.)

Duke Predicted and Measured Critical Heights

Unit/Cycle	Bank	Critical Height (Steps) Measured	Critical Height (Steps) Predicted	Difference (Steps)
1/3	CA	127	117	10
	CB	180	172	8
	CC	224	201	23
	CD	163	156	7
	SA	127	111	16
	SC	139	133	6
	SD	141	133	8
	SE	132	126	6
Σ		-	-	84
Σ of Absolute Value		-	-	84
Standard Deviation		-	-	6.00

Difference (Steps) = Measured - Predicted

Table 3 (Cont.)

Duke Predicted and Measured Critical Heights

Unit/Cycle	Bank	Critical Height (Steps) Measured	Critical Height (Steps) Predicted	Difference (Steps)
1/4	CA	108	121	-13
	CB	201	203	-2
	CD	179	191	-12
	SA	136	149	-13
	SB	218	216	2
	SC	147	161	-14
	SD	136	161	-25
	SE	151	163	-12
Σ		-	-	-89
Σ of Absolute Value		-	-	93
Standard Deviation		-	-	8.15

Difference (Steps) = Measured - Predicted

Table 3 (Cont.)

Duke Predicted and Measured Critical Heights

Unit/Cycle	Bank	Critical Height (Steps) Measured	Predicted	Difference (Steps)
2/2	CA	153	146	7
	CB	190	191	-1
	CD	202	205	-3
	SA	198	186	12
	SB	194	183	11
	SC	185	182	3
	SD	184	182	2
	SE	149	141	8
Σ		-	-	39
Σ of Absolute Value		-	-	47
Standard Deviation		-	-	5.49

Difference (Steps) = Measured - Predicted

Table 3 (Cont.)

Duke Predicted and Measured Critical Heights

Unit/Cycle	Bank	Critical Height (Steps) Measured	Predicted	Difference (Steps)
2/3	CA	99	112	-13
	CB	173	191	-18
	CD	158	179	-21
	SA	123	145	-22
	SB	228	228	0
	SC	130	159	-29
	SD	131	159	-28
	SE	131	147	-16
Σ		-	-	-147
Σ of Absolute Value		-	-	147
Standard Deviation		-	-	9.24

Difference (Steps) = Measured - Predicted

Table 4

Summary of Duke Predicted and Measured Critical Heights

	Duke Calculated
Σ (Differences)	-155
Σ (Absolute Value of Differences)	425
Standard Deviation (of the Differences)	12.63

Difference (Steps) = Measured - Predicted

Table 5

α Factors

Unit/Cycle	Bank	Calculated
1/3	CA	1.042
	CB	0.877
	CC	0.870
	CD	1.161
	SA	1.060
	SC	1.052
	SD	1.050
	SE	0.903

7. References

1. Duke Power Company, "Nuclear Physics Methodology for Reload Design", DPC-NF-2010A, June 1985.

2. Duke Power Company, McGuire Nuclear Station, "Control Rod Worth Measurement: Rod Swap Test Procedure", PT/0/A/4150/11A, April 1984.

3. Duke Power Company, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P", DPC-NE-1004A, Revision 1, SER Dated April 26, 1997.

APPENDIX A

NRC/DPC Correspondence Including DPC Responses to NRC Requests for Additional Information

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

February 11, 1987

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

Subject: McGuire Nuclear Station
Docket Nos. 50-369/370
Catawba Nuclear Station
Docket Nos. 50-413/414
Determination of Rod Worth Using
Rod Swap Methodology

Gentlemen:

By letter dated December 4, 1986, Duke submitted for information to NRC a description of the method by which bank worths are determined in startup physics testing. By letter of January 12, 1987, the Staff responded to the submittal with a request for additional information. Attached are the responses to the Staff's questions.

It is intended that the methodology described in the December 4, 1986 submittal will be used for the next reloads of Duke's Westinghouse plants; the first of which is scheduled for May 1, 1987.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

SAG/54/jgm

Attachment

xc: Mr. Darl Hood, Project Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. J. Nelson Grace, Regional Administrator
U.S. Nuclear Regulatory Commission - Region II
101 Marietta Street NW - Suite 2900
Atlanta, GA 30323

Mr. W.T. Orders
NRC Resident Inspector
McGuire Nuclear Station

bxc: <u>w/o attachment</u>
R.H. Clark
M.S. Kitlan
E.O. McCraw
R. Van Namen
N.A. Rutherford
R.L. Gill
MC-801.02
(7)

QUESTION 1: Are all the rod worth calculations done with the EPRI-NODE-P Code, including both rod swap and rod worth for shutdown margin?

RESPONSE: Shutdown Margin calculations are performed according to the methodology approved in DPC-NF-2010A. Rod worths for both the shutdown margin calculation and the rod swap calculations are done using EPRI-NODE-P.

NOTE: See Section 5.4 of DPC-NF-2010A for the procedure for shutdown margin calculations.

QUESTION 2: Section 3, "Measurement Procedure": submit detailed procedures for the measurements. Include the actual boron dilution rate and the flux level for each of the tests included in the report.

RESPONSE: The most current procedures used in the rod swap measurements are enclosed as Attachments 1, 2, and 3.

A summary of the reactivity insertion rates and flux levels for each of the tests in the reference is presented below. Flux levels are values as measured on the reactivity computer picoammeter.

UNIT/CYCLE	REACTIVITY INSERTION RATE (PCM/HR)	TEST RANGE (AMPS)	POINT OF ADDING NUCLEAR HEAT (AMPS)
M1C2	450	1 E-8 TO 1 E-7	1.4 E-6
M1C3	460	1 E-8 TO 1 E-7	4.25 E-7
M1C4	420	1 E-8 TO 1 E-7	5.1 E-7
M2C2	480	1 E-7 TO 1 E-6	1.6 E-6
M2C3	720	1 E-7 TO 1 E-6	1.65 E-6

QUESTION 3: Section 4, "Calculational Procedure" - under 5: How many calculations are performed for each bank and at what positions.

RESPONSE: One α is calculated for each bank (except for the reference bank) at the predicted critical height. These calculations use the results of cases performed for Sections 4.3 and 4.4 of the reference. Cases are done with the reference bank being inserted in approximately 6-step increments both by itself and in the presence of the bank being predicted.

QUESTION 4: Table 3, "α's": Are the values given at the predicted heights?

RESPONSE: Alpha (α) is the ratio of the reference bank worth from the predicted critical height to out of the core with and without bank X in the core. Values for are given at the predicted critical heights. However, the ratio of the reference bank worth with and without bank X in the core is insensitive to variations in the predicted critical heights and will have no significant impact on the inferred worth.

QUESTION 5: Submit a copy of Reference 2.

RESPONSE: Reference 2: Duke Power Company McGuire Nuclear Station, "Control Rod Worth Measurement: Rod Swap Test Procedure," PT/0/A/4150/11A, April, 1984 test procedure is enclosed as Attachment 4.

QUESTION 6: Provide data for at least 2 sets of side-by-side comparisons of boron dilution and rod swap data - predicted and measured. The data may be either for your plants or measured data from another plant and predictions by Duke.

RESPONSE: Table with requested data is provided below. All rod worths are given in units of PCM.

UNIT/ CYCLE	BANK	PREDICTED WORTH	BOR/DILUTION MEAS WORTH	% DIFF ((P-M)/M)*100)	ROD SWAP INF WORTH	% DIFF ((P-I)/I)*100
M1C2	CD	616	566	8.8	586	5.1
M1C3	CD	488	483	1.0	466	4.7
M1C4	CD	581	580	0.2	556	4.5
M2C2	CD	654	665	-1.7	664	-1.5
M2C3	CD	591	556	6.3	530	11.5
MEAN				2.9		4.9
STANDARD DEVIATION				4.4		4.6

QUESTION 7: What Organization does the safety analysis for the Duke Plants? When this is not done by Duke, what is done (e.g. tests, comparisons, etc.) to show that the startup test results adequately represent the plant features and assumptions used in the safety analyses?

RESPONSE: Cycle specific safety reviews and any safety re-analyses required for McGuire and Catawba are performed by Westinghouse, the current fuel vendor. Assuming all startup tests meet acceptance criteria, transmittal of the results to Westinghouse is formally accomplished by providing them a copy of the startup report prepared for the NRC. If any review or acceptance criteria are exceeded, the the action statements in the procedure are followed. Actions required usually include review of the test data and predicted values, assessment of impacts on safety analyses and technical specification limits, etc. Groups involved in these reviews include the Site Reactor Group, the General Office Nuclear Design Group and, as necessary, Site Compliance, G.O. Licensing, G.O. Safety Analysis, and Westinghouse.

The main safety analysis assumption verified by the rod swap procedure is that the plant will maintain adequate shutdown margin per technical specifications. One of the purposes of rod swap measurements and comparisons is to verify the accuracy of the total rod worth prediction used as an input to the shutdown margin calculation. An independent Duke Power shutdown margin is evaluated for each cycle using methods approved by the NRC in DPC-NF-2010A. The N-1 rod worth used in this prediction is reduced by 10% for conservatism. Acceptance criteria listed in the procedure indicate that the total inferred rod worth as measured in the rod swap testing must be within 10% of the total predicted worth. If the total measured rod worth is less than the predicted worth by more than 10%, a review of the shutdown margin is made to determine if the current rod insertion limits provide adequate shutdown margin. If the shutdown margin is adequate, then no revision of the limits is necessary. However, if the margin is not maintained, then Duke will notify Westinghouse, revise the rod insertion limits, and submit any necessary changes in the technical specifications to the NRC.

Reference

McGuire Nuclear Station, Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing, DPC-NE-1003, Rev. 1, December 1986.

QUESTION 2

Attachment 1

Form 34731 (R9-86)

(1) ID No. PT/0/A/4150/21

Change(s) 0 to 31 Incorporated

FOR INFORMATION

PREPARATION

(2) STATION McGUIRE

(3) PROCEDURE TITLE POST REFUELING CONTROLLING PROCEDURE FOR CRITICALITY, ZERO POWER PHYSICS, AND POWER ESCALATION TESTING

(4) PREPARED BY Charles W. Roberson DATE 1/6/87

(5) REVIEWED BY M. Todd Keller DATE 1/6/87

Cross-Disciplinary Review By ________ N/R MTK

(6) TEMPORARY APPROVAL (If Necessary)

By ________ (SRO) DATE

By ________ DATE

(7) APPROVED BY Bruce Hamilton DATE 1/7/87

(8) MISCELLANEOUS

Reviewed/Approved By ________ DATE ________

Reviewed/Approved By ________ DATE ________

(9) COMMENTS (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.) ADDITIONAL CHANGES INCLUDED. ☐ Yes ☒ No

(10) COMPARED WITH CONTROL COPY TW Miracia DATE 1/29/87

COMPLETION

(11) DATE(S) PERFORMED ________

(12) PROCEDURE COMPLETION VERIFICATION

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Acceptance criteria met?

VERIFIED BY ________ DATE ________

(13) PROCEDURE COMPLETION APPROVED ________ DATE ________

(14) REMARKS (Attach additional pages, if necessary.)

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION
POST REFUELING CONTROLLING PROCEDURE FOR CRITICALITY, ZERO POWER PHYSICS, AND POWER ESCALATION TESTING

1.0 Purpose

1.1 To provide a sequence of tests for the orderly startup of the unit after refueling.

1.2 To perform nuclear instrumentation overlap verification.

1.3 To determine the point of nuclear heat.

1.4 To establish the neutron flux levels corresponding to the Zero Power Physics Test Band.

1.5 To perform a checkout of the reactivity computer.

2.0 References

2.1 McGuire Nuclear Station Technical Specifications

2.2 WCAP-9648, Post-Refueling Nuclear Testing Program Criticality to Full Power.

2.3 The appropriate unit and cycle Nuclear Design Report.

3.0 Time Required

5 days, 2 engineers per shift - 3 shifts

4.0 Prerequisite Tests

Initial/Date

____/____ 4.1 PT/0/A/4600/14B, NIS Intermediate Range Calibration Functional Test (see Step 7.4).

____/____ 4.2 PT/0/A/4600/14A, NIS Power Range Calibration Functional Test (see Step 7.5)

NOTE: The tests in 4.1 and 4.2 must be completed within 12 hours prior to beginning Physics Testing. Physics testing is defined as beginning when Control Rods are being withdrawn to achieve criticality. This occurs in Step 12.9 of PT/0/A/4150/28, Criticality Following a Change in Core Nuclear Characteristics.

5.0 Test Equipment

5.1 Reactivity Computer connected to one power range detector (Enclosure 13.6) (See Step 8.2 for installation step.)

5.2 Chart recorders to display reactivity, flux, pressurizer level, and T_{avg}.

5.3 Stopwatch or timer

5.4 Communications between Control Room operators and testing work station.

6.0 Limits and Precautions

6.1 The startup rate is administratively limited to 0.5 DPM.

6.2 During the Zero Power Physics Tests (Steps 12.3 - 12.10.20) Special Test Exception 3.10.3 will be invoked. The appropriate Surveillance Requirements will be monitored by Operations.

6.3 Notify Westinghouse if any incore tilts exceed 2%.

6.4 The primary indication of core power will be ΔT, which should be cross checked with the NIS and the Thermal Power calculation on the OAC. If the thermal power and Power Range NIS disagree by more than 2%, then adjustment is necessary per Tech Spec 3/4.3.1, Table 4.3-1, notation 2. (IP/0/A/3007/17)

6.5 If the excore power indications are conservative, use caution when increasing power to avoid the high level trip setpoints.

6.6 Observe the Fuel Maneuvering Limits as outlined in Data Book Section 1.3.

7.0 Required Unit Status

Initial/Date

____/____ 7.1 The unit is in Mode 3 - Hot Standby

____/____ 7.2 The points listed on Enclosure 13.1 are being logged on OAC Gen. 24 program once per 6 minutes printed every 8 hours.

____/____ 7.3 Record the unit and cycle to which this procedure is being applied, in the test log.

8.0 Prerequisite System Conditions

____/____ 8.1 All RCC control banks and shutdown banks are fully inserted.

____/____ 8.2 Begin to install the reactivity computer per Enclosure 13.6. The reactivity computer shall be installed before beginning Step 12.4.

____/____ 8.3 An evaluation of the impact of the core alterations on the excore detector sensitivity has been made. Document the results in the test log. Attach to this procedure any correspondence from offsite personnel on this subject.

____/____ 8.4 Perform Enclosure 13.10 to demonstrate adequate Shutdown Margin at the zero power insertion limits per Tech Spec 4.1.1.1.1d.

____/____ 8.5 Perform Enclosure 13.9 to verify adequate Shutdown Margin during Rod Swap.

____/____ 8.6 Provide I&E 7300 Systems Engineer with the new cycle 100% F.P. predicted value of Reactor Vessel Tave.

____/____ 8.7 I&E 7300 Systems Engineers have set ΔT values to conservative numbers as necessary in the protection cabinets. Record in the test log the values which have been set in the cabinets.

9.0 Test Method

The reactor is brought critical with the procedure for criticality. Then, the Intermediate Range (I/R) NIS overlap data is recorded, the point-of-nuclear-heat flux level determined, and the Zero Power Physics Test (ZPPT) band is established. Also, the reactivity computer is verified to be set up correctly by making reactivity changes and comparing the computer response to the calculated reactor period.

Next, the ZPPT's are performed to measure the ARO boron concentration, control rod worths, moderator temperature coefficients, and the low-power core power distribution (if necessary).

Finally, power escalation is begun, with a full core flux map between 10% and 50% full power. During the escalation above 50% full power, data is taken for the Power Range NIS calibrations. At ~80% full power, the P/R NIS is calibrated, then power is increased 100% full power. At 100% full power, the core power distribution, the NIS calibration, the thermal power output program, and the reactivity anomolies are all checked. Also, the target flux difference is measured, and Reactor Coolant System Flow Test is performed.

10.0 Data Required

10.1 Nuclear instrumentation overlap will be recorded on Enclosure 13.2.

10.2 The point of nuclear heat will be recorded on Enclosure 13.3.

10.3 The reactivity computer checkout results will be recorded on Enclosure 13.4.

10.4 Output of OAC Gen. 24 program per Enclosure 13.1.

10.5 Intermediate range high level trip setpoints on Enclosure 13.7.

10.6 Verification of adequate Shutdown Margin at the zero power insertion limits on Enclosure 13.10.

10.7 Verification of Shutdown Margin during Rod Swap on Enclosure 13.9.

11.0 Acceptance Criteria

11.1 There is at least one decade overlap on the NIS between the Source and Intermediate Ranges, and between the Intermediate and Power Ranges (NOTE: Power Ranges are calibrated to Thermal Power, Best Est. (P1385). Use P1385 for Power Range overlap data).

11.2 The value of the reactivity measured by the reactivity computer is within .04 (4%) or 1 PCM, whichever is greater, of the reactivity inferred from the reactor period, or doubling time.

$$\left|\frac{\Delta\rho_c - \Delta\rho_{DT}}{\Delta\rho_{DT}}\right| \leq .04\ (4\%) \text{ or } 1 \text{ pcm}$$

11.3 All acceptance criteria in each test procedure for the tests contained in this controlling procedure have been met.

12.0 Procedure

Initial/Date

____/____ 12.1 Attach as Page 2 of Enclosure 13.4 the table of "reactivity and doubling time as a function of stable reactor period at BOL, HZP conditions" for the appropriate unit and cycle. Also attach as Page 3 of Enclosure 13.4 the curve (if provided) "Reactor Period and Doubling Time as a Function of Reactivity at BOL, HZP, No Xenon" for the appropriate unit and cycle.

____/____ 12.2 Inform the Operations Shift Supervisor that Special Test Exception Tech Spec 3.10.3 will be entered during criticality and Zero Power Physics Testing (Steps 12.3 - 12.10). Operations shall monitor the appropriate Surveillance Requirements during these Steps.

____/____ 12.3 Complete PT/0/A/4150/28, Criticality Following a Change in Core Nuclear Characteristics. It is permissible to sign off this step prior to signing off Steps 12.18 and 12.19 in PT/0/A/4150/28.

<u>NOTE</u>: Section 7.0 of this procedure will have been completed earlier.

<u>NOTE</u>: See Step 4.1 and 4.2.

____/____ 12.4 Begin PT/0/B/4600/55, Reactivity Computer Periodic Test approximately 4-6 hours prior to Step 12.6.

____/____ 12.5 Record the IR high level trip setpoints on Enclosure 13.7.

____/____ 12.6 With a Source Range reading of $\cong 10^3$ cps and the reactor just critical withdraw Control Bank D or add demineralized water, to establish a slow positive startup rate (<50 pcm). When the Intermediate Range indication comes on scale, halt the flux level increase, establish just critical conditions, and record data as required by Enclosure 13.2, Page 1 of 2.

____/____ 12.7 Continue to increase the flux level, stopping, establishing just critical conditions, and recording data with each decade increase in the Intermediate Range until the Source Range is blocked.

<u>CAUTION</u>: Do not exceed 10^5 cps on the Source Range unless the Source Range is blocked, as a reactor trip will occur.

CAUTION: I/R high level trip setpoints are on Enclosure 13.7; do not exceed these values.

___/___ 12.7.1 Verify from Enclosure 13.2 Page 1 of 2 that a minimum of one full decade of overlap exists between the Source Range and Intermediate Range before the Source Range reaches 10^5 cps.

12.8 Determine the flux level at which the point of nuclear heat occurs by the following steps.

___/___ 12.8.1 Set up 1, 2 pen strip chart recorder with T_{avg} and reactivity, another 2 pen strip chart recorder with pressurizer level and flux signal.

___/___ 12.8.2 Establish just critical conditions with reactivity computer picoammeter reading of about 1×10^{-8} amps. Adjust the scale setting on the reactivity computer picoammeter (if necessary) such that the indicator is on scale and indicating a value near the low end of the scale. Record start values on Enclosure 13.3.

NOTE: Stop increase if nuclear heat is observed prior to reaching this level, and repeat Step 12.5.2 from 1×10^{-9} amps on the reactivity computer picoammeter.

___/___ 12.8.3 Establish a slow positive startup rate by rod withdrawal of about 20 pcm and allow the flux level to increase until nuclear heat is observed. At this time, re-establish just critical conditions by Control Bank D adjustment. Record Nuclear Heat Data on Enclosure 13.3.

NOTE: Nuclear heat can be best observed as an increase T_{avg} accompanied by a change in the reactivity trace and an increase in pressurizer level.

NOTE: It is permissible to also trend pressurizer level, Intermediate Range Level, and NC Loop Highest Average Temperature on the OAC to aid in the determination of nuclear heat.

___/___ 12.8.4 Repeat Steps 12.8.2 and 12.8.3 a second time and record all data as requested on Enclosure 13.3.

___/___ 12.8.5 Determine the Zero Power Physics Testing Range from the reactivity computer picoammeter flux levels on Enclosure 13.3. Record on Enclosure 13.3.

NOTE: The range for all Zero Power Physics Testing will be defined as the next lowest whole decade such that the upper end of the decade is not within $\sqrt{10}$ of nuclear heat.

EXAMPLE: If nuclear heat is found at 5×10^{-6} amps on the picoammeter then

$$\frac{5 \times 10^{-6}}{\sqrt{10}} = 1.5 \times 10^{-6} \text{ and}$$

the range for zero power testing is 1.0×10^{-7} to 1.0×10^{-6} amps.

NOTE: If the signal is not clear for the decade defined, evaluate the situation and if changes are needed to be made to the testing decade, fully document in the test log the reason for the change before continuing.

___/___ 12.8.6 Insert Control Bank D slightly, allow the flux to decrease until the reactivity computer picoammeter reads near the low end within the Zero Power Physics Test range determined above, and level out again.

12.9 Perform a checkout of the reactivity computer.

___/___ 12.9.1 Withdraw Control Bank D until a reactivity gain of approximately +25 pcm is indicated by the reactivity computer.

___/___ 12.9.2 Let the flux increase to a stable period and measure the doubling time at two or three different times over the decade using a stopwatch or timer. From the doubling time, calculate the period from the following equation and record on Enclosure 13.4, page 1:

$$\text{period} = \frac{DT}{0.693}$$

___/___ 12.9.3 Using the table on Page 2 of Enclosure 13.4, or the curve (if provided) on Page 3 of Enclosure 13.4, convert the observed period to reactivity and record on page 1 of Enclosure 13.4.

___/___ 12.9.4 Record all data on Enclosure 13.4.

___/___ 12.9.5 Repeat measurement as needed until at least three checks have been performed.

___/___ 12.9.6 Repeat Steps 12.9.1 through 12.9.4 for a reactivity addition of +50 pcm.

___/___ 12.9.7 Repeat measurement as needed until at least three checks have been performed.

___/___ 12.9.8 Verify the Acceptance Criteria of 11.2 has been met for the positive reactivity insertions only.

___/___ 12.9.9 Verify a negative reactivity insertion check has been performed satisfactorily on the reactivity computer per PT/0/B/4600/55, Reactivity Computer Periodic Test.

___/___ 12.9.10 Position Control Bank D at ≅220 steps by boration or dilution.

___/___ 12.10 Zero Power Physics Testing

Complete the tests listed below. Normal operating procedures shall be used to reconfigure the plant to meet any prerequisites. All tests should be performed within the test band established in Step 12.8.5, except power will be increased up to ≅3-4% full power for the low power flux map if it is taken.

___/___ 12.10.1 Perform PT/0/A/4150/10, Boron Endpoint Measurement.

___/___ 12.10.2 Perform PT/0/A/4150/12, Isothermal Temperature Coefficient Measurement for the ARO case.

___/___ 12.10.3 Perform PT/0/A/4150/31, Determination of Rod Withdrawal Limits to Ensure Moderator Temperatures Within Limits of Technical Specifications. Testing may continue under Special Test Exception Tech Spec 3.10.3; however, PT/0/A/4150/31 Section 12.1 must be performed prior to the completion of data gathering for the Rod Swap test of Step 12.10.5. If the MTC calculated in Step 12.10.2 is less than 0 pcm/°F, mark this step N/A.

___/___ 12.10.4 Record on Enclosure 13.8 the Reference Bank, rod banks, and sequence to be measured by rod swap.

<u>NOTE</u>: If the predicted worth of any bank is close to the predicted worth of the reference bank, measure this bank last.

___/___ 12.10.5 Perform PT/0/A/4150/11A, Control Rod Worth Measurement - Rod Swap. This measurement is to be done for the rod banks identified on Enclosure 13.8.

___/___ 12.10.6 Following Rod Swap Measurements swap Control Bank D with the reference bank until Bank D is fully inserted.

___/___ 12.10.7 If Section 12.1 of PT/0/A/4150/31, Determination of Rod Withdrawal Limits procedure indicates no rod withdrawal limits are needed mark Step 12.10.8, 12.10.9, and 12.10.11 as N/A and continue. If the indication is that rod withdrawal limits will be needed, perform Steps 12.10.8, 12.10.9 and 12.10.11.

<u>NOTE</u>: It is permissible to perform Steps 12.10.8 and 12.10.9 if desired even though it might not be required. In that case, N/A Step 12.10.11.

___/___ 12.10.8 Place the rods close to a D-in only configuration by borating the reference bank out.

___/___ 12.10.9 Perform PT/0/A/4150/12 Isothermal Temperature Coefficient Measurement for the D-in case.

___/___ 12.10.10 Perform PT/0/A/4150/11 Control Rod Worth Measurement. This measurement is to be done only for Control D as it is completely withdrawn by boration.

____/____ 12.10.11 Perform Section 12.2 of PT/0/A/4150/31, Determintaion of Rod Withdrawal Limits to Ensure Moderator Temperature Coefficient within Limits of Technical Specifications.

____/____ 12.10.12 Perform the following steps to reset bank overlap once Control Bank D is about 215 steps withdrawn.

____/____ 12.10.12.1 Go to the Master Cycler Cabinet and reset the Bank Overlap Digital Counter to 000 by pushing the reset button.

____/____ 12.10.12.2 Reset the Bank Overlap Counter to 345 plus the present Control Bank D position by pushing the button to count up from 000 to the desired value (one push of the button is one digit change on the display).

NOTE: Perform Steps 12.10.13 and 12.10.14 in any order or concurrently.

____/____ 12.10.13 Increase reactor power by dilution or Control D withdrawal so that both approximately 3-4% full power and Control D about 215 steps withdrawn are achieved.

NOTE: Control D may be placed in a configuration for power increase if Step 12.10.17 is to be marked N/A.

____/____ 12.10.14 Remove reactivity computer from the Power Range NIS Channel to which it is connected and return the Channel to OPERABLE status using Enclosure 13.6.

____/____ 12.10.15 Verify that Thermal Power, Best Est. reasonably agrees with the indicated loop ΔT's. Resolve any problems.

NOTE: Thermal Power should be approximately: [(loop avg $\Delta T(^\circ F) \cdot (\frac{75\%}{45^\circ F})$)], between 0-75% full power.

____/____ 12.10.16 Verify all power range channels are operable.

CAUTION: Do not continue until Step 12.10.16 is completed.

____/____ 12.10.17 Perform PT/0/A/4150/02A, Core Power Distribution if any rod swap acceptance criteria were not met in PT/0/A/4150/11A. Mark N/A here and also Step 12.10.19 if all criteria were met.

NOTE: It is permissible to perform Step 12.10.17 in any case if desired. In that case do not mark Step 12.10.19 as N/A.

____/____ 12.10.18 Record the Intermediate Range NIS overlap data at 3-4% full power on Enclosure 13.2.

____/____ 12.10.19 Perform PT/0/A/4150/23, Quarter-Core Flux Map Qualification Test.

NOTE: Testing may continue here; however, PT/0/A/4150/23, if performed now, must be complete prior to starting Step 12.11.7.

____/____ 12.10.20 Place Control Bank D at ~160 to 180 steps withdrawn to have sufficient reactivity to put the turbine on line.

____/____ 12.10.21 Verify the following:

____/____ 12.10.21.1 Acceptance criteria for each Zero Power Physics Test performed was met or any discrepancies have been resolved.

____/____ 12.10.21.2 All shutdown banks completely withdrawn and within $\pm$ 12 steps of group step counter demand position.

____/____ 12.10.21.3 Control banks above insertion limits and within $\pm$ 12 steps of group step counter demand position.

____/____ 12.10.21.4 Verify that the rod withdrawal limits are in place if they were required.

____/____ 12.10.21.5 Verify NC lowest operating loop Tave $\geq 551^{o}$ F.

____/____ 12.10.22 Inform the Operations Shift Supervisor that Special Test Exception Tech Spec 3.10.3 is being left. Appropriate surveillance can be stopped. Enclosure 13.1 data trending can be discontinued.

NOTE: Do not exceed 5% full power prior to completing steps 12.10.21 and 12.21.22.

____/____ 12.10.23 Review Data Book curves 6.1 and 6.3A and reissue these as needed to reflect actual measured data.

12.11 Power Escalation Testing

___/___ 12.11.1 Reset Power Range high level trip setpoints to
___/___IV 109% F.P. This step need not be completed prior to going on-line, only before ~20% F.P.

NOTE: Prior to putting the turbine on-line, verify Control D bank at ~160 to 180 steps. This will ensure the availability of reactivity which will be needed while placing the turbine on-line. Make sure that Control D bank is returned to a position >200 steps before reaching 20% F.P. per Data Book Section 1.3.

___/___ 12.11.2 Verify the Power Range High Level Trip Setpoints are set to 109% full power and inform the Control Room operator of that fact. This step need not be completed prior to going on-line, only before ~20% F.P.

___/___ 12.11.3 Between 10% and 50% F.P., perform PT/0/A/4150/02A, Core Power Distribution. (It is suggested to perform this at the 30% F.P. hold for Chemistry.)

NOTE: Equilibrium xenon is not necessary for this flux map. Boron samples may be waived also.

___/___ 12.11.4 Following the flux map, perform PT/0/A/4150/23 Quarter Core Flux Map Qualfication Test. This Step can be marked N/A if it was performed in Step 12.10.19.

___/___ 12.11.5 Begin increasing reactor power from 3-4% to 50% full power at a rate of approximately 2.5% per hour (not to exceed 3% per hour). See Limit and Precaution 6.6.

NOTE: A suggested sequence for power increase is to increase load at 1 MWe/min for 30 minutes then hold for the remainder of the hour.

___/___ 12.11.5.1 As power is increased and the unit goes on-line, check all inputs to the Thermal Power Calculation by using OAC program Nuclear 28 (Thermal Power Outputs Dump). Resolve all problems prior to the 50% full power plateau.

___/___ 12.11.5.2 Record the Intermediate Range NIS overlap data at 10%, 20% and 25% full power on Enclosure 13.2.

___/___ 12.11.5.2.1 Complete Enclosure 13.5.

___/___ 12.11.5.2.2 Complete new Data Book Table 2.2.1 from the data on Enclosure 13.5.

___/___ 12.11.5.2.3 Write a procedure change to place the new Table 2.2.1 in the appropriate unit's Data Book.

___/___ 12.11.5.2.4 Generate a work request to have IAE recalibrate N35 and N36 and calibrate bistables NC-203 and NC-206 using IP/0/A/3206/02K and new Data Book Table 2.2.1.

<u>NOTE</u>: DO NOT exceed 25% Full Power until IAE has completed calibrations of Step 12.11.5.2.4.

___/___ 12.11.5.3 When approximately 40-50% full power, if the excore quadrant tilts exceed 1.02, and it is expected that these tilts might not clear within 24 hours of exceeding 50% RTP, perform the data taking for PT/0/A/4600/02F, Incore and NIS Interim Recalibration with a QCFM while reactor power is between power increases. If the excore quadrant tilts are less than 1.02, or expected to be less than 1.02, mark this step N/A.

___/___ 12.11.5.4 Record the Intermediate Range NIS overlap data at 50% full power on Enclosure 13.2.

___/___ 12.11.6 Begin increasing reactor power from 50% to approximately 80% full power at a rate of approximately 2.5% per hour (not to exceed 3% per hour). See Limit and Precaution 6.6.

___/___ 12.11.7 Perform PT/0/A/4600/02E, Incore and NIS Recalibration: Post Outage, between 50% and 80% full power.

<u>NOTE</u>: Closely check the data acquired in Step 12.11.7 which is to be used for calibration for consistency since some of the data was acquired at <75% full power.

___/___ 12.11.8 Record the Intermediate Range NIS overlap data at 75% full power on Enclosure 13.2.

___/___ 12.11.9 Remain below approximately 80% full power until the recalibration work performed in Step 12.11.7 is completed by I&E.

___/___ 12.11.10 While holding at below 80% power call I&E 7300 System Engineer to take data on Thot and Tcold.

___/___ 12.11.11 I&E has evaluated data gathered in Step 12.11.10 to ensure operation at 100% will be acceptable with respect to ΔT. Record in the log any I&E setpoint changes in 7300.

___/___ 12.11.12 Begin increasing reactor power from 80% to 100% full power at a rate of 2.5% per hour (not to exceed 3% per hour). See Limit and Precaution 6.6.

12.11.13 At 100% full power, perform the following tests (steps) in any order (a suggested order is listed).

___/___ 12.11.13.1 Perform PT/0/A/4150/03, Thermal Power Output Calculation.

___/___ 12.11.13.2 Perform PT/0/A/4150/02A, Core Power Distribution.

___/___ 12.11.13.3 Perform PT/0/A/4150/08, Target Flux Difference Calculation.

___/___ 12.11.13.4 Perform PT/0/A/4600/02A, Incore and NIS Correlation Check.

___/___ 12.11.13.5 Perform PT/0/A/4150/04, Reactivity Anomolies Calculation.

___/___ 12.11.13.6 Record the Intermediate Range NIS overlap data at 100% full power on Enclosure 13.2 and forward a copy of the enclosure to the appropriate I&E engineer.

___/___ 12.11.13.7 Perform PT/1 or 2/A/4150/13, NC Flow Test.

NOTE: Once Step 12.11.13.6 is complete, Step 12.11.14 may be performed.

NOTE: Perform the next two steps in any order.

___/___ 12.11.14 I&E has received data from the NC Flow Test and has made a final ΔT evaluation for the cycle at 100% F.P.

13.0 Enclosures

13.1 PAO Data
13.2 Nuclear Instrumentation Overlap Data Sheet
13.3 Nuclear Heat Determination Data Sheet
13.4 Reactivity Computer Checkout Data Sheet
13.5 Intermediate Range Channels Worksheets
13.6 Connecting the Reactivity Computer
13.7 Intermediate Range High Level Trip Setpoints
13.8 Sequence of Control Rod Banks for Rod Swap
13.9 Verification of Shutdown Martin During Rod Swap
13.10 Shutdown Margin at Zero Power

Enclosure 13.1
PAO Data
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

P1393	Control Bank D Position
A0819	Loop A T_{avg}
A0825	Loop B T_{avg}
A0831	Loop C T_{avg}
A0837	Loop D T_{avg}
A1058	Loop A ΔT
A1070	Loop B ΔT
A1082	Loop C ΔT
A1094	Loop D ΔT
A1106	Reference Temperature T_{ref}
P1355	Rx. Thermal Power - Best Estimate
P1385	Rx. Thermal Power - Best Estimate
A1081	Generator Megawatts
P1447	Primary Thermal Output %
P1445	Secondary Thermal Output %
P1469	P/R Avg. Level 1 Min. Avg. Quad. 4 (N-44)
P1467	P/R Avg. Level 1 Min. Avg. Quad. 2 (N-42)
P1466	P/R Avg. Level 1 Min. Avg. Quad. 1 (N-43)
P1468	P/R Avg. Level 1 Min. Avg. Quad. 3 (N-44)
A1006	Turbine Impulse Chamber Pressure I

Enclosure 13.2
Nuclear Instrumentation Overlap Data Sheet
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

	Source Range		Intermediate Range	
	N-31	N-32	N-35	N-36
Control Board	CPS	CPS	amps	amps
NIS Cabinet	CPS	CPS	amps	amps

Picoammeter __________ amps

After one decade increase on IR

	Source Range		Intermediate Range	
	N-31	N-32	N-35	N-36
Control Board	CPS	CPS	amps	amps
NIS Cabinet	CPS	CPS	amps	amps

Picoammeter __________ amps

After one decade increase on IR

	Source Range		Intermediate Range	
	N-31	N-32	N-35	N-36
Control Board	CPS	CPS	amps	amps
NIS Cabinet	CPS	CPS	amps	amps

Picoammeter __________ amps

Readings when Source Range blocked

	Source Range		Intermediate Range	
	N-31	N-32	N-35	N-36
Control Board	CPS	CPS	amps	amps
NIS Cabinet	CPS	CPS	amps	amps

Picoammeter __________ amps

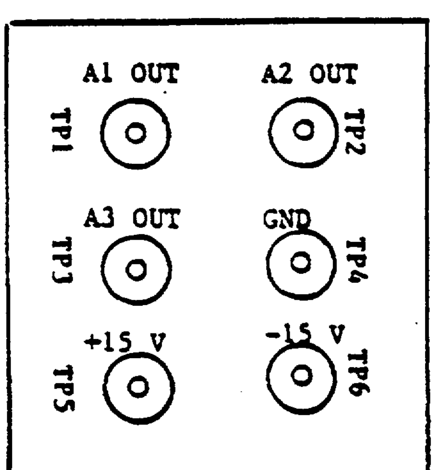
Recorded By ________________ Date ______________ Unit _____ Cycle _____
Checked By ________________ Date ______________

Enclosure 13.2
Nuclear Instrumentation Overlap Data Sheet
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

Power Level	Volts N-35	Volts N-36	Thermal Power, Best Est. (P1385)	Recorded By (Date/Time)
3%				
10%				
20%				
25%				
50%				
75%				
100%				

Unit _______ Cycle _______

NOTE: Data at 20 and 25% are needed to complete Enclosure 13.5. All other data are for info only.



Note: IR voltage data is to be taken inside each IR drawer. Take readings across terminals TP3 and TP4 as shown on schematic. Set Fluke to DC Volts, 0 to 10 volt scale. TP3 is a grey terminal and TP4 is a black terminal.

Enclosure 13.3
Nuclear Heat Determination Data Sheet
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

		Flux Levels (amps)		
	Time	Reac. Comp. Picoammeter from P.R. ___	N-35	N-36
Try 1 Start				
Nuclear Heat				
Try 2 Start				
Nuclear Heat				
Avg. of 2 nuclear heat readings				

Zero Power Physics Testing Range

__________ amps to __________ amps on power range NI __________

Recorded By ________________________

Date _______________

Checked By ________________________

Date _______________

McGuire Unit _______ Cycle ______

Enclosure 13.4
Reactivity Computer Checkout Data Sheet
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

McGuire Unit ____
Cycle ____

Recorded By ____________
Date ____________
Checked By ____________
Date ____________

Date	Time	Initial Flux Level (amps) Picoammeter	Measured Doubling Time Seconds	Calculated Period Seconds	Measured Reactivity $\Delta\rho c$ (from computer) pcm	Calculated Reactivity $\Delta\rho_{DT}$ (from period) pcm	$\left\lvert\frac{\Delta\rho_c - \Delta\rho_{DT}}{\Delta\rho_{DT}}\right\rvert$

Enclosure 13.5
Intermediate Range Channels Worksheet
Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing

Step 1: From Enclosure 13.2 record below the values of Thermal Power Best Estimate which most closely correspond to 20% and 25% power levels.

Step 2: From Enclosure 13.2 record below the voltage data given for the power levels above.

Step 3: Convert amp voltage, Eout, from Step 2 to Current, Iin, by using the following equation. Record values below on table.

$$Iin = \left\{ (1 \times 10^{-4}) \left(10^{\left[\frac{Eout}{1.25} - 7\right]}\right) \right\} - 1 \times 10^{-11}$$

	Step 1 Power Level	Step 2 Eout N35 Volts	Step 2 Eout N36 Volts	Step 3 Iin Current N35 Amps	Step 3 Iin Current N36 Amps
a)					
b)					

Step 4: Complete page 2 of 3 and 3 of 3 of this enclosure by linearly extrapolating above data to 15%, 20%, 25% and 30% power as indicated, and then converting to volts as indicated.

Calculated By _______________ Date __________

Checked By _______________ Date __________

Enclosure 13.5
Intermediate Range Channels Worksheet
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

NOTE: Data is from Enclosure 13.5 page 1 of 3.

1. 20% power current for N36 (Rod Stop) = ________ amps

2. 25% power current for N36 (High Flux Trip) = ________ amps

3. 30% power current for N36 (T.S. Allowable Value) = ________ amps

4. N36 Hi Flux Rod Stop Reset at 15% RTP = ________ amps

5. N36 Hi Flux Trip Reset at 20% RTP = ________ amps

NOTE: Convert the values found in Step 1 through 5 from amps to volts using the following equation. (Round to 3 decimal places.)

$$Eout = 8.75 + 1.25 \, Log_{10}\left(\frac{Iin + Iid}{Iref}\right), \text{ volts}$$

where $Iid = 1 \times 10^{-11}$ amps

$Iref = 1 \times 10^{-4}$ amps

6. N36 20% power voltage (Rod Stop) (Use Step 1.0 as Iin) ________ volts

7. N36 25% power voltage (High Flux Trip) (Use Step 2.0 as Iin) ________ volts

8. N36 30% power voltage (T.S. Allowable Value) (Use Step 3.0 as Iin) ________ volts

9. N36 Hi Flux Rod Stop Reset at 15% Power (Use Step 4.0 as Iin) ________ volts

10. N36 Hi Flux Trip Reset at 20% Power (Use Step 5.0 as Iin) ________ volts

Calculated By ____________ Date ________

Checked By ____________ Date ________

Enclosure 13.5
Intermediate Range Channels Worksheet
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

NOTE: Data is from Enclosure 13.5 page 1 of 3.

1. 20% power current for N35 (Rod Stop) = _______ amps

2. 25% power current for N35 (High Flux Trip) = _______ amps

3. 30% power current for N35 (T.S. Allowable Value) = _______ amps

4. N35 Hi Flux Rod Stop Reset at 15% RTP = _______ amps

5. N35 Hi Flux Trip Reset at 20% RTP = _______ amps

NOTE: Convert the values found in Step 1 through 5 from amps to volts using the following equation. (Round to 3 decimal places.)

$$Eout = 8.75 + 1.25 \, Log_{10}\left(\frac{Iin + Iid}{Iref}\right), \text{ volts}$$

where $Iid = 1 \times 10^{-11}$ amps

$Iref = 1 \times 10^{-4}$ amps

6. N35 20% power voltage (Rod Stop) (Use Step 1.0 as Iin) _______ volts

7. N35 25% power voltage (High Flux Trip) (Use Step 2.0 as Iin) _______ volts

8. N35 30% power voltage (T.S. Allowable Value) (Use Step 3.0 as Iin) _______ volts

9. N35 Hi Flux Rod Stop Reset at 15% Power (Use Step 4.0 as Iin) _______ volts

10. N35 Hi Flux Trip Reset at 20% Power (Use Step 5.0 as Iin) _______ volts

Calculated By __________ Date _______

Checked By __________ Date _______

Enclosure 13.6
Connecting the Reactivity Computer
Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing

NOTE: Any one of the four power range channels may be used. For clarity NI-43 is chosen arbitrarily.

Initial/Date		
____/____ ____/____IV	13.6.1	Have IAE place Channel NI-43 in the tripped condition with input plugs removed by using the "Prerequisites" and "Removing Power Channel from Service" sections of IP/0/A/3207/03K (power range cal.) <u>in</u> <u>their</u> <u>entirety</u>. <u>NOTE</u>: This procedure does not necessarily <u>require</u> that the channel be placed in the tripped condition, or that the input plugs be removed. Inform the technician that these things are necessary for Performance testing.
____/____ ____/____IV	13.6.2	Verify detector A and B input plugs and high voltage plug have been disconnected.
____/____	13.6.3	Clean all three cable connectors.
____/____ ____/____IV	13.6.4	Connect the A input plug to the A connector, the B input plug to the B connector, and the HV plug to the HV connector on the Reactivity Computer Black Box.
____/____ ____/____IV	13.6.5	Connect the HV cable and P cable from the reactivity computer to the HV and Det AB Signal terminals on the Black Box.
____/____	13.6.6	Secure the Black Box to a rack mount with a tie wrap.
____/____	13.6.7	To return NI-43 to service, verify the high voltage power supply and picoammeter at the Reactivity Computer are off.
____/____		13.6.7.1 Inform Shift Supervisor you are returning NI-43 to service.
____/____ ____/____IV	13.6.8	Disconnect the A and B input plugs and the HV input plug from the Reactivity Computer Black Box.
____/____	13.6.9	Clean all three connectors.
____/____ ____/____IV	13.6.10	Have IAE return Channel NI-43 to service by performing the "Prerequisites" and "Returning Power Range Channel to Service" sections of IP/0/A/3207/03K (power range cal.) <u>in</u> <u>their</u> <u>entirety</u>.

Enclosure 13.7
Intermediate Range High Level Trip Setpoints
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

N-35 trip setpoint (25% full power)

= __________ amps

N-36 trip setpoint (25% full power)

= __________ amps

Recorded By _______________

Date _______________

Unit _______ Cycle _______

Enclosure 13.8
Sequence of Control Rod Banks for Rod Swap
Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing

Reference Bank	______________
First Bank	______________
Second Bank	______________
Third Bank	______________
Fourth Bank	______________
Fifth Bank	______________
Sixth Bank	______________
Seventh Bank	______________
Eighth Bank	______________

NOTE: Some of the Banks may not be measured by rod swap; mark these Banks in the sequence N/A. Indicate justification in the test log if banks will not be measured.

Recorded By ______________

Date ______________

Unit ______ Cycle ______

Enclosure 13.9
Verification of Shutdown Margin During Rod Swap
Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing

1. Inserted control rod worth at BOL and at zero power insertion limits (from Enclosure 13.10, Step 2) ________ pcm

2. Rod swap Reference Bank worth ________ pcm

3. Step 1. > 1.10 · Step 2. (10% conservatism on the predicted Reference Bank Worth) Yes ____ No ____

Recorded By ________________

Checked By ________________

Date ________________

Unit _____ Cycle _____

Enclosure 13.10
Shutdown Margin at Zero Power
Post Refueling Controlling Procedure for Criticality,
Zero Power Physics, and Power Escalation Testing

ntrol rod position at zero power insertion limits:

CB ________ steps withdrawn

CC ________ steps withdrawn

CD ________ steps withdrawn

serted control rod worth at BOL and at the zero power sertion limits (Data Book Curve 6.3A) ________pcm

L, HZP, no xenon total rod worth ata Book Table 6.3.1) ________pcm

ailable rod worth at BOL and at zero power insertion limits tep 3 - Step 2) ________pcm

rth of highest worth stuck rod at BOL ata Book Table 6.3.2) ________pcm

ailable Shutdown Margin at BOL and at zero power insertion limits Step 4 - Step 5) · 0.90] ________pcm

quired Shutdown Margin ________pcm

ep 6 > Step 7 Yes ________ No ________

Recorded By ________________

Checked By ________________

Date ________________

Unit ________ Cycle ________

QUESTION 2

Attachment 2

FOR INFORMATION ONLY

Form 34731 (10-81)
(Formerly SPD-1002-1)

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: PT/0/A/4150/11
Change(s) 0 to 3 Incorporated

(2) STATION: McGuire

(3) PROCEDURE TITLE: Control Rod Worth Measurement

(4) PREPARED BY: Michael S. Kitlan Jr DATE: 3/30/84

(5) REVIEWED BY: D K Marquis DATE: 4/25/84

Cross-Disciplinary Review By: ______ N/R: DKM

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: Tony L. McConnell Date: 4/26/84

(8) MISCELLANEOUS:

Reviewed/Approved By: ______ Date: ______

Reviewed/Approved By: ______ Date: ______

This copy has been compared with the
Control Copy and is verified correct.

Initial ______ Date ______ Time ______

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION
CONTROL ROD WORTH MEASUREMENT

1.0 Purpose

1.1 To measure the differential and integral worth of any of the Controlling Banks or Shutdown Banks.

1.2 To measure the differential boron worth over the range being tested.

2.0 References

2.1 Rod and Boron Worth Measurements During Boron Dilution, DAP/DBP-SU-7.4.

3.0 Time Required

3.1 2 hours, 2 engineer for each Rod Bank measured.

4.0 Prerequisite Tests

None

5.0 Test Equipment

5.1 Reactivity Computer (with flux signal from top and bottom of one power range channel).

5.2 Two pen strip chart recorder with reactivity and T_{avg} signals.

5.3 Two pen strip chart recorder with pressurizer water level and flux signal.

6.0 Limits and Precautions

6.1 The NC System temperature is controlled preferably by secondary steam bypass to the condenser or by secondary steam dump to the atmosphere. Temperature control may alternatively by affected by steam generator blowdown.

6.2 Normally all reactor coolant pumps should be operating for maximum mixing in the NCS. If all reactor coolant pumps are not operating, the operating pumps should be those on the NCS charging loops (A&D). See Tech Spec 3.4.1.1 and 3.10.4 if all reactor coolant pumps are not operating.

6.3 The rod insertion limit will be violated during this test. The operators should be made aware in advance and should anticipate the associated alarms. Technical Specification 3.10.3 allows for this.

6.4 Chart speeds for rod worth measurements should be about .2 to 1 in./min. The sawtooth of the reactivity trace should be kept at about a 45° angle.

7.0 <u>Required Unit Status</u>

Initial/Date

____/____ 7.1 The unit is just critical in the Startup Mode (Mode 2) at zero power with the flux level in the required testing range.

____/____ 7.2 Record in the log the unit to which this test applies.

8.0 <u>Prerequisite System Conditions</u>

____/____ 8.1 The reactor coolant system pressure is at 2235 ±50 psig.
<u>NOTE</u>: Maintain NCS pressure within ±25 psig of established pressure during the test.

____/____ 8.2 The reactor coolant system temperature is 557°F +1, -5°F.
<u>NOTE</u>: Maintain NCS temperature within ±1°F of established temperature during the test.

____/____ 8.3 T~~he pressurizer spray control is in~~ manual ~~with spray flow established at the maximum rate cons~~is~~tant with pressurize~~r heater capability.
Pressurizer Control is set to maximize pressure per capability
[illegible] 5/6/85

____/____ 8.4 Test equipment is set up per Section 5.0.

____/____ 8.5 The unit is sufficiently stable as determined by the test coordinator.

____/____ 8.6 The indicated core reactivity is less than ±1 pcm.

____/____ 8.7 Record the requested data on Enclosure 13.1 for this step.

____/____ 8.8 The Control Rods are positioned as specified by the Test Coordinator.

____/____ 8.9 Complete Enclosure 13.4 only if no overlap data is to be taken. Mark this step, Step 8.9.1, and Enclosure 13.4 N/A if overlap data is to be taken.

____/____ 8.9.1 Bank selector switch is positioned in bank select to the bank being measured if 8.9 is not N/A.

___/___ 8.10 Complete Enclosure 13.5 only if overlap data is to be taken. Mark this step, Step 8.10.1 and Enclosure 13.5 N/A if this is not the case.

___/___ 8.10.1 Bank select switch is in overlap (manual) unless 8.10 is N/A.

9.0 <u>Test Method</u>

With the RCCA's positioned as requested by the Test Coordinator, the amount of demineralized water/boric acid required to compensate for the forthcoming configuration adjustment is determined. A continuous boron concentration change is initiated at a rate of approximately 500 pcm/hr. The RCCA's are moved in discrete increments to compensate for the change in boron concentration. From the data gathered, the differential and integral worth of RCCAs being measured is determined.

10.0 <u>Data Required</u>

10.1 Rod positions and reactivity will be recorded on Enclosure 13.1.

10.2 The following data should be recorded on the strip charts: (attach charts to this procedure)

10.2.1 RCCA positions before and after each discrete increment.

10.2.2 Parameter scale and chart speed should be written on the chart.

10.3 Plot of integral and differential rod worth on Enclosure 13.2.

10.4 Predicted data on Enclosure 13.4.

11.0 <u>Acceptance Criteria</u>

11.1 The rod worth of the rod or bank being measured is within ±10% of the predicted rod worth as given on Enclosure 13.4.

11.2 The integral rod worth of Control Banks A, B, D, D in overlap is within ±4% of the total measured values of Control Banks A, B, C and D individually as given on Enclosure 13.5. This <u>only</u> applies if overlap data is to be taken.

CH. #5 6/26/95 [illegible]

11.3 IF THE BANK BEING MEASURED IS THE REFERENCE BANK FOR ROD SWAP, THE ABSOLUTE VALUE OF THE PERCENT DIFFERENCE BETWEEN MEASURED AND PREDICTED INTEGRAL WORTH IS ≤ 15%.

CH. 6/1 [illegible]

NOTE: THIS ACCEPTANCE CRITEREON DOES NOT APPLY IF THE BANK BEING MEASURED IS THE REFERENCE BANK FOR ROD SWA

12.0 Procedure

itial/Date

NOTE: The following steps explain the general method for performing rod worth measurements for single RCCA's, Groups of RCCA's, or Banks of RCCA's during either dilution or boration.

___/___ 12.1 Verify that the strip chart recorders specified in Section 5.0 are operable and set up as required.

___/___ 12.2 Determine the amount of demineralized water/boric acid to compensate for the required configuration adjustment. See Enclosure 13.3 for an example of how to determine this.

___/___ 12.3 Record the beginning boric acid and primary water integrator values in the test log. If possible, reinitialize readings to 0.0.

___/___ 12.4 Using the reactivity computer, measure the worth of the bank being tested from its current position to the fully withdrawn/inserted position. Record the data on Enclosure 13.1.

NOTE: This is similar to a Boron Endpoint Measurement.

___/___ 12.5 Using the number obtained in Step 12.2, initiate the required boron concentration change at a rate which will not cause a reactivity rate of change of >500 pcm/hr.

NOTE: This guideline corresponds to a dilution rate of approximately 2500 gallons per hour (40 GPM) or a boration rate of approximately 250 gallons per hour (4 GPM) of 4 w/o boric acid. See Enclosure 13.3 for an example of this.

___/___ 12.6 Insert/withdraw RCCA's in discrete increments in order to compensate for dilution/boration. These increments should be limited such that the resultant reactivity change are within the guidelines of approximately ±20 pcm. During these measurements, record all relevant data on the strip charts in use. See Section 10.2.

___/___ 12.7 Terminate the boron concentration change such that the desired rod configuration is achieved.

NOTE: A delay of some minutes (typically 15 minutes) is unavoidable between termination of the transient and stabilization. This delay should be anticipated.

NOTE: Normally the desired rod configuration will be either overshot or undershot. The Test Coordinator must evaluate the effects of this situation on the results of the affected test. If effects are unacceptable, the Test Coordinator can repeat Steps 12.2 through 12.5 to correct the situation.

NOTE: If there is any overshoot, the bank selector switch may be changed to the next bank.

NOTE: For rod swap measurements, terminate the boron concentration change such that the final position of the bank is almost to the fully inserted position.

___/___ 12.8 Using the reactivity computer, measure the worth of the bank being measured from its current position to the fully inserted/withdrawn position. Record the data in the test log for later entry into Enclosure 13.1. Mark this step as N/A if this data is already obtained (i.e., overshoot to next bank).

NOTE: This is similar to a Boron Endpoint Measurement.

___/___ 12.9 Record the final primary water and/or boric acid integrator values in the test log.

___/___ 12.10 Record the "FINAL" data requested on Enclosure 13.1.

___/___ 12.11 After the test is over, record the required data on Enclosure 13.1 from the strip charts.

___/___ 12.12 Verify the acceptance criteria has been met.

___/___ 12.13 Using the data on Enclosure 13.1, complete the plot(s) on Enclosure 13.2.

___/___ 12.14 In the log, calculate the differential boron worth over the bank being measured by dividing the measured rod worth by the difference in boron concentration over the rod worth measurement.

13.0 Enclosures

13.1 Rod Worth Measurement Data Sheet

13.2 Rod Worth Curves

13.3 Example of Determination of Dilution Rate

13.4 Predicted Rod Worth Data

13.5 Rod Worth Data if Worths in Overlap are to be Taken

Control Rod Worth Measurement
Enclosure 13.1
Rod Worth Measurement Data Sheet

Step 8.7 McGuire Unit _____ Cycle _____

Bank or RCCA Identification __________ Boration ☐ Dilution ☐ (Check one

Date ______________ Power (amps) ______________

Initial Shutdown Bank Positions: A_____ B_____ C_____ D_____ E_____

Initial Control Bank Positions: A_____ B_____ C_____ D_____

Overlap Measurement (yes or no) _____

	Initial	Final (Step 12.10)
NCS Boron Concentration (ppm):		
Pressurizer Boron Concentration (ppm):		
NCS Temperature (T_{avg}) ^{o}F:		
NCS Pressure, psig:		

Step 12.9

Time	RCC Position (Steps Withdrawn)			Delta H	Reactivity (pcm)		
	Initial	Final	Average	(Δh)	$\Delta\rho$	$\Sigma\ \Delta\rho$	$\Delta\rho/\Delta h$

REMARKS ____

Recorded By ______________________ Page ___ of ___

Date

Control Rod Worth Measurement
Enclosure 13.1
Rod Worth Measurement
Data Sheet

Step 12.9 (continued)

Time	RCC Position (Steps Withdrawn)			Delta H	Reactivity (pcm)		
	Initial	Final	Average	(Δh)	Δρ	Σ Δρ	Δρ/Δh

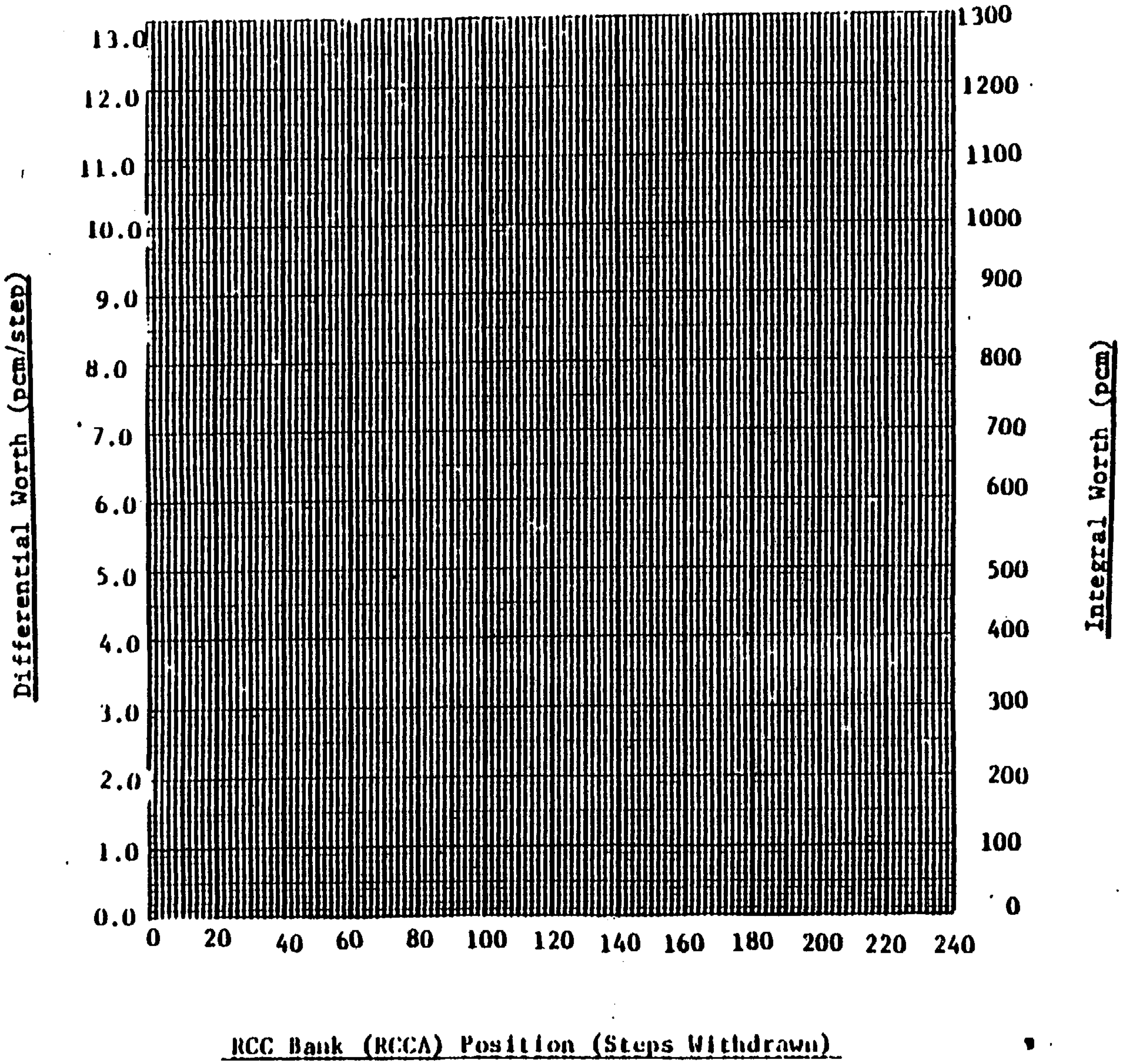
REMARKS ____

____

Recorded By ____________________ Page ___ of ___

Control Rod Worth Measurement
Enclosure 13.2
Rod Worth Curves

Differential and Integral RCC Bank (RCCA) Worth



McGuire Unit: ____________

Bank: ____________

Date: ____________

Test Conditions:

1. RCC Bank Positions:
   - SDA ____________
   - SDB ____________
   - SDC ____________
   - SDD ____________
   - SDE ____________
   - CA ____________
   - CB ____________
   - CC ____________
   - CD ____________
2. Power Level:
   ____________ amps
3. NC Temp.:
   - Initial: ____________ °F
   - Final: ____________ °F
4. NC Press.:
   - Initial: ____________ °F
   - Final: ____________ °F
5. Avg. Core Burnup:
   ____________ MWD/MTU
   ____________ efpd

Control Rod Worth Measurement
Enclosure 13.3
Example of Determination of Dilution Rate
(Illustration Purposes Only)

It is desired to dilute Control Bank B from 223 to 0 steps at a rate not to exceed 500 pcm/hr.

1. The starting point is known: Initial Boron Concentration is 1130 ppm.

2. Go to Figure A.3 in the Core Design Report (or any other applicable document). The Integral Rod worth for Control Bank B from 223 to 0 is 909 pcm.

3. Go to Curve 6.2 in the Data Book at 1130 ppm BOL and get -10.7 pcm/ppm for the differential boron worth.

4. 990 pcm ÷ 10.7 pcm/ppm = -92.5 ppm change (dilute)

5. 1130 ppm - 92.5 ppm = 1037 ppm ending boron concentration.

6. Go to Figure 5.1 in the Data Book. To go from 1130 to 1037 ppm, add about 5656 gallons of demineralized water.

7. An alternate method is to use the Boron Predict Program on the OAC.

8. The maximum rate is 500 pcm/hr; therefore:

$$\frac{5656 \text{ gal.}}{990 \text{ pcm}} \times \frac{500 \text{ pcm}}{\text{hour}} \times \frac{1 \text{ hour}}{60 \text{ min.}} = 47.6 \text{ gpm}$$

9. To be conservative, go at 45 gpm.

10. Expect the time for the rod worth measurement to be

$$\frac{990 \text{ pcm}}{500 \text{ pcm/hr}} \cong 2 \text{ hours}$$

Control Rod Worth Measurement
Enclosure 13.4
Predicted Rod Worth Data

Step 8.9

Complete one of the following two lines. Mark the other N/A.

Bank Being Measured (i.e., Control/Shutdown) ______________________
Rod Being Measured ______________________

Predicted Rod Worth Value for the above condition.

____________ pcm ±10%

OR

____________ pcm ± ____________ pcm

This information was transmitted to McGuire Nuclear Station by/in (list reference):

____

Reason for this test (refueling, etc.): ______________________________

____

Recorded By ____________________
Date ____________________
McGuire Unit ____________________
Cycle ____________________

Control Rod Worth Measurement
Enclosure 13.5
Rod Worth Data if Worths in Overlap are to be Taken

Step 8.10

Individual Measured Rod Worth Values (not in overlap):

Control Bank A ______________ pcm
Control Bank B ______________ pcm
Control Bank C ______________ pcm
Control Bank D ______________ pcm

Sum of Control Bank A, B, C and D individual rod worths: ______________ pcm ±4%

OR

______________ pcm ± ______________ pcm

The above individual measured rod worth values were obtained from (list procedures):

____

which were performed on (list dates): ________________________________

Recorded By ____________________
Date ____________________
McGuire Unit ____________________
Cycle ____________________

Fn 34805 (7-81)

Sheet ______ of ______

Procedure No. PT/0/A/4150/11

Test Logger ______

DUKE POWER COMPANY

McGuire NUCLEAR STATION

TEST LOG

Date	Time	Remarks	Initial

QUESTION 2

Form 34731 (R8-85)

FOR INFORMATION ONLY

Attachment 3

(1) ID No. PT/0/A/4150

Change(s) 0 to 5 Incorporated

PREPARATION

(2) STATION McGuire

(3) PROCEDURE TITLE Control Rod Worth Measurement: Rod Swap

(4) PREPARED BY L J Kunkes DATE 6/10/86

(5) REVIEWED BY M Todd Killer DATE 6/10/86

Cross-Disciplinary Review By N/R MTK

(6) TEMPORARY APPROVAL (If Necessary)

By (SRO) Date

By Date

(7) APPROVED BY Bruce Hamilton DATE 6/11/86

(8) MISCELLANEOUS

Reviewed/Approved By Date

Reviewed/Approved By Date

(9) COMMENTS (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.) ☑ ADDITIONAL CHANGES INCLUDED.

COMPLETION

(10) COMPARED WITH CONTROL COPY TWMirnia DATE 1/29/87

(11) DATE(S) PERFORMED

(12) PROCEDURE COMPLETION VERIFICATION

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Acceptance criteria met?

VERIFIED BY DATE

(13) PROCEDURE COMPLETION APPROVED DATE

(14) REMARKS (Attach additional pages, if necessary.)

RECEIVED
FEB 3 1987

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION
CONTROL ROD WORTH MEASUREMENT: ROD SWAP

1.0 Purpose

1.1 To verify that the reactivity worth of the Reference RCC bank, as determined through reactivity computer measurement data, is consistent with design predictions.

NOTE: The reference RCC bank is the bank which has the predicted highest reactivity worth of all control and shutdown banks when inserted into an otherwise unrodded core.

1.2 To verify that the reactivity worth of each control and shutdown bank (except the reference bank), as inferred from data following iso-reactivity interchange with the reference bank, is consistent with design predictions.

2.0 References

2.1 Rod Bank Worth Measurements Utilizing Bank Exchange, WCAP-9863-A, May 1982.

2.2 Control Rod Worth Measurement, PT/0/A/4150/11

2.3 Post Refueling Controlling Procedure for Criticality, ZPPT, and Power Escalation Testing, PT/0/A/4150/21

2.4 Technical Specifications 3.4.1.1, 3.10.4, 3.10.3, and 3.10.2.

3.0 Time Required

3.1 8 hours, 1 engineer

4.0 Prerequisite Tests

Initial/Date

____/____ 4.1 PT/0/A/4150/10, ARO Boron Endpoint Measurement

NOTE: It is only necessary to obtain a value for ARO Boron Endpoint.

5.0 Test Equipment

5.1 Reactivity Computer (with flux signal from top and bottom of one power range channel).

5.2 Two two-pen strip chart recorders. One chart recorder should have reactivity (on a scale of 10 pcm/inch, with 0 pcm being the center of the recorder sheet), and T_{avg} from one loop (on a scale of 1°F/inch for 556 to 558°F set up on one side of the recorder sheet). The other chart recorder should have flux (on a scale of 0 to the top end of the testing decade in amps) and pressurizer level (on a scale of 10% level/inch). Chart speeds should be 1 inch/min.

NOTE: The specifications in this step may be altered by the Test Coordinator as necessary to accommodate equipment limitations, as long as all four signals are recorded or trended.

6.0 Limits and Precautions

6.1 The NC system temperature is controlled preferably by steam dump to the condenser. Temperature control may alternatively be affected by steam generator blowdown.

6.2 Normally all reactor coolant pumps should be operating for maximum mixing in the NCS. If all reactor coolant pumps are not operating, the operating pumps should be those on the NCS charging loops (A and/or D). See Tech Spec 3.4.1.1 and 3.10.4 if all reactor coolant pumps are not operating.

6.3 The rod insertion limit and bank overlap sequence will be violated during this test. The operators should be made aware in advance and should anticipate the associated alarms. Technical Specification 3.10.2 and 3.10.3 allows for this.

6.4 Maintain the flux level in the zero power test range established in Reference 2.3.

6.5 Prior to switching the rod control selector switch from one bank to another, verify both groups of the bank (if the bank has two groups) are at the same position in order to avoid group misalignment.

7.0 Required Unit Status

Initial/Date

____/____ 7.1 The unit is just critical in the Startup Mode (Mode 2) at zero power with the flux level in the zero power test range established in PT/0/A/4150/21, "Post Refueling Controlling Procedure for Criticality, ZPPT, and Power Escalation Testing."

____/____ 7.2 Record in the log the unit and cycle to which this test applies.

8.0 Prerequisite System Conditions

NOTE: The following steps may be signed off in any order.

____/____ 8.1 The reactor coolant system temperature is 557°F +1, -5°F.

NOTE: Maintain NCS temperature within ±1°F of established temperature during the test.

____/____ 8.2 The difference between NC loop, pressurizer, and VCT boron concentrations is less than 20 ppm. List on Enclosure 13.3.

NOTE: Do not use the boronometer.

Boron samples are desirable but are not necessary for completion of test. Samples may be waived if reason is logged in the test log. Samples may be taken during the data taking at the test coordinator's request.

____/____ 8.3 Xenon worth rate is changing less than ±.1 pcm/min.

____/____ 8.4 Test equipment is set up per Section 5.0.

____/____ 8.5 All available pressurizer heaters are on as needed, in order to improve mixing by maximizing the pressurizer spray.

____/____ 8.6 All control and shutdown banks are fully withdrawn except Control Bank D which is at a position greater than about 215 steps withdrawn.

____/____ 8.7 The Rod Control Selector switch is in Bank Select Mode set on Control Bank D.

____/____ 8.8 Complete Enclosure 13.1 with the predicted data. See Reference 2.3, Enclosure 13.8 for banks to be measured. See Enclosure 13.2 for an explanation of nomenclature used in this test.

NOTE: If any banks are not being measured mark the blanks on Enclosure 13.1 N/A.

9.0 Test Method

The RCC bank with the highest predicted value of reactivity worth is measured using the dilution technique per PT/0/A/4150/11. This bank serves as a reference. The integral worth of the remaining RCC banks is implied from the difference in the critical rod position of the reference bank with and without the insertion of bank being tested. The implied integral worths are then compared to predicted rod worths.

10.0 Data Required

10.1 The following conditions for the approximate time of criticality before each bank exchange, recorded on Enclosure 13.4:

Time

Just critical height of reference bank

10.2 Nuclear design predictions on Enclosure 13.1.

10.3 Boron concentration information for the NCS and pressurizer on Enclosure 13.3. Boron samples are desirable but are not necessary for completion of test. Samples may be waived if the reason is logged in the test log.

10.4 A copy of the rod positions and rod worths for the reference bank from Enclosure 13.1 of PT/0/A/4150/11 when this test is complete.

10.5 The calculated, implied integral worth (W_x^I) for each RCC bank except the reference bank. List data on Enclosure 13.4.

10.6 The percent difference between inferred and predicted worths for each individual RCC banks (ε_1) and for the sum of all banks (ε_2) on Enclosure 13.5.

11.0 Acceptance Criteria

11.1 The absolute value of the percent difference between measured and predicted integral worth for the reference bank is ≦15% (from Enclosure 13.5 $(\varepsilon_1)_1$ ≦15%).

11.2 From Enclosure 13.5, the calculated value ε_2 ≦10%.

11.3 For all RCC banks other than the reference bank; either:

a) From Enclosure 13.5, ε_{1_x} ≦30% for each bank or

or

b) $W_x^I - W_x^P$ ≦200 pcm for each bank,

whichever is greater.

12.0 Procedure

Initial/Date

NOTE: See Enclosure 13.2 for an explanation of all nomenclature used in this test.

12.1 Measure the integral reactivity worth of the reference bank as follows:

NOTE: The reference bank is defined as that bank which is predicted to have the highest worth, of all control and shutdown banks, when inserted into an otherwise un-rodded core (see Enclosure 13.1 for the identity of this bank). In this procedure, all banks will be referred to by the bank number, except the reference bank. If the reference bank is currently positioned at less than 228 steps withdrawn (i.e., if it is Control Bank D), continue with step 12.1.5. Mark steps 12.1.1 to 12.1.4 NA. If the reference bank is positioned at 228 steps withdrawn, continue on at Step 12.1.1.

____/____ 12.1.1 Insert the reference bank until the indicated reactivity is approximately -10 pcm.

____/____ 12.1.2 Withdraw the bank inserted below 228 until the indicated reactivity is approximately +10 pcm.

____/____ 12.1.3 Repeat steps 12.2.1 and 12.2.2 until the previously inserted bank is fully withdrawn.

____/____ 12.1.4 Adjust the position of the reference bank until the reactor is just critical. Record this position in the test log.

____/____ 12.1.5 Perform Control Rod Worth Measurement per PT/0/A/4150/11 on the reference Bank.

____/____ 12.1.6 Attach a completed copy of PT/0/A/4150/11 Enclosure 13.1 to this procedure.

____/____ 12.1.7 Record the total reference bank rod worth from PT/0/A/4150/11 Enclosure 13.1 on Enclosure 13.4 as shown.

____/____ 12.1.8 Ensure the reactor is critical at the same reference bank position as was obtained at the end of PT/0/A/4150/11.

12.2 Measure the reactivity worth of the remaining control and shutdown banks, relative to the reference bank, as follows:

<u>NOTE</u>: The relative worth of each RCC bank is obtained from the critical position of the reference bank (initially nearly fully inserted) after full insertion of the bank being measured (initially fully withdrawn), at constant RCS boron concentration.

____/____
(√) ___ ___ ___ ___ ___
2 3 4 5 6
___ ___
7 8

12.2.1 Record the initial critical bank configuration on Enclosure 13.4 for the reference bank.

____/____
(√) ___ ___ ___ ___ ___
2 3 4 5 6
___ ___
7 8

12.2.2 Insert bank 1 (identify this bank on top of Enclosure 13.5; i.e., Bank 1 is S/D E or Cont. B, etc.) until the reactivity indicated by the reactivity computer is approximately -20 pcm.

____/____
(√) ___ ___ ___ ___ ___
2 3 4 5 6
___ ___
7 8

12.2.3 Withdraw the reference Bank until the indicated reactivity is approximately +20 pcm.

<u>NOTE</u>: Maintain the flux within the zero power test range established in Reference 2.4.

____/____
(√) ___ ___ ___ ___ ___
2 3 4 5 6
___ ___
7 8

12.2.4 Repeat Steps 12.4.2 and 12.4.3 until bank 1 is fully inserted. Keep the indicated reactivity within ±20 pcm.

____/____
(√) ___ ___ ___ ___ ___
2 3 4 5 6
___ ___
7 8

12.2.5 Adjust the position of reference bank until the reactor is just critical. Record the final critical configuration data on Enclosure 13.4.

____/____
(√) ___ ___ ___ ___ ___
2 3 4 5 6
___ ___
7 8

12.2.6 Insert the reference Bank 1 until the indicated reactivity is approximately -20 pcm.

___/___ 12.2.7 Withdraw bank 1 until the indicated reactivity is approximately +20 pcm.

(√) ___ ___ ___ ___ ___
2 3 4 5 6

___ ___
7 8

___/___ 12.2.8 Repeat Steps 12.4.6 and 12.4.7 until bank 1 is fully withdrawn.

(√) ___ ___ ___ ___ ___
2 3 4 5 6

___ ___
7 8

___/___ 12.2.9 Adjust the position of reference Bank until the reactor is just critical. Record the critical configuration data on Enclosure 13.4.

(√) ___ ___ ___ ___ ___
2 3 4 5 6

___ ___
7 8

___/___ 12.2.10 Repeat Steps 12.4.2 through 12.4.9 for the remaining, unmeasured control and shutdown banks numbered 2 through 8 instead of bank 1. Identify the bank beside the bank number on Enclosure 13.4.

<u>NOTE</u>: If any banks are not being measured mark the blanks on Enclosure 13.4 and the check off blanks in step 12.4.2 through 12.4.9 N/A.

___/___ 12.3 Have Chemistry take a NC & pressurizer boron sample and write the results on Enclosure 13.2.

<u>NOTE:</u> The test may continue while waiting for the boron samples.

<u>NOTE:</u> This completes the data acquisition section of the test.

___/___ 12.4 Compute the average of the reference bank critical position on Enclosure 13.4.

12.5 Compute the inferred worth for each control and shutdown bank (except the reference bank) as follows:

___/___ 12.5.1 Using the data from Enclosure 13.4, and the worth measurement data for the reference bank from Enclosure 13.1 of PT/0/A/4150/11, compute the value of $(\Delta\rho_1)_x$ as described below and record on Enclosure 13.4:

$$(\Delta\rho_1)_x = \left[W_R^M\right]_o^{(h_x^M)_o \text{ avg}}$$

where:

$\left[W_R^M\right]_o^{(h_x^M)_o \text{ avg}}$ is the measured integral worth of the reference bank from 0-steps to $(h_x^M)_o$ avg from Enclosure 13.1 of PT/0/A/4150/11.

NOTE: Linearly interpolate if $(h_x^M)_o$ avg does not correspond to the steps on Enclosure 13.1 of PT/0/A/4150/11.

and $(h_x^M)_o$ avg is the average of the initial and return critical positions of the reference bank before and after interchange with bank x as given on Enclosure 13.4.

____/____ 12.5.2 Using the data from Enclosure 13.4, the worth measurement data for the reference bank from Enclosure 13.1 of PT/0/A/4150/11 and the design data of Enclosure 13.1, compute the value of $\alpha_x\ (\Delta\rho_2)_x$ as described below and record on Enclosure 13.4:

$$\alpha_x(\Delta\rho_2)_x = \alpha_x \left[W_R^M\right]_{h_x^M}^{228}$$

where: $\left[W_R^M\right]_{h_x^M}^{228}$ is the measured integral worth of the reference bank from h_x^M to the fully withdrawn position from PT/0/A/4150/11, Enclosure 13.1. Linearly interpolate if h_x^M does not correspond to the steps on PT/0/A/4150/11 Enclosure 13.1. using measured IRW for Ref Bank based on h_x^m

h_x^M is the measured critical position of the reference bank after interchange with bank x from Enclosure 13.4.

and

α_x is a correction factor from Enclosure 13.1 to account for the influence of bank x on the worth of the reference bank.

Chow
mx
6/23/86

Note: If bank being measured has a worth greater than the Reference Bank worth, See Enclosure 13.6. Replace $\alpha_x(\Delta\rho_2)_x$ with WE on Enclosure 13.4.

____/____ 12.5.3 Compute the inferred integral worth of each bank x, W_x^I, as described below and record on Enclosure 13.4:

$$W_x^I = W_R^M - (\Delta\rho_1)_x - \alpha_x(\Delta\rho_2)_x$$

where: W_R^M is the measured total integral reference bank worth from PT/0/A/4150/11 Enclosure 13.1.

$(\Delta\rho_1)_x$ is from step 12.5.1.

and

$\alpha_x(\Delta\rho_2)_x$ is from step 12.5.2

Note: If Bank being recovered has a worth greater than the reference bank worth, compute the inferred integral worth of the bank as shown on Enclosure 13.6, where WE is given in the column marked $\alpha_x(\Delta\rho_2)_x$ on Enclosure 13.4 and WRo is given in the column marked $(\Delta\rho_1)_x$ on Enclosure 13.4.

Ch 06
MK
6/23/86

____/____ 12.5.4 Compute the percent difference between inferred and predicted worths for each individual RCC bank and the sum of all banks described below.

$$(\varepsilon_1)_x = \left[\frac{W_x^I - W_x^P}{W_x^P}\right] \times 100, \text{ in } \%$$

$$\varepsilon_2 = \left[\frac{\sum_{i=1}^{N} W_i^I - \sum_{i=1}^{N} W_i^P}{\sum_{i=1}^{N} W_i^P}\right] \times 100, \text{ in } \%$$

Fill in all blanks and summarize the calculations on Enclosure 13.5.

____/____ 12.6 Verify all acceptance criteria have been met.

13.0 <u>Enclosures</u>

13.1 Nuclear Design Predictions for Rod Interchange Measurements

13.2 Nomenclature

13.3 Log of Boron Concentrations

13.4 Critical Configuration Data

13.5 Comparison of Inferred Bank Worths with Design Predictions

13.6 Letter on Rod Swap

Control Rod Worth Measurement: Rod Swap
Enclosure 13.1
Nuclear Design Predictions
for Rod Interchange Measurements

McGuire Unit _____ Cycle _____

Bank No. (x)	Bank Identity +	W_x^P (pcm)	(b) h_x^P (steps)	(c) α_x
(a) Reference				
1				
2				
3				
4				
5				
6				
7				
8				

(a) Reference bank - the bank with the highest predicted integral worth.

(b) Reference bank critical position after interchange with bank x.

(c) Ratio of integral worth of the reference bank from h_x^P to the fully withdrawn position with and without x in the core.

\+ Control Bank C, Shutdown Bank E, etc.

NOTE: See Enclosure 13.2 for a complete listing of nomenclature used in this test.

Recorded By ____________________ Date ____________________
This data came from (list source): ____________________
____

Control Rod Worth Measurement: Rod Swap
Enclosure 13.2
Nomenclature

1.	W_x^P	Predicted reactivity worth of each control and shutdown bank when inserted individually into an otherwise unrodded core.
2.	W_x^I	The calculated, implied rod bank worths of bank x from rod exchange
3.	W_R^M	Measured rod bank worth of reference bank
4.	α_x	A correction factor which accounts for the effect of bank x on the partial integral worth of the reference bank, equal to the ratio of the integral worth of the reference bank from h_x^P to the fully withdrawn position with and without x in the core.
5.	$(\Delta\rho)_x$	The measured integral worth of the reference bank from h_x^M to the fully withdrawn position.
6.	h_x^P	The predicted critical position of the reference bank after interchange with bank x starting with reference bank at 0, bank x fully withdrawn.
7.	h_x^M	The measured critical position of the reference bank after interchange with bank x.
8.	$\left[W_R^M\right]_o^{(h_x^M)_{o\ avg}}$	Is the measured integral worth of the reference bank from 0 steps to $(h_x^M)_{o\ avg}$
9.	$(h_x^M)_{o\ avg}$	Is the average of the initial and return critical positions of the reference bank before and after interchange with bank x.
10.	$\left[W_R^M\right]_{h_x^M}^{228}$	Is the measured integral worth of the reference bank from h_x^M to the fully withdrawn position.

Control Rod Worth Measurement: Rod Swap
Enclosure 13.3
Log of Boron Concentrations

Date	Time Sample Taken	NCS Boron Concentrations			Comments	Recorded By
		+VCT	NCS	Press.		

McGuire Unit ___________
Cycle _____

NOTE: VCT sample needed only once at start of the test. Mark this block as N/A after this.

Unit _____ Cycle _____

Control Rod Worth Measurement: Rod Swap
Enclosure 13.4
Critical Configuration and Worth Calculation Sheet

MA
Chow
6/21/16

Bank (x) No.	Bank (x) Ident.	Date	Time	$(h_x^M)_o$ (steps) Initial (step 12.2.1)	$(h_x^M)_o$ (steps) Return (step 12.2.9)	$(h_x^M)_o$ (steps) Average (step 12.4)	h_x^M (steps) (step 12.2.5)	★ $(\Delta\rho_1)_x$ (pcm) (step 12.5.1)	★ $\alpha_x(\Delta\rho_2)_x$ (pcm) (step 12.5.2)	★ W_x^I (pcm) (step 12.5.3)
1										
2										
3										
4										
5										
6										
7										
8										

★ NOTE: IF BANK BEING MEASURED HAS A WORTH GREATER THAN THE REFERENCE BANK WORTH, THESE VALUES WILL BE AS GIVEN IN STEPS 12.5.2 AND 12.5.3 AND ENCLOSURE 13.6

CHOW
MA
6/20/16

Recorded By ____________________ Date ________

Checked By ____________________ Date ________

Step 12.1.7 W_R^M = ________ pcm

Control Rod Worth Measurement: Rod Swap
Enclosure 13.5
Comparison of Inferred Bank Worths
With Design Predictions

Unit _____ Cycle _____
Date _______________

Bank (x)		++ W_x^I	* W_x^P	$(\varepsilon_1)_x$
No.	Ident.	(pcm)	(pcm)	(%)
reference		+		
1				
2				
3				
4				
5				
6				
7				
8				
		$\sum W_x^I$ (pcm)	$\sum W_x^P$ (pcm)	ε_2 (%)

tep 12.2.7: Record the measured worth of the reference bank here.
rom Enclosure 13.1
from Enclosure 13.4 Recorded By _______________ Date______

Checked By _______________ Date______

rom: LANGFORD.F.L (WES1974) Posted: Wed 10 pr-85 9:06 EST Sys 49
ject: Rod Swap Information for Mike Kitlan

n, please forward this to Mike Kitlan.

e,

s is to document our telecon on 4/9/85 on the actions to be taken n the test bank could be worthore than the reference bank for the Swap bank worth measurement. When this occurs, the following nts should be noted:

1) Do not change the reference bank designation.

2) Exchange the highest worth test bank last.

3) With the reference bank fully out and the test bank nearly fully inserted, measure the remaining worth of the test bank by one of two methods.

   a) Perform an "endpoint type" maneuver and insert the test bank from the critical position to zero steps and measure the reactivity worth using the reactivity computer.

   b) If the remaining worth of the test bank is larger than approximately 50 pcm, then dilute the test bank in from the critical position to zero steps and measure the reactivity worth using the reactivity computer. This will render the measurement of the just critical position of the reference bank alone after the swap N/A.

4) The worth of the test bank will be equal to the total worth of the reference bank plus the measured remaining worth of the test bank minus the worth of the reference bank from just critical to zero steps. Or in equation form:

$$WX = WR + WE - WRo$$

where: WX is the worth of the test bank,

WR is the total worth of the reference bank,

WE is the remaining worth of the test bank with the reference bank fully withdrawn, and

WRo is the worth of the reference bank from the just critical position to fully inserted (Delta-Rho-1 in the procedure).

Note that Alpha-x times Delta-Rho-2-x (procedure notation) is not used since Delta-Rho-2-x is zero.

opefully this meets your documentation requirements for this unique roblem. Also note that this was done at Zion last year without any roblems.

egards,
. R. Grobmyer
estinghouse NTD
uclear Operations

QUESTION 5

FOR INFORMATION ONLY

Attachment 4

Form 34731 (10-81)
(Formerly SPD-1002-1)

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: PT/0/A/4150/11A
Change(s) 0 to
0 Incorporated

(2) STATION: McGuire

(3) PROCEDURE TITLE: Control Rod Worth Measurement: Rod Swap

(4) PREPARED BY: Michael S. Kitlan Jr. DATE: 4/4/84

(5) REVIEWED BY: ______ DATE: ______

Cross-Disciplinary Review By: ______ N/R: ______

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: ______ Date: ______

(8) MISCELLANEOUS:

Reviewed/Approved By: ______ Date: ______

Reviewed/Approved By: ______ Date: ______

Form 34731 (10-81)
(Formerly SPD-1002-1)

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: PT/0/A/4150/11A
Change(s) 0 to
0 Incorporated

(2) STATION: McGuire

(3) PROCEDURE TITLE: Control Rod Worth Measurement: Rod Swap

(4) PREPARED BY: Michael S. Kitlan Jr DATE: 4/4/84

(5) REVIEWED BY: ______ DATE: ______

Cross-Disciplinary Review By: ______ N/R: ______

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: ______ Date: ______

(8) MISCELLANEOUS:

Reviewed/Approved By: ______ Date: ______

Reviewed/Approved By: ______ Date: ______

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) STATION: McGuire UNIT: 1 x 2 x 3 ____
OTHER:

(2) CHECK LIST APPLICABLE TO: ~~PT/0/A/4150/28~~ PT/0/A/4150/11 A
rsk

(3) SAFETY EVALUATION - PART A

The item to which this evaluation is applicable represents:

Yes ____ No X A change to the station or procedures as described in the FSAR or a test or experiment not described in the FSAR?

If the answer to the above is "Yes", attach a detailed description of the item being evaluated and an identification of the affected section(s) of the FSAR.

(4) SAFETY EVALUATION - PART B

Yes ____ No X Will this item require a change to the station Technical Specifications?

If the answer to the above is "Yes," identify the specification(s) affected and/or attach the applicable pages(s) with the change(s) indicated.

(5) SAFETY EVALUATION - PART C

As a result of the item to which this evaluation is applicable:

Yes ____ No X Will the probability of an accident previously evaluated in the FSAR be increased?

Yes ____ No X Will the consequences of an accident previously evaluated in the FSAR be increased?

Yes ____ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?

Yes ____ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No X May the possibility of malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

Yes ____ No X Will the margin of safety as defined in the bases to any Technical Specification be reduced?

If the answer to any of the preceding is "Yes", an unreviewed safety question is involved. Justify the conclusion that an unreviewed safety question is or is not involved. Attach additional pages as necessary.

(6) PREPARED BY: Michael S. Kitlan DATE: 4/4/84

(7) REVIEWED BY: ____ DATE: ____

(8) Page 1 of 1

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION
CONTROL ROD WORTH MEASUREMENT:
ROD SWAP

1.0 Purpose

1.1 To verify that the reactivity worth of the Reference RCC bank, as determined through reactivity computer measurement data, is consistent with design predictions.

NOTE: The reference RCC bank is the bank which has the predicted highest reactivity worth of all control and shutdown banks when inserted into an otherwise unrodded core.

1.2 To verify that the reactivity worth of each control and shutdown bank (except the reference bank), as inferred from data following iso-reactivity interchange with the reference bank, is consistent with design predictions.

2.0 References

2.1 Rod Bank Worth Measurements Utilizing Bank Exchange, WCAP-9863-A, May 1982.

2.2 Control Rod Worth Measurement, PT/0/A/4150/11

2.3 Post Refueling Controlling Procedure for Criticality, ZPPT, and Power Escalation Testing, PT/0/A/4150/21

2.4 Technical Specifications 3.4.1.1, 3.10.4, 3.10.3, and 3.10.2.

3.0 Time Required

3.1 8 hours, 1 engineer

4.0 Prerequisite Tests

tial/Date

__/____ 4.1 PT/0/A/4150/10, ARO Boron Endpoint Measurement

5.0 Test Equipment

5.1 Reactivity Computer (with flux signal from top and bottom of one power range channel).

5.2 Two two-pen strip chart recorders. One chart recorder should have reactivity (on a scale of 10 pcm/inch, with 0 pcm being the center of the recorder sheet), and T_{avg} from one loop (on a scale of 1°F/inch for 556 to 558°F set up on one side of the recorder sheet). The other chart recorder should have flux (on a scale of 0 to the top end of the testing decade in amps) and pressurizer level (on a scale of 10% level/inch). Chart speeds should be 1 inch/min.

NOTE: The specifications in this step may be altered by the Test Coordinator as necessary to accommodate equipment limitations, as long as all four signals are recorded or trended.

6.0 Limits and Precautions

6.1 The NC system temperature is controlled preferably by steam dump to the condenser. Temperature control may alternatively be affected by steam generator blowdown.

6.2 Normally all reactor coolant pumps should be operating for maximum mixing in the NCS. If all reactor coolant pumps are not operating, the operating pumps should be those on the NCS charging loops (A and/or D). See Tech Spec 3.4.1.1 and 3.10.4 if all reactor coolant pumps are not operating.

6.3 The rod insertion limit and bank overlap sequence will be violated during this test. The operators should be made aware in advance and should anticipate the associated alarms. Technical Specification 3.10.2 and 3.10.3 allows for this.

6.4 Maintain the flux level in the zero power test range established in Reference 2.4.

6.5 Prior to switching the rod control selector switch from one bank to another, verify both groups of the bank (if the bank has two groups) are at the same position in order to avoid group misalignment.

7.0 Required Unit Status

Initial/Date

____/____ 7.1 The unit is just critical in the Startup Mode (Mode 2) at zero power with the flux level in the zero power test range established in PT/0/A/4150/21, "Post Refueling Controlling Procedure for Criticality, ZPPT, and Power Escalation Testing."

____/____ 7.2 Record in the log the unit to which this test applies.

8.0 Prerequisite System Conditions

NOTE: The following steps may be signed off in any order.

____/____ 8.1 The reactor coolant system temperature is 557°F +1, -5°F.

NOTE: Maintain NCS temperature within ±1°F of established temperature during the test.

____/____ 8.2 The difference between NC loop, pressurizer, and VCT boron concentrations is less than 20 ppm. List on Enclosure 13.3.

NOTE: Do not use the boronometer.

____/____ 8.3 Xenon worth rate is changing less than ±.1 pcm/min.

____/____ 8.4 Test equipment is set up per Section 5.0.

____/____ 8.5 All available pressurizer heaters are on as needed, in order to improve mixing by maximizing the pressurizer spray.

____/____ 8.6 All control and shutdown banks are fully withdrawn except Control Bank D which is at a position greater than about 215 steps withdrawn.

____/____ 8.7 The Rod Control Selector switch is in Bank Select Mode set on Control Bank D.

____/____ 8.8 Complete Enclosure 13.1 with the predicted data. See Enclosure 13.9 for an explanation of nomenclature used in this test.

____/____ 8.9 Trend the points listed in Enclosure 13.2 every 15 minutes or less on the OAC.

9.0 Test Method

The RCC bank with the highest predicted value of reactivity worth is measured using the dilution technique per PT/0/A/4150/11. This bank serves as a reference. The integral worth of the remaining RCC banks is implied from the difference in the critical rod position of the reference bank with and without the insertion of bank being tested. The implied integral worths are then compared to predicted rod worths.

10.0 Data Required

10.1 The following conditions for the approximate time of criticality before and after each bank exchange, recorded on Enclosure 13.5:

Time

NCS Tavg

NCS Boron Concentration

Just critical height of reference bank

10.2 Nuclear design predictions on Enclosure 13.1.

10.3 Boron concentration information for the NCS and pressurizer on Enclosure 13.3.

10.4 A copy of the rod positions and rod worths for the reference bank from Enclosure 13.1 of PT/0/A/4150/11 when this test is complete.

10.5 The calculated, implied integral worth (W_x^I) for each RCC bank except the reference bank. List data on Enclosure 13.7.

10.6 The percent difference between inferred and predicted worths for each individual RCC banks (ε_1) and for the sum of all banks (ε_2) on Enclosure 13.8.

11.0 Acceptance Criteria

11.1 The absolute value of the percent difference between measured and predicted integral worth for the reference bank is ≦15% (from Enclosure 13.8 $(\varepsilon_1)_1 \leqq 15\%$).

11.2 From Enclosure 13.8, the calculated value $\varepsilon_2 \leqq 10\%$.

11.3 For all RCC banks other than the reference bank; either:

a) From Enclosure 13.8, $\varepsilon_{1_x} \leqq 30\%$ for each bank or

b) $W_x^I - W_x^P \leqq 200$ pcm for each bank, whichever is greater.

12.0 Procedure

Initial/Date

NOTE: See Enclosure 13.9 for an explanation of all nomenclature used in this test.

____/____ 12.1 Request NCS and Pressurizer samples to be taken at approximately 15-20 minute intervals until all banks are measured.

NOTE: The Boronometer may not be used for NC loop concentrations.

NOTE: Notify Chemistry that the unused portions of the samples should be retained in appropriately labeled containers, for possible future re-analysis, until all acceptance criteria are met or as specified by the test coordinator.

12.2 Measure the integral reactivity worth of the reference bank as follows:

NOTE: The reference bank is defined as that bank which is predicted to have the highest worth, of all control and shutdown banks, when inserted into an otherwise un-rodded core (see Enclosure 13.1 for the identity of this bank). In this procedure, the banks will be referred to by the bank number, the reference bank being number 1. If the reference bank is currently positioned at less than 228 steps withdrawn (i.e., if it is Control Bank D), continue with step 12.2.5. Mark steps 12.2.1 to 12.2.4 NA. If the reference bank is positioned at 228 steps withdrawn, continue on at Step 12.2.1.

____/____ 12.2.1 Insert the reference bank 1 until the indicated reactivity is approximately -10 pcm.

____/____ 12.2.2 Withdraw the bank inserted below 228 until the indicated reactivity is approximately +10 pcm.

____/____ 12.2.3 Repeat steps 12.2.1 and 12.2.2 until the previously inserted bank is fully withdrawn.

____/____ 12.2.4 Adjust the position of the reference bank 1 until the reactor is just critical. Record this position in the test log.

____/____ 12.2.5 Perform Control Rod Worth Measurement per PT/0/A/4150/11 on the reference Bank 1.

___/___ 12.2.6 Attach a completed copy of PT/0/A/4150/11 Enclosure 13.1 to this procedure.

___/___ 12.2.7 Record the total reference bank rod worth from PT/0/A/4150/11 Enclosure 13.1 on Enclosure 13.7 and 13.8 as shown.

___/___ 12.2.8 Ensure the reactor is critical at the same reference bank position as was obtained at the end of PT/0/A/4150/11.

12.4 Measure the reactivity worth of the remaining control and shutdown banks, relative to the reference Bank 1, as follows:

NOTE: The relative worth of each RCC bank is obtained from the critical position of the reference bank (initially nearly fully inserted) after full insertion of the bank being measured (initially fully withdrawn), at constant RCS boron concentration.

NOTE: Perform rod swap measurements on Control Bank D last if possible.

___/___ 12.4.1 Record the initial critical bank configuration on Enclosure 13.4 and 13.5 for the reference bank. Also record the initial NC boron concentration and average T_{avg} on Enclosure 13.5.

(√) ___ ___ ___ ___ ___
3 4 5 6 7
___ ___
8 9

___/___ 12.4.2 Insert bank 2 (identify this bank on top of Enclosure 13.5; i.e., Bank 2 is S/D E or Cont. B, etc.) until the reactivity indicated by the reactivity computer is approximately -20 pcm.

(√) ___ ___ ___ ___ ___
3 4 5 6 7
___ ___
8 9

___/___ 12.4.3 Withdraw the reference Bank 1 until the indicated reactivity is approximately +20 pcm.

NOTE: Maintain the flux within the zero power test range established in Reference 2.4.

(√) ___ ___ ___ ___ ___
3 4 5 6 7
___ ___
8 9

____/____ (√) ___ ___ ___ ___ ___ 3 4 5 6 7 ___ ___ 8 9	12.4.4	Repeat Steps 12.4.2 and 12.4.3 until bank 2 is fully inserted. Keep the indicated reactivity within ±20 pcm.
____/____ (√) ___ ___ ___ ___ ___ 3 4 5 6 7 ___ ___ 8 9	12.4.5	Adjust the position of reference bank 1 until the reactor is just critical. Record the final critical configuration data on Enclosure 13.5. Also record the final NC Boron Concentration and average T_{avg} on Enclosure 13.5. NOTE: If time permits, measure the differential reactivity worth of reference Bank 1 with the reactivity computer by sequential bank insertions and withdrawals around the critical position. Record information in the test log if this is performed. (Analysis may be performed at a later time.)
____/____ (√) ___ ___ ___ ___ ___ 3 4 5 6 7 ___ ___ 8 9	12.4.6	Insert the reference Bank 1 until the indicated reactivity is approximately -20 pcm.
____/____ (√) ___ ___ ___ ___ ___ 3 4 5 6 7 ___ ___ 8 9	12.4.7	Withdraw bank 2 until the indicated reactivity is approximately +20 pcm.
____/____ (√) ___ ___ ___ ___ ___ 3 4 5 6 7 ___ ___ 8 9	12.4.8	Repeat Steps 12.4.6 and 12.4.7 until bank 2 is fully withdrawn.
____/____ (√) ___ ___ ___ ___ ___ 3 4 5 6 7 ___ ___ 8 9	12.4.9	Adjust the position of reference Bank 1 until the reactor is just critical. Record the critical configuration data on Enclosures 13.4 and 13.5.

__/___ 12.4.10 Repeat Steps 12.4.2 through 12.4.9 for the remaining, unmeasured control and shutdown banks numbered 3 through 9 instead of bank 2. The banks may be measured in any order except that Control Bank D should be measured last. Identify the bank beside the bank number on Enclosures 13.4, 13.6, 13.7 and 13.8.

NOTE: If a Control Bank D-in ITC measurement is to be made, perform Steps 12.5, 12.5.1 and 12.5.2. If not, proceed directly to Step 12.5.3 and mark Steps 12.5, 12.5.1 and 12.5.2 as N/A.

__/___ 12.5 After all rod measurements have been made, again swap Control Bank D for the reference bank 1.

__/___ 12.5.1 By NC Boron Adjustment, reposition Control Bank D and the reference Bank 1 such that Control Bank D is almost fully inserted into the core and the reference bank 1 fully withdrawn from the core. (It is acceptable to have Control Bank D fully inserted and the reference bank 1 almost fully withdrawn.)

__/___ 12.5.2 Perform PT/0/A/4150/12B, Moderator Temperature Coefficient of Reactivity During Startup Mode.

__/___ 12.5.3 By NC boron adjustment, reposition control and shutdown banks to the desired normal operating configuration of Control Bank D at about 215 steps withdrawn. Do not go out of Bank Control.

12.6 Perform the following steps once Control Bank D is about 215 steps withdrawn.

__/___ 12.6.1 Go to the Master Cycler Cabinet and reset the Bank Overlap Digital Counter to 000 by pushing the reset button.

__/___ 12.6.2 Reset the Bank Overlap Counter to 345 plus the present Control Bank D position by pushing the button to count up from 000 to the desired value (one push of the button is one digit change on the display).

__/___ 12.6.3 Place rod control to manual.

NOTE: This completes the data acquisition section of this test.

__/___ 12.7 If boron samples are no longer needed to be gathered, notify Chemistry.

12.8 Compute the average reference bank critical position on Enclosure 13.4.

12.9 Compute the inferred worth for each control and shutdown bank (except the reference bank 1) as follows:

12.9.1 Using the data from Enclosure 13.4, and the worth measurement data for the reference bank from Enclosure 13.1 of PT/0/A/4150/11, compute the value of $(\Delta\rho_1)_x$ as described below.

$$(\Delta\rho_1)_x = \left[W_R^M \right]_o^{(h_x^M)_o \text{ avg}}$$

where:

$\left[W_R^M \right]_o^{(h_x^M)_o \text{ avg}}$ is the measured integral worth of the reference bank from 0 steps to $(h_x^M)_o$ avg from Enclosure 13.1 of PT/0/A/4150/11.

and $(h_x^M)_o$ avg is the average of the initial and return critical positions of the reference bank before and after interchange with bank x as given on Enclosure 13.4.

Fill in all blanks and complete the calculations on Enclosure 13.4 in the appropriate column.

/____ 12.9.2 Using the data from Enclosure 13.5, the worth measurement data for the reference bank from Enclosure 13.1 of PT/0/A/4150/11 and the design data of Enclosure 13.1, compute the value of $\alpha_x\ (\Delta\rho_2)_x$ as described below:

$$\alpha_x(\Delta\rho_2)_x = \alpha_x \left[W_R^M \right]_{h_x^M}^{228}$$

where: $\left[W_R^M \right]_{h_x^M}^{228}$ is the measured integral worth of the reference bank from h_x^M to the fully withdrawn position from PT/0/A/4150/11, Enclosure 13.1.

h_x^M is the measured critical position of the reference bank after interchange with bank x from Enclosure 13.5.

and

α_x is a correction factor from Enclosure 13.1 to account for the influence of bank x on the worth of the reference bank.

Fill in all blanks and complete the calculations on Enclosure 13.6.

___/____ 12.9.3 Compute the inferred integral worth of each bank x, W_x^I, as indicated on Enclosure 13.7.

_/___ 12.9.4 Compute the percent difference between inferred and predicted worths for each individual RCC bank and the sum of all banks described below.

$$(\varepsilon_1)_x = \left[\frac{W_x^I - W_x^P}{W_x^P}\right] \times 100, \text{ in } \%$$

$$\varepsilon_2 = \left[\frac{\sum_{i=1}^{N} W_i^I - \sum_{i=1}^{N} W_i^P}{\sum_{i=1}^{N} W_i^P}\right] \times 100, \text{ in } \%$$

Fill in all blanks and summarize the calculations on Enclosure 13.8.

_/___ 12.10 Verify all acceptance criteria have been met.

_/___ 12.11 Inform Chemistry to discard the Chemistry samples they have saved, once all results of this test are acceptable.

13.0 <u>Enclosures</u>

13.1 Nuclear Design Predictions for Rod Interchange Measurements

13.2 PAO Data

13.3 Log of Boron Concentrations

13.4 Calculation of $(\Delta\rho_1)x$

13.5 Critical Configuration Data

13.6 Calculation of $\alpha_x(\Delta\rho_2)x$

13.7 Calculation of Inferred Integral Bank Worths

13.8 Comparison of Inferred Bank Worths with Design Predictions

13.9 Nomenclature

Control Rod Worth Measurement: Rod Swap
Enclosure 13.1
Nuclear Design Predictions
for Rod Interchange Measurements

McGuire Unit _____ Cycle _____

Bank No. (x)	Bank Identity +	W_x^P (pcm)	(b) h_x^P (steps)	(c) α_x
(a) 1 (Reference)				
2				
3				
4				
5				
6				
7				
8				
9				

(a) Reference bank - the bank with the highest predicted integral worth.

(b) Reference bank critical position after interchange with bank x.

(c) Ratio of integral worth of the reference bank from h_x^P to the fully withdrawn position with and without x in the core.

\+ Control Bank C, Shutdown Bank E, etc.

NOTE: See Enclosure 13.9 for a complete listing of nomenclature used in this test.

Recorded By ______________________ Date ____________________
This data came from (list source): ________________________________
____

Control Rod Worth Measurement: Rod Swap
Enclosure 13.2
PAO Data

A0821	Boric Acid Makeup Blended Flow
P1390	Control Bank A Position
P1391	Control Bank B Position
P1392	Control Bank C Position
P1393	Control Bank D Position
P1546	Shtudown Bank A Position
P1547	Shutdown Bank B Position
P1548	Shutdown Bank C Position
P1549	Shutdown Bank D Position
P1550	Shutdown Bank E Position
A0632	Intermediate Range Channel N35
A0633	Intermediate Range Channel N36
A0819	Loop A T_{avg}
A0825	Loop B T_{avg}
A0831	Loop C T_{avg}
A0837	Loop D T_{avg}
A1058	Loop A ΔT
A1070	Loop B ΔT
A1082	Loop C ΔT
A1094	Loop D ΔT
A1118	Pressurizer Pressure
A0602	Boronometer
A1124	Pressurizer Level
P1461	NC Avg. T_{avg}
P0828	Avg. Incore T/C Temperature
A0603	Boric Acid Flow to Blender
A1064	NC Loop A NR Cold Leg Temperature
A1076	NC Loop B NR Cold Leg Temperature
A1088	NC Loop C NR Cold Leg Temperature
A1100	NC Loop D NR Cold Leg Temperature

Control Rod Worth Measurement: Rod Swap
Enclosure 13.3
Log of Boron Concentrations

Date	Time Sample Taken	NCS Boron Concentrations			Comments	Recorded By
		+VCT	NCS	Press.		

McGuire Unit ___________
Cycle _____

NOTE: VCT sample needed only once at start of the test. Mark this block as N/A after this.

Control Rod Worth Measurement: Rod Swap
Enclosure 13.4
Calculation of $(\Delta\rho_1)_x$

Unit _____ Cycle _____

Date ____________________

Bank (x)		$(h^M_x)_o$ (steps)			++ $(\Delta\rho_1)_x$
No.	Ident.	+Initial	*Return	**Average	(pcm)
2					
3					
4					
5					
6					
7					
8					
9					

+Step 12.4.1 - reference bank initial critical position.

***Step 12.4.9 - reference bank final critical position upon exchange with bank x (bank x if out of core).**

++Step 12.9.1 ****Step 12.8**

Recorded By ____________________

____

nit ______
ycle ______
ate ______________

Control Rod Worth Measurement: Rod Swap
Enclosure 13.5
Critical Configuration Data

Time (hrs.)	NC T_{avg} (°F)	NC Boron Conc. (ppm)	Reference Bank Position (steps) $(h_x^M)_o$	Reference Bank Position (steps) (h_x^M)	RCC Bank Positions No. 2 ( )	No. 3 ( )	No. 4 ( )	No. 5 ( )	No. 6 ( )	No. 7 ( )	No. 8 ( )	No. 9 ( )
			+	N/A	228	228	228	228	228	228	228	228
			N/A	*	0	228	228	228	228	228	228	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	0	228	228	228	228	228	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	228	0	228	228	228	228	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	228	228	0	228	228	228	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	228	228	228	0	228	228	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	228	228	228	228	0	228	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	228	228	228	228	228	0	228
			**	N/A	228	228	228	228	228	228	228	228
			N/A	*	228	228	228	228	228	228	228	0
			**	N/A	228	228	228	228	228	228	228	228

Step 12.4.1 - initial critical bank position   *Step 12.4.5 - final critical bank position   **Step 12.4.9

ecorded By ______________________

Control Rod Worth Measurement: Rod Swap
Enclosure 13.6
Calculation of $\alpha_x(\Delta\rho_2)_x$

Unit _____ Cycle _____

Date ________________

Bank (x)		$^{+}h_x^M$	(1) $\left[W_R^M\right]_{h_x^M}^{228}$	(2) $^{*}\alpha_x$	(1) x (2) $\alpha_x(\Delta\rho_2)_x$
No.	Ident.	steps	(pcm)		(pcm)
2					
3					
4					
5					
6					
7					
8					
9					

+from Enclosure 13.5

*from Enclosure 13.1

Recorded By ________________

____

Control Rod Worth Measurement: Rod Swap
Enclosure 13.7
Calculation of Inferred Integral Bank Worths

Unit _____ Cycle _____ Step 12.2.7 W_R^M = _________ pcm
Date ______________

Bank (x)		$^{+}(\Delta\rho_1)_x$	$^{*}\alpha_x(\Delta\rho_2)_x$	W_x^I (a)
No.	Ident.	(pcm)	(pcm)	(pcm)
2				
3				
4				
5				
6				
7				
8				
9				

(a) $W^I = W_R^M - (\Delta\rho_1)_x - \alpha_x(\Delta\rho_2)_x$
+from Enclosure 13.4

*from Enclosure 13.6

Recorded by ______________________

Control Rod Worth Measurement: Rod Swap
Enclosure 13.8
Comparison of Inferred Bank Worths
With Design Predictions

Unit _____ Cycle _____
Date ______________

Bank (x)		++ W^I_x	* W^P_x	$(\varepsilon_1)_x$
No.	Ident.	(pcm)	(pcm)	(%)
1 reference		+		
2				
3				
4				
5				
6				
7				
8				
9				

$\sum W^I_x$ (pcm)	$\sum W^P_x$ (pcm)	ε_2 (%)

+Step 12.2.7: Record the measured worth of the reference bank here.

*from Enclosure 13.1

++from Enclosure 13.7

Recorded By ______________________

____

APPENDIX B

Original Issue NRC SER

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

MAY 22 1987

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

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

Dear Mr. Tucker:

Subject: ROD SWAP METHODOLOGY REPORT FOR STARTUP PHYSICS TESTING, MCGUIRE AND CATAWBA NUCLEAR STATIONS, UNITS 1 AND 2 (TACs 62981, 62982, 62983, 62984)

By letter dated December 4, 1986, you submitted a report titled "Rod Swap Methodology Report for Startup Physics Testing," and you submitted additional information by letters dated February 11 and March 11, 1987. In addition, telephone discussions were held on May 1, 1987 with members of your company regarding conditions associated with our approval.

We have reviewed the material submitted and find the rod swap methodology as described to be acceptable for rod worth measurement of reloaded cores for McGuire and Catawba Stations, Units 1 and 2. This approval recognizes your prior agreement to certain conditions listed in our enclosed Safety Evaluation Report.

Should you have any questions regarding the enclosure, contact me at (301) 492-8961 or K. Jabbour at (301) 492-7367. In any future correspondence regarding this approval, please include a reference to TACs 62981, 62982, 62983 and 62984.

Sincerely,

Darl Hood

Darl Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

Enclosure: Safety Evaluation Report

cc w/encl
See next page

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

Enclosure

SAFETY EVALUATION REPORT

FOR DUKE POWER COMPANY'S

"ROD SWAP METHODOLOGY REPORT FOR STARTUP PHYSICS TESTING"

Introduction

Duke Power Company (the licensee) submitted a report titled "Rod Swap Methodology Report for Startup Physics Testing" on December 4, 1986. Answers to NRC questions and additional information were submitted by letters dated February 11, 1987 (Ref. 2) and March 11, 1987 (Ref. 3). The report describes the rod swap methodology which Duke Power Company would like to use for rod worth measurement for the McGuire 1 and 2 and the Catawba 1 and 2 units after each reload. While the rod swap technique has been used on Duke plants in the past, the methodology was the Westinghouse methodology which NRC approved on May 28, 1983. Due to the complexities of Rod Swap, the May 28, 1983 approval stated that the method was approved for use by Westinghouse only. Thus, it is necessary for Duke to obtain NRC approval before using the Duke Rod Swap methodology.

Background

The reactivity worth of the control rods is measured at the beginning of each cycle. Rod worth measurements are made in order to verify shutdown margin. The measurement conditions are not those used in the accident analysis but comparison of measurement and predicted rod worths for a known set of conditions gives assurance that rod worths and the shutdown margin predicted for the worst conditions are accurate. For reload cores, usually, not all rod banks are measured. Normally, the control banks (approximately 4 banks, worth about half the total worth) are measured.

The traditional method of rod worth measurement is by boron dilution. Starting from an all rods out critical configuration, the bank is inserted a few steps at a time and the reactor is kept critical by diluting the boron concentration. One control bank would be inserted until it is all the way in and then the next bank would be started. A reactivity computer is also used to measure the reactivity change at each position. The reactivity worth of the bank is the sum of all the reactivity changes recorded by the reactivity computer. The worth of the bank is also equal to the difference in boron concentrations from the bank fully withdrawn to fully inserted positions.

Several years ago an alternative method of rod worth measurement called rod swap or rod exchange was proposed. In this method the highest worth bank, called the reference bank, is measured by boron dilution and remaining banks, called test banks, are measured by "swapping" the test bank with the reference bank. The critical position of each measurement is the reference bank position when the test bank is fully inserted. This method is an indirect method in that it does not measure the worth of banks in combination (i.e. banks D + C + B + A). Rod Swap does have some advantages over boron dilution, however. It does not require the large change in boron concentration and subsequent processing of thousands of gallons of water. It is less time consuming and thus all banks can be measured in much less time than it would take to measure one half the banks by boron dilution.

Evaluation

The Duke Report presents a minimal description of the methodology and a comparison of calculated and inferred worths for several cycles on McGuire 1 and 2. Additional information supplied more details of the procedure. The Duke methodology is very similar to the methodology NRC approved for use by Westinghouse. Duke will use previously approved physics codes and methodologies as described in Reference 4 for the calculations of rod worths and critical heights.

As verification of the methodology, Duke supplied rod swap data for 5 cycles (McGuire Unit 1, Cycles 2, 3 and and 4, McGuire Unit 2, Cycles 2 and 3). This data compares measured and predicted worth for each bank. In addition we have made comparisons of this data with that presented in the Startup Reports for these cycles. (This data is different since Westinghouse did the calculations for these cycles). Examination of the data reveals that the greatest deviation on any one bank was 103 pcm or 24% on a small bank. The greatest deviation on the total worth was 6.9% for Unit 2, Cycle 2. The average total difference was 4.94% which compares favorably with the 6.38% for the Westinghouse predictions.

While for some of the McGuire data the difference between measurement and prediction is greater than usually seen, it is still within the acceptable range. Duke did not perform a side-by-side comparison of boron dilution and rod swap for the same cycle. However, Duke supplied data from the initial startup of Catawba 1 and 2, Catawba 1 using boron dilution and Catawba 2 using Rod Swap. The cores are essentially identical as confirmed by as built parameters and other physics test measurements. The rod worth measurements were within acceptable limits.

Based on our review of the material submitted, we find the rod swap methodology as proposed by Duke Power Company to be acceptable subject to the following conditions, to which Duke Power Company has agreed:

1) The boron dilution rate for measurement of the reference bank shall not exceed 500 pcm.

2) All banks, both control and shutdown banks, must be measured.

3) The review criteria are:

   A. The absolute value of the percent difference between measured and predicted integral worth for the reference bank is ≤ 10 percent.

   B. For all banks other than the reference bank, either (whichever is greater);

      1) the absolute value of the percent difference between inferred and predicted integral worths is ≤ 15 percent or

2) the absolute value of the reactivity difference between inferred and predicted integral worths is $\leq$ 100 pcm.*

C. The sum of the measured/inferred worth of all the rods must be $\leq$ 110 percent of the predicted worth.

4) The acceptance criteria are:

(1) The sum of the measured/inferred worth of all the rods must be $>$ 90 percent of the predicted rod worth.

(2) For all banks other than the reference bank, either (whichever is greater)
a) the absolute value of the percent difference between inferred and predicted integral worth is $<$ 30 percent or
b) the absolute value of the reactivity difference between inferred and predicted integral worths is $<$ 200 pcm.

(3) The absolute value of the percent difference between measured and predicted integral worth for the reference bank is $<$ 15 percent.

5) Additional testing is required if the reference bank boron concentrations and reactivity computer worth do not agree. Remedial action for failure of an acceptance or review criterion require . investigation and solution within 30 days (for acceptance criterion) or 60 days (for review criterion). The licensee must then submit a report of the findings to the NRC within 45 days of the test (for acceptance criterion) or within 75 days of the test (for review criterion).

---

*A pcm is equal to 10^{-5} $\Delta k/k$.

REFERENCES

1) Letter, H. B. Tucker, Duke Power Company, to Harold R. Denton, NRC, December 4, 1986.

2) Letter, H. B. Tucker, Duke Power Company, to Nuclear Regulatory Commission, Document Control Desk, dated February 11, 1987.

3) Letter, H. B. Tucker, Duke Power Company, to Nuclear Regulatory Commission, Document Control Desk, dated March 11, 1987.

4) Duke Power Company, "Nuclear Physics Methodology for Reload Design," DPC-NF-2010A, June 1985.

Mr. H. B. Tucker
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Catawba Nuclear Station

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