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Michael S. Tuckman
Executive Vice President
Nuclear Generation

August 7, 2002

U. S. Nuclear Regulatory Commission
Washington D.C. 20555-0001
ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 370

Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 414

Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343, and MB3344) and License Amendment Request Supplement

This purpose of this letter is to provide Duke Energy Corporation's (Duke) response to an NRC request for additional information (RAI) and to supplement a Duke license amendment request (LAR) previously submitted pursuant to 10CFR50.90. Please note that some of the information contained in this submittal package has been determined to be proprietary and is being submitted pursuant to 10CFR2.790. This proprietary information is discussed below.

Duke submitted[1] a LAR applicable to McGuire and Catawba Technical Specifications (TS) 5.6.5.a and 5.6.5.b. Also included in this submittal were proposed revisions to the four Duke Topical Reports listed below.

[1] Reference 1: Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated October 7, 2001, SUBJECT: License Amendment Request Applicable to Technical Specification 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003

APO1

- DPC-NE-2009-P, *Duke Power Company Westinghouse Fuel Transition Report*, Revision 1;

- DPC-NF-2010, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*, Revision 1;

- DPC-NE-2011-P, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, Revision 1;

- DPC-NE-1003, McGuire Nuclear Station and Catawba Nuclear Station *Rod Swap Methodology Report for Startup Physics Testing*, Revision 1.

The NRC RAI[2] asked questions on these topical reports. As described below, the Duke responses to these questions are included in the attachments to this letter.

In a subsequent submittal,[3] Duke proposed another LAR for McGuire and Catawba TS 5.6.5, but this LAR was only applicable to TS 5.6.5.b. The information contained herein explains the necessary coordination for changing TS 5.6.5.b for McGuire and Catawba. This LAR implements the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler.[4] The NRC has approved and issued this LAR for both McGuire[5] and Catawba.[6] Implementation of the

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[2] Reference 2: Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation, Dated June 26, 2002, SUBJECT: Request for Additional Information, Application for Changes to Technical Specifications (TAC Nos. MB3222, MB3223, MB3343, and MB3344

[3] Reference 3, Letter, Duke Energy Corporation to U.S. Nuclear Regulatory Commission, ATTENTION: Document Control Desk, Dated December 20, 2001, SUBJECT: License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report – Oconee, McGuire, and Catawba Technical Specification 5.6.5

[4] TSTF-363, "Revise Topical Report References in ITS 5.6.5 CCLR"

[5] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 10, 2002, SUBJECT: McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3702 and MB3703)

[6] Letter, U. S. Nuclear Regulatory Commission to Duke Energy Corporation Dated July 2, 2002, SUBJECT: Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3728 and MB3729)

referenced industry traveler eliminates the need for the changes Duke proposed to McGuire and Catawba TS 5.6.5.b in Reference 1. The LAR supplement transmitted herein deletes the proposed changes to McGuire and Catawba TS 5.6.5.b contained in Reference 1. The attached McGuire and Catawba TS pages (both marked and reprinted versions) update Reference 1 such that it contains the latest approved version of the affected TS pages and only applies to McGuire and Catawba TS 5.6.5.a. The affected TS pages are:

> McGuire Units 1 and 2 Pages: 5.6-2, 5.6-3, B3.2.1-11, and B3.2.3-4; and
>
> Catawba Units 1 and 2 Pages: 5.6-3, B3.2.1-11, and B3.2.3-4.

As shown, conforming Bases changes have been made and the necessary Bases pages are also included.

The attachments to this letter are listed and described below.

- Attachment 1 provides the Duke response to the NRC's general questions on Topical Reports DPC-NF-2010 and DPC-NE-2011-P.

- Attachment 2 provides the Duke response to the NRC's specific questions on Topical Report DPC-NF-2010.

- Attachments 3a and 3b provide the Duke responses to the NRC's specific questions on Topical Report DPC-NE-2011-P. Attachment 3a is the proprietary version and Attachment 3b is the non-proprietary version.

- Attachment 4 provides the Duke response to the NRC's specific questions on Topical Report DPC-NE-1003.

- Attachment 5 provides the Duke response to an NRC concern on Topical Report DPC-NE-2009-P. This concern was not included in the NRC's RAI,[2] however it was discussed during an NRC/Duke telephone conference held on July 24, 2002.

- Attachments 6a and 6b provide a marked copy of the existing approved Technical Specifications pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively. These marked copies show the proposed changes.

- Attachments 7a and 7b provide the reprinted Technical Specifications and Bases pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively.

Duke has determined that the revisions contained in this LAR supplement, as shown in Attachments 6a, 6b, 7a, and 7b have no impact on the determination of no significant hazards consideration that was included in Reference 1.

This submittal package contains information that Duke considers proprietary. This information is contained within the proprietary version of the response to the NRC questions on Topical Report DPC-NE-2011-P that is provided as Attachment 3a to this letter. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. An affidavit that attests to the proprietary nature of this information is included with this letter. A non-proprietary version of this response is also provided as Attachment 3b to this letter.

Inquiries on this matter should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

xc w/Attachments:

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M. S. Tuckman, affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: August 7, 2002
Date

Mary P. Nelms, Notary Public

My commission expires: JAN 22, 2006

SEAL

bxc w/Attachments:

M. T. Cash
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MNS Master File - MG01DM
Catawba Master File - CN04DM
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Catawba Owners:
Saluda River Electric Corporation
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Laurens, SC 29360-0929

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Attachment 1

Responses to Request for Additional Information

Topical Reports Numbered DPC-NE-2011-P, Revision 1, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors and DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design (TAC NOS. MB3343, MB3344, MB3222, MB3223)

General

Subsequent to receiving the NRC RAI package, a clarification of Questions 1, 2, and 3 was obtained from the NRC during a conference call on Thursday July 18, 2002. Responses to all questions in the NRC RAI are given below, and responses to Questions 1, 2, and 3 take into account the clarification received from the NRC.

Question 1. Please provide a detailed qualitative technical justification for the requested changes to the topical reports (methodologies), DPC-NE-2011 and DPC-NF-2010. (i.e., why are these changes being made?).

Response

Subsequent to the approval of the current version of these reports, there have been various changes in calculation methods and plant operating philosophy. Therefore, sections of these topical reports affected by these changes have been reviewed and updated to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. These revisions do not change approved methods nor introduce new methods. These changes and justifications were identified and described in the October 7, 2001 DEC submittal.

Question 2. To expedite the review process, please provide a qualitative and quantitative technical basis for each of the changes in the above stated topical reports.

Response

Qualitative and quantitative bases for each change to DPC-NF-2010 and DPC-NE-2011-P are provided in Attachments 7a and 8a, respectively in the License Amendment Request package submitted by Duke with a cover letter date of October 7, 2001.

Question 3. Please provide validation data, bench-marking the results of comparisons between the old and the new models (changes).

Response

These revisions do not change approved methods nor introduce new methods; therefore, additional benchmarking is not necessary.

Question 4. If the changes to these topical reports/methodologies impact the safe operation of the reactor core, please provide the safety significance (impact) of each of these changes?

Response

The methodology changes correspond to previously approved methodologies or licensing basis documents, or to administrative non-technical changes. Therefore, these changes do not impact the safe operation of the reactor core.

Question 5. Please provide the basis as to why the proposed changes to the above stated topical reports should be found acceptable.

Response
The purpose for these changes is to maintain the topical reports in a condition that is consistent with other current, NRC approved licensing related documents and to improve clarity and continuity in order to avoid ambiguities and inconsistencies that could be misconstrued. The changes do not change previously approved methodologies.

Attachment 2
Responses to Request for Additional Information
Topical Report Numbered DPC-NF-2010, Revision 1, Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design
(TAC NOS. MB3343, MB3344, MB3222, MB3223)

Question 1. In the revision history section on page ii, the licensee provides the staff with the reason for the submittal. Since this is a licensing action, please list/Tabulate what Technical Specification(s), Basis, FSAR, conformance to regulatory documents, criteria, generic letters, etc., etc. are impacted by the request for these changes within the licensing framework?

Response
The impact to licensing basis documents by changes made to DPC-NF-2010 is described below.

- Technical Specifications and Bases: TS 5.6.5.b

No Technical Specification or Bases requires a change as a result of these revisions. Even the Licensing Amendment Request to change Technical Specification 5.6.5b for this proposed topical report revision is no longer required (see the License Amendment Request to implement the provisions of an NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Traveler (TSTF 363, "Revise Topical Report References in ITS 5.6.5 COLR")).

- UFSAR Sections: 1.6.3, 4.3, and 15.0
- Topical Reports: DPC-NE-1004, DPC-NE-1003, DPC-NE-2004P, DPC-NE-2007P DPC-NE-2009P, DPC-NE-3001P

These documents contain general references to the methods contained in the proposed topical report. Changes to these documents are expected to be made as part of the normal UFSAR and Topical Report update processes.

Question 2. Section 4.2.4.2, second paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Cases are run with the moderator temperature at 5 °F above and at the reference temperatures."

Proposed Sentence: "Cases are run changing the moderator temperature from the reference temperature."

The original sentence may imply that the calculation of the moderator temperature coefficient will be performed by only changing the moderator temperature +5 °F. Whereas, these calculations may be more appropriately performed using a -5 °F change, using an average of the +5 and -5 °F results, or using a different temperature change depending on actual plant conditions. Therefore, specificity is removed to reflect that calculations are performed to match plant conditions or intended use of the data.

Question 3. In Attachment 7a–Detailed Listing of the Changes to DPC-NF2010A, it is stated in many places, that "this change is made to avoid difficulties with the literal interpretation of the original description". Please provide clarification of this statement with a supporting example.

Response
Changes documented in Attachment 7a which state "this change is made to avoid difficulties with the literal interpretation of the original description" also provide additional information about the reason why the literal interpretation could potentially be misconstrued. Changes with this statement can be categorized into 3 types: (1) descriptions of plant operations, (2) descriptions of calculations, and (3) administrative. An example within each category is provided below.

Descriptions of Plant Operations
Example: Change #3

Section 1.1, First Paragraph
Description: Changed the third sentence to give examples of intervals between refueling outages.
Justification: The original sentence implies a maximum fuel cycle length of 18 months, and possible fuel cycle lengths are not limited to 18 months. This change is made to avoid difficulties with the literal interpretation of the original description.

The current version states: "Refueling occurs at intervals of 6 to 18 months, depending on the utility's operational requirements."

The proposed version states: "Refueling occurs at intervals appropriate for the power production needed, for example 12, 18, or 24 months."

A literal interpretation of the current version may imply that development of a core design is limited to a 6 to 18 month fuel cycle, whereas current core designs may be different from the exact range of 6 to 18 months.

Descriptions of Calculations
Example: Change #32

Section 4.2.1, Third Paragraph
Description: Clarified the first sentence.
Justification: Depletion model statepoints may be specified in MWD/MTU or EFPD and may be different than those listed. This change is made to avoid difficulties with the literal interpretation of the original description.

The current version states: "The cycle is then depleted in steps corresponding to 0, 150, 500, 1000, 2000, 4000 ... MWD/MTU to verify that power peaking versus burnup remains acceptable."

The proposed version states: "The cycle is then depleted to various times in the cycle to verify that power peaking versus burnup remains acceptable."

A literal interpretation of the current version may imply that core depletions would have to be performed at the burnup statepoints listed, using MWD/MTU units, and at specific burnup intervals. Current core depletions may use a different set of burnup statepoints and intervals depending on fuel and burnable poison depletion effects. Also, burnup statepoints may be specified in units other than MWD/MTU (for example EFPD).

Administrative
Example: Change #104

Section 9.1.2, First Paragraph
Description: Changed the last sentence for clarity.
Justification: This change is made to avoid difficulties with the literal interpretation of the original description. Equilibrium xenon worth data may be shown in plot or table format.

The current version states: "The results are displayed in a format similar to Figure 9-4."

The proposed version states: "Figure 9-4 shows the results of a typical equilibrium xenon worth calculation."

A literal interpretation of the current version may imply that equilibrium xenon worth calculation results would be displayed in a plot format to be used in startup test predictions and core physics parameters. However, it is also acceptable to provide this information in a table or electronic database.

Question 4. Section 4.2.4.4, fifth paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Then a second EPRI-NODE case is run with the core power level reduced 5% while holding everything else constant."

Proposed Sentence: "Then a second case is run with the core power reduced while holding control rods, boron, and xenon constant."

The original sentence may imply that the calculation of the power coefficient will be performed by changing the core power -5%. Whereas, these calculations may be more appropriately performed using a different power reduction or increase depending on actual plant conditions. Therefore, specificity is removed to reflect that calculations are performed to match plant conditions or intended use of the data. By removing the reference to the core simulator, the implication is made that any NRC approved model may be used. Finally, the revised sentence removes the ambiguity of the statement "everything else".

Question 5. Section 8.1, first paragraph. Is the added equation the same as that in the current version of the DPC-NF-2010A topical? If not, please provide technical justification for its use.

Response
The equation is in the current approved version of DPC-NF-2010. This equation is located in Section 6.2.1.2 (Page 6-2) of the current version and is labeled Equation "6-1". Section 6 of the proposed version was rewritten for reasons explained in Attachment 7a of the Licensing Amendment Request Package dated October 7, 2001.

Section 6 was rewritten, because subsequent to the initial NRC approval of this topical report, methods for performing safety related calculations were approved by the NRC in References 1, 2, and 3 (below). The NRC excluded Section 6.3 when the NRC SER of the original version of this report was issued. The rewrite of this section references safety analysis methods approved by the NRC (References 1 and 2, below) and provides a brief outline of the physics parameters and power peaking analyses performed, including the application of uncertainty factors. These changes make the methods consistent with current NRC approved methods.

Reference 1 - "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.
Reference 2 - "Multidimensional Reactor Transient's and Safety Analysis Physics Parameter Methodology", DPC-NE-3001P-A, November 1991.
Reference 3 - "FSAR Chapter 15 System Transient Analysis Methodology", DPC-NE-3002-A, Revision 3, SER Dated February 5, 1999.

Question 6. Section 9.1.5, first paragraph. Please provide clarification of this change and the technical justification for it. Please provide comparison between the old sentence and the new sentence.

Response
Original Sentence: "Calculations using EPRI-NODE are run at these power levels and nominal conditions to provide predicted power distributions for comparison."

Proposed Sentence: "Calculations are performed at these power levels and nominal conditions to provide predicted power distributions for comparison."

Specifically the words "Calculations using EPRI-NODE are run" were changed to "Calculations are performed". This change makes the description in this section valid when other NRC approved design methods are used (for example, SIMULATE).

The specific Fq limit of 2.32 was removed, because this value may be reload specific and the current process is to control the Fq limit in the COLR. By making the topical report consistent with the Technical Specifications and COLR, an inconsistency between Technical Specifications and DPC-NE-2011 is removed.

While developing this response, DEC noted a typographical error in Section 6.1 on Page 6-1 of the proposed version of this topical report (namely, several 'less than' (<) signs should have been 'less than or equal to' (≤) signs). A marked up copy and a reprinted copy of this page (Page 6-1) is is included at the end of Attachment 3b.

Question 4. Section 6.2, where is UMR listed in section 6.2? Please provide original definition and new definition for comparison.

Response
The changes listed in Attachment 8a of the LAR submitted by Duke correspond to the section numbering found in the current version of this topical report. Therefore, all the changes associated with Section 6.2 in Attachment 8a are located in Section 6.3 of the proposed version of the topical report. UMR is not used in Section 6.2 of the proposed.version of the topical report but is used in Section 6.3.

Original Definition: In Section 6.2 of the current version of the topical report, UMR is defined "Uncertainty value for measured radial peaks, taken as 1.04 in the current Technical Specifications (2, 3)."

Proposed Definition: In Section 6.3 of the proposed topical report, UMR is defined "Uncertainty factor on the measured radial peaks, provided in the Technical Specifications (2, 3)."

This definition was updated to reflect that the value for UMR is to be found in the COLR as referenced by the Technical Specifications. This change is made to avoid a conflict if this value were to change in the future. As a result, the topical report now references Technical Specifications.

Attachment 3b - Non-Proprietary

Responses to Request for Additional Information

Topical Report Numbered DPC-NE-2011-P, Revision 1, Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors (TAC NOS. MB3343, MB3344, MB3222, MB3223)

The following page of this Attachment contains the marked up and reprinted page that is revised from the proposed version of this topical report. This page is being provided in response to Question 3.

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(ΔI) 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:

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

Where:

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

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:

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

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

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.

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation (Duke); and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

(ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuckman

M. S. Tuckman

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(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of Duke's response to NRC questions on Topical Report DPC-NE-2011-P, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors,* Revision 1. This information enables Duke to:

(a) Respond to NRC requests for additional information regarding transient response of Westinghouse PWRs.

(b) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.

(c) Perform safety evaluations per 10CFR50.59.

(d) Support core reload design activities for McGuire and Catawba Nuclear Stations.

(e) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.

M. S. Tuckman

M. S. Tuckman

(Continued)

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: August 7, 2002
Date

Mary P. Nehus, Notary Public

My Commission Expires:

JAN 22, 2006
Date

SEAL

Attachment 4
Responses to Request for Additional Information
Topical Report Numbered DPC-NE-1003, Revision 1, McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testings
(TAC NOS. MB3343, MB3344, MB3222, MB3223)

General

Subsequent to receiving the NRC RAI package, a clarification of Question 4.d. was obtained from the NRC during a conference call on Monday July 15, 2002. Responses to all questions in the NRC RAI are given below, and responses to Question 4.d. takes into account the clarification received from the NRC. Some of the responses require making revisions to the proposed version of this topical report. The revised pages are included at the end of this Attachment.

Question 1. Appendix A of topical report DPC-NE-1003, Revision 1, contains two versions of DPC's rod swap measurement procedures PT/O/A/4150/11A: Attachment 3 (dated June 1986) and Attachment 4 (dated April 1984). There are differences in these two versions of procedures. For example, in the Attachment 3 version, Steps 12.2.2.and 12.2.3, respectively, specify the insertion of bank 1 until the indicated reactivity is approximately -20 pcm, and the withdrawal of reference bank until the indicated reactivity is approximately +20 pcm; whereas in the Attachment 4 version, the insertion and withdrawal of bank 1 and reference bank, respectively, of steps 12.2.1 and 12.2.2 specify reactivity change of -/+ 10 pcm.

a. Since the Attachment 3 version of procedures is more recent, why is the Attachment 4 version referenced in Revision 1 of the topical report (Reference 2)?

b. Which of these two versions of rod swap measurement procedures will be used for McGuire and Catawba Units?

Response 1.a.

Appendix A of the submitted report is labeled "NRC/DPC Correspondence Including DPC Responses to NRC Requests for Additional Information." The information currently in Appendix A contains information provided by DPC in response to the NRC RAI (letter dated 1/12/87) associated with the original submittal of this report. The differences in Attachment 3 and Attachment 4 are due to the timing of the submittals of this topical report, NRC RAI, and DPC responses.

Attachment 3 contains the then most current versions of the procedures for rod swap measurements and were provided in response to Question 2 in the NRC RAI mentioned above. Attachment 4 is an earlier version of the Rod Swap procedure, and this procedure was provided in response to Question 5 of the NRC RAI mentioned above.

The reference list in the proposed version of this topical report was not updated, because the procedure is referenced in a general way and because some of the measured data used to perform the benchmark calculations was processed using the procedure referenced in the original submittal.

Response 1.b.
Duke currently employs the Westinghouse Dynamic Rod Worth Measurement technique for determining rod worth during ZPPT; however, rod swap may be used as a contingency. The procedure to be used in the event the rod swap test is to be performed now is not the same as those shown in Attachments 3 and 4. An information only version of the current procedure is provided in Attachment 4a (see response to Question 4.c.)

Question 2. In the Attachment 3 version of rod swap measurement procedures PT/O/A/4150/11A, Step 12.1.3 states that: "Repeat steps 12.2.1 and 12.2.2 until the previously inserted bank fully withdrawn." Is there a typographic error in the words "steps 12.2.1 and 12.2.2"? Should the correct words appear to be "steps 12.1.1 and 12.1.2"?

Response
Yes, this is a typographical error. This error is not in the current Rod Swap procedure.

Question 3. The equation in Section 3, Measurement Procedure, of the topical report for calculating the inferred rod worth of bank x is different from the equation in Step 12.5.3 of the Attachment 3 procedures. The difference appears to be due to the initial height of the reference bank for performing the rod swap measurement of the measured bank. Clarify the exact procedure to be used in the rod swap test, and make all necessary corrections in the topical report and the procedures to be consistent.

Response
The difference is the initial height of the reference bank for measuring the other banks. In the situation where the reference bank only inserted critical position is 0 SWD, the results of the topical report equation and the procedure equation are the same. If the critical position of the reference bank only inserted is not 0 SWD, it is necessary to account for this small amount of reactivity. This situation may arise as a result of small temperature or boron changes during the test. The proposed topical report has been modified to reflect this, and the revised pages (Pages 2 and 3) are included at the end of this Attachment.

Question 4. The third sentence in Section 3 of the topical report is revised to read: "All other banks are then exchanged with the reference bank or other test banks at constant boron conditions until the measured bank is fully inserted." It is stated, in Attachment 9a - Detailed Listing of Changes to DPC-NE-1003A, that the third sentence in Section 3 is revised to make the report consistent with current procedures. The "Revision History" in the topical report states that this revision [Revision 1] also reflects a refinement in the rod swap to make use of two test banks.

a. What is the "current procedures"? What is the date of the current procedures?
b. Are the current procedures the same or different from the one in Attachment 3? The Attachment 3 procedures did not include the exchange of a test bank with other test bank.
c. If the "current procedures" are different from that of Attachment 3 or 4, provide a copy of the procedures, and appropriately reference it in the report.
d. Is the statement in "Revision History" referring to this revision? Please explain what the statement means.

Response 4.a.
The current McGuire procedure is PT/0/A/4150/11A, dated 1/19/96.

Response 4.b.
The current procedure is not the same as Attachment 3. The current procedure allows for the exchange of two test banks, namely of the bank to be measured and the bank just measured. This exchange takes place while moving the test bank to be measured into the fully inserted position.

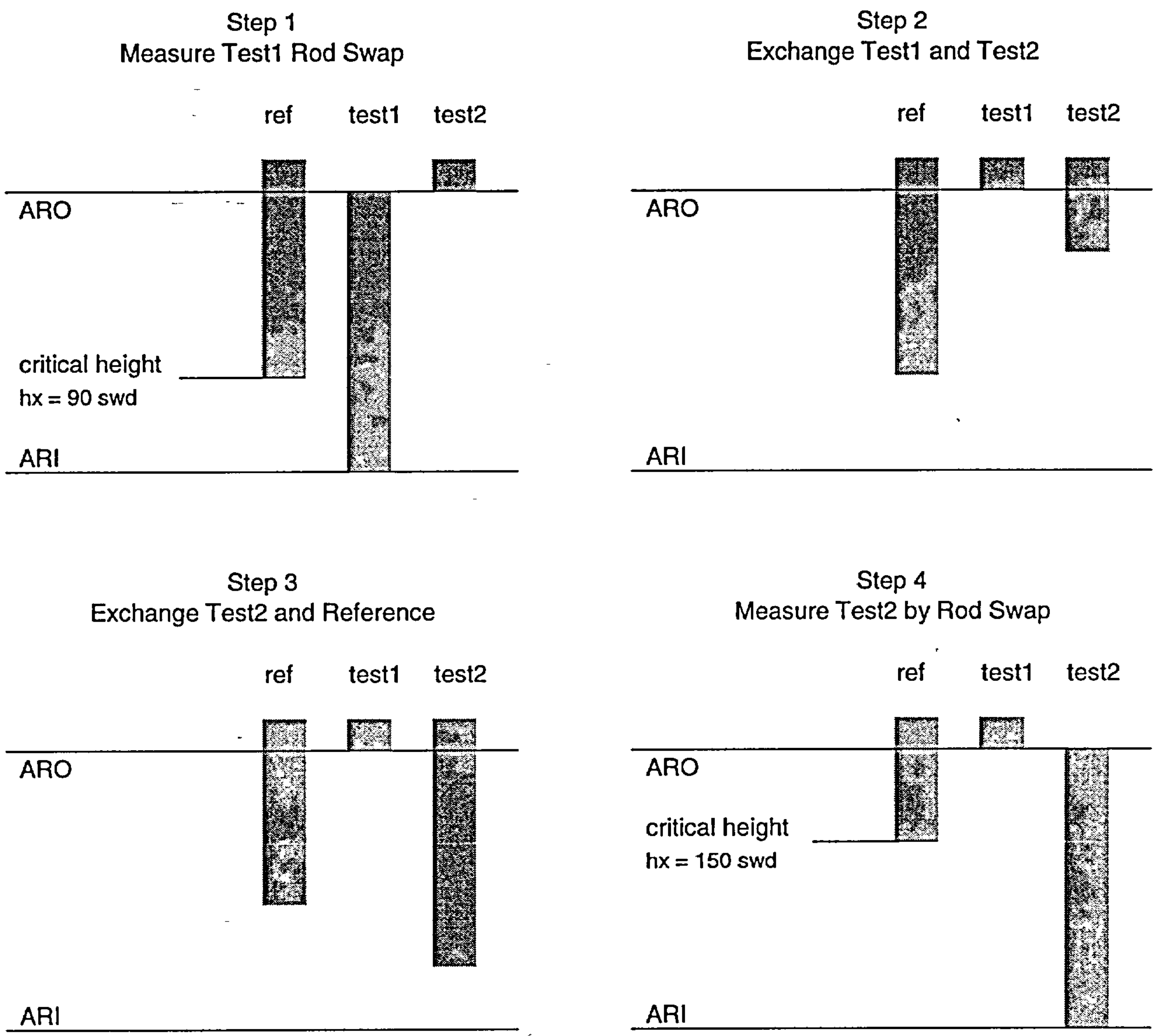
Response 4.c.
An information only copy of the current McGuire procedure is included in Attachment 4 of this response package. The topical report only makes a general reference to the plant procedure.

Response 4.d.
The statement "This revision also reflects a refinement in the rod swap to make use of two test banks." in the Revision History of this topical report does apply to this proposed revision. The statement refers to the description of intermediate steps of exchanging two test banks after measuring the worth of one test bank and before measuring the worth of the next test bank.

The test bank to be measured is moved into the fully inserted position by exchanging first with the previous test bank and then with the reference bank as necessary. The final test bank/reference bank configuration, and therefore measured worth of the test bank, is the same whether it is exchanged with the reference bank or with the previous test bank. This evolution is shown pictorially on the next page.

Clarification of Appendix A
An additional correspondence between DPC and the NRC became known subsequent to the submittal of the proposed version of this topical report. Appendix A of the proposed version of this topical report has been modified to include this additional correspondence. The pages to be added to Appendix A are provided at the end of Attachment 4

Rod Swap
Rod Exchange with Two Test Banks



Attachment 4

Responses to Request for Additional Information

Topical Report Numbered DPC-NE-1003, Revision 1, McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testings (TAC NOS. MB3343, MB3344, MB3222, MB3223)

The following pages of this Attachment contain the marked up and reprinted pages that are revised from the proposed version of this topical report. These pages are being provided in response to Question 3.

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

- $(\Delta\rho)_x$ 2 - 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.

- $(h^M_x)_0$ - The initial critical position of the reference bank before exchange with bank x.

- $(\Delta\rho_1)$ - The measured integral worth of the reference bank from 0 steps to $(h^M_x)_0$.

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:

$$W^{I}_{x} = W^{M}_{ref} - \alpha_x (\Delta\rho_{2})_x - (\Delta\rho_1)$$

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

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.

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- 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

- $(\Delta\rho_2)_x$ - 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.

- $(h^m_x)_o$ - The initial critical position of the reference bank before exchange with bank x.

- $(\Delta\rho_1)$ - The measured integral worth of the reference bank from 0 steps to $(h^m_x)_o$.

3. Measurement Procedure

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$$W^{I}_{x} = W^{M}_{ref} - \alpha_{x} \ (\Delta\rho_{2})_{x} - (\Delta\rho_{1})$$

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

Attachment 4
Responses to Request for Additional Information
Topical Report Numbered DPC-NE-1003, Revision 1, McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testings
(TAC NOS. MB3343, MB3344, MB3222, MB3223)

The following pages of this Attachment contain the additional DPC correspondence to be included in Appendix A of the proposed version of this topical report.

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

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

March 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:

Pursuant to telecons of February 19, 1987 and March 10, 1987 between D.S. Hood (ONRR) et. al., and S.A. Gewehr (DPC) et. al., attached are revised responses to Questions 6 and 7 of D.S. Hood's request for information dated January 12, 1987.

Very truly yours,

H.B. Tucker /

Hal B. Tucker

SAG/61/jgm

Attachment

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:

In the original Nuclear Physics Methodology Topical, DPC-NF-2010A, Duke Power Company benchmarked its methods for predicting rod worths against measurements made during the startup testing for both initial cores at the McGuire Nuclear Station. These measurements were made using the boration/ dilution technique for determining rod worths in sequential insertion. In its review of this topical, the NRC accepted the capability of Duke Power to adequately predict control rod worths and shutdown margin using the outlined methodology.

In the Rod Swap Methodology Report recently sent to the Commission, Duke Power benchmarked its methodology for predicting rod worths using the rod swap technique against 5 cycles of actual rod swap measurements. This methodology utilized the same computer codes previously benchmarked in DPC-NF-2010A. All predictions, when compared to the measured results, met the acceptance criteria as outlined in the rod swap plant procedure.

It has been noted in previous conversations with the NRC that the two benchmarking studies noted above do not make comparisons of the same units for the same cycles. It is Duke Power's position that there is really no benefit from this type of comparison. A valid comparison cannot be expected since boration/-dilution is a sequential measured worth calculation and rod swap consists of a summation of the worths of each rod individually inserted into an otherwise unrodded core. It is therefore impossible to make direct comparisons between worths of the two methods. The only thing that can be looked at is the percent difference between measured and predicted for th two methods. When looking at percent differences between measured and predicted, one does not have to look at the same unit and cycle to verify methodologies are correct. Comparisons of predicted and measured rod worths done using boration/ dilution and rod swap on the two Catawba units are enclosed. The boration/dilution technique was used to measure rod worths in sequential bank insertion for the Catawba 1 Cycle 1 core while Catawba 2 Cycle 1 measurements were done using the rod swap technique (Table 1). From a neutronics standpoint, the two cores are almost identical. This assumption can be justified by examining the core loadings and the results of the Zero Power Physics Testing for each of the units. Several key parameters concerning the core are shown in Table 2. Also enclosed are the quarter core loading pattern (Figure 1) and a comparison of the quarter core assembly power distribution from the zero power map taken during the startup physics testing (Figure 2).

It should also be pointed out that the rod worths from the rod swap predictions are not the worths used to calculate the shutdown margin. Rod swap only verifies the code's ability to predict rod worths. The rod worth used in the shutdown margin calculation is the N-1 worth.

Duke Power has provided a total of nine cycle of predicted rod worth comparisons to measured data with good to excellent results. This demonstrates the ability of the codes and methods used to adequately model reactivity effects due to control rods in any configuration. Therefore, the use of Duke Power predictions in the verification of shutdown margin with appropriate factors of conservatism applied to the calculation as outlined in DPC-NF-2010A Section 4.2.2.2 is justified.

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:

The safety analyses for the McGuire and Catawba Nuclear Stations have been performed by the current fuel vendor. The analyses utilized NRC-approved codes and methodologies and conservative input assumptions including values for key nuclear physics parameters such as reactivity coefficients, core power distributions, and shutdown margins, which are expected to bound the actual values of these parameters for current and future reload cores. An evaluation is performed for each reload cycle which consists of comparing nuclear design predictions to the safety analyses assumptions to ensure the safety analyses remain bounding. The cycle-specific evaluation process is described in WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology." Core physics testing performed for each cycle verifies the nuclear design predictions and ensures the actual core physics parameters are conservative with respect to the safety analyses.

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 to predicted values 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 to Technical Specifications to the NRC.

In order to tie the rod swap measurements to the verification of inputs to the safety analysis, Duke Power will perform an independent shutdown margin for each reload cycle using methods approved by the NRC in DPC-NF-2010A. In addition, for each cycle where Duke generates the rod swap prediction but the safety analysis has been performed by a vendor, a comparison between the Duke and vendor predicted total rod worth will be made at beginning-of-cycle, hot zero power conditions. Any significant discrepancies will be documented, reviewed, and resolved prior to startup physics testing.

Reference

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

TABLE 1

Rod Worth Measurement Data
Comparison of Rod Swap and Boration/Dilution Techniques

Bank	Rod Swap Integral Worths			Boron/Dilution Integral Worths		
	Predicted (PCM)	Measured (PCM)	% Diff**	Predicted (PCM)	Measured (PCM)	% Diff**
D	772	794	-2.85	773	788	-1.94
C	790	849	-7.47	1214	1203	0.91
B*	852	882	-3.52	1190	1171	1.60
A	249	250	-0.40	572	548	4.20
SE	377	385	-2.12	508	460	9.45
SD	497	525	-5.63	755	772	-2.25
SC	497	522	-5.03	1098	1099	-0.09
SB	765	834	-9.02	-	-	-
SA	674	706	-4.75	-	-	-
N-1	-	-	-	7370	7414	-.60
N	5473	5747	-5.01			

* Reference Bank

** % Diff = [(P-M)/P]*100

TABLE 2

Catawba 1 Cycle 1 and Catawba 2 Cycle 1
Comparison of Core Parameters

	Unit 1	Unit 2
KG U/ASSY		
Batch 1 1.6	424.169	424.623
Batch 2 2.4	423.508	425.805
Batch 3 3.1	423.676	424.519
AVE ENR		
Batch 1	1.6101	1.6104
Batch 2	2.3999	2.4014
Batch 3	3.1022	3.0954
ARO BORON ENDPOINT (PPMB)	975	975
ISO. TEMP. COEFF (PCM/°F)	-1.745	-1.81

Figure 1

CATAWBA 1 CYCLE 1 AND CATAWBA 2 CYCLE 1
QUARTER CORE LOADING PATTERN

	H	G	F	E	D	C	B	A
8	1.60	2.40 16	1.60	2.40 12	1.60	2.40 16	1.60	3.10 6
9	2.40 16	1.60	2.40 12	1.60	2.40 12	1.60	3.10 20	3.10
10	1.60	2.40 12	1.60	2.40 12	1.60	2.40 16	1.60	3.10 6
11	2.40 12	1.60	2.40 12	1.60	2.40 16	1.60	3.10 16	3.10
12	1.60	2.40 12	1.60	2.40 16	2.40	2.40 16	3.10	
13	2.40 16	1.60	2.40 16	1.60	2.40 16	3.10 15	3.10	
14	1.60	3.10 20	1.60	3.10 16	3.10	3.10	ENRICHMENT NUMBER OF BA'S PER ASSY	
15	3.10 6	3.10	3.10 6	3.10				

Figure 2

CIC1 AND C2C1 HZP POWER DISTRIBUTIONS @ MWD/MTU

ARO, HZP, NO XE, NO SM

Row		H	G	F	E	D	C	B	A
8	C1C1	.72	.82	.84	1.00	.89	.97	.95	.97
	C2C1	.72	.82	.82	.97	.87	.94	.93	.95
	% DIFF	0.00	0.00	2.44	3.09	2.30	3.19	2.15	2.11
9	C1C1	.82	.79	.95	.88	1.02	.91	1.11	1.01
	C2C1	.82	.78	.95	.88	1.01	.89	1.10	1.00
	% DIFF	0.00	1.28	0.00	0.00	.99	2.25	.91	1.00
10	C1C1	.83	.96	.89	1.04	.93	1.01	.93	.90
	C2C1	.83	.95	.89	1.05	.93	1.01	.93	.89
	% DIFF	0.00	1.05	0.00	-.95	0.00	0.00	0.00	1.12
11	C1C1	1.01	.90	1.06	.99	1.12	1.03	1.16	.73
	C2C1	1.00	.89	1.06	.99	1.12	1.03	1.17	.74
	% DIFF	1.00	1.12	0.00	0.00	0.00	0.00	-.85	-1.35
12	C1C1	.91	1.04	.94	1.12	1.43	1.19	1.22	
	C2C1	.91	1.04	.93	1.13	1.44	1.19	1.23	
	% DIFF	0.00	0.00	1.08	-.88	-.69	0.00	-.81	
13	C1C1	.98	.92	1.03	1.04	1.19	1.22	.89	
	C2C1	1.01	.94	1.04	1.06	1.20	1.21	.88	
	% DIFF	-2.97	-2.13	-.96	-1.89	-.83	.03	1.14	
14	C1C1	.97	1.14	.96	1.18	1.22	.88		
	C2C1	.98	1.16	.97	1.19	1.22	.89		
	% DIFF	-1.02	-1.72	-1.03	-.84	0.00	-1.12		
15	C1C1	1.00	1.04	.92	.74				
	C2C1	1.00	1.04	.93	.76				
	% DIFF	0.00	0.00	-1.08	-2.63				

```
C1C1      CORE AVERAGE   1.00
C2C1      CORE AVERAGE   1.00
% DIFF    CORE AVERAGE   -.05


C1C1      MAXIMUM MAGNITUDE IS  1.43 AT ASSEMBLY D - 12
C2C1      MAXIMUM MAGNITUDE IS  1.44 AT ASSEMBLY D - 12
% DIFF    MAXIMUM MAGNITUDE IS  3.19 AT ASSEMBLY C - 8
     PERCENT ERROR BETWEEN THE MAXIMUM VALUES IS   -.69


AVERAGE ABSOLUTE RELATIVE-ERROR            .88 PERCENT
ROOT MEAN SQUARE OF THE RELATIVE ERROR    1.22 PERCENT
ROOT MEAN SQUARE OF THE DIFFERENCE        1.16 PERCENT
```

Attachment 4
Responses to Request for Additional Information
Topical Report Numbered DPC-NE-1003, Revision 1, McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testings
(TAC NOS. MB3343, MB3344, MB3222, MB3223)

The following pages of this Attachment contain an information only copy of the current rod swap procedure. This is being provided in response to Question 4.c.

Form 34731 (R8-94)

Duke Power Company
PROCEDURE PROCESS RECORD

ID No. PT/0/A/4150/11A
Change(s) 0 to 24 incorporated

PREPARATION

(2) Station McGuire Nuclear Station

(3) Procedure Title Control Rod Worth Measurement

(4) Prepared By [signature] Date 01/19/96

(5) Requires 10CFR50.59 evaluation?

- [X] Yes (New procedure or reissue with major changes)
- [ ] No (Reissue with minor changes OR to incorporate previously approved changes)

(6) Reviewed By [signature] Date 1-19-96

Cross-Disciplinary Review By N/R JBQ Date

(7) Additional Reviews

Reviewed By SC Ballard Date 1-19-96

Reviewed By Date

(8) Temporary Approval *(if necessary)*

By (SRO) Date

By Date

(9) Approved By [signature] Date 1/19/96

PERFORMANCE (compare with control copy every 14 calendar days)

(10) Compared with Control Copy Date

Compared with Control Copy Date

Compared with Control Copy Date

(11) Date(s) Performed

Work Order Number (WO#)

COMPLETION

(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 Procedure requirements met?

Verified By Date

(13) Procedure Completion Approved Date

(14) Remarks *(attach additional pages, if necessary)*

FOR INFORMATION ONLY

ATTACHMENT TO THE PROCEDURE PROCESS RECORD:

Procedure Title: Control Rod Worth Measurement: Rod Swap

Changes included in the reissue:

Section 2.0 The following references were added:
- FSAR Section 14.3.2.3
- Technical Specification 3.10.3
- SER for Duke Power Rod Swap Methodology Report for Startup Physics Testing, May 22, 1987

The following Support Documents were added:
- PT/0/A/4150/10, Boron Endpoint Measurement

Section 3.0 Time requirements changed from six to eight hours

Section 4.0 PT/0/A/4150/10, Boron Endpoint Measurement was removed as a prerequisite test.

Section 5.0 Step 5.1 was revised to better define the required reactivity computer.

Step 5.2 was added to define the scale of the 2 pen strip chart recorder for the reactivity computer setup.

Step 5.3 was added to recommend (optional) monitoring of Tave during testing.

Section 6.0 The following Limits and Precautions were added:
-If a stable startup rate of 0.5 DPM is achieved, insert rods to reduce startup rate to less than 0.5 DPM. If the startup rate is greater than or equal to 1.0 DPM, immediately trip the reactor.
- Avoid makeup to the VCT during rod swap evolution.
- Keep reactivity between - 50 pcm and 75 pcm during rod swap.
-Adjustments to procedure are required if any bank (other than Bank 8) is worth more than the reference bank.

Section 8.0 - Step 8.2 was deleted.
- Step 8.3 was changed to specify a pcm limit.
- Step 8.6 was revised to match the Startup Physics Test Program notation of 2235 $\pm$ 50 psig in addition to providing the corresponding pressure range.
- Step 8.5 was revised to match the Startup Physics Test Program notation of 557 $\pm$ 2 °F in addition to providing the corresponding temperature range.

The following Prerequisite System Conditions were added to ensure stable test conditions, to ensure NC system boron remains stable and to aid in setup of the reactivity computer:
- Test equipment setup per Section 5.0
- Rod Control System has been checked per Enclosure 13.10

Section 11.0 - Changes all references of Design Engineering to G.O. Nuclear Engineering

Section 12.0 - Revised procedure to reflect change #24 throughout (20 to 40 pcm limit).

DPC-NE-2009-P

Duke Power Company Westinghouse Fuel Transition Report, Revision 1

Marked Copy

the RELAP5 model, which is used to model the mass and energy release from LOCAs, are also anticipated. The RETRAN and RELAP5 model changes for the RFA design are not significant enough to require reanalyses. Future reanalyses will incorporate the RFA design model revisions.

6.5 LOCA Analyses

Large and small break LOCA analyses will be performed by Westinghouse using approved versions of the Westinghouse Appendix K LOCA evaluation models. All features employed have been approved by the NRC as required and annual model reports for the evaluation models have been supplied to the NRC, the most recent of which is found in Reference 6-22. Therefore, no NRC review of the evaluation model features is necessary, and only methodology with respect to analyzing McGuire/Catawba will be presented in this section. New LOCA analyses will be performed to support the licensing of McGuire/Catawba during the transition and full core operation of the RFA design.

6.5.1 Small Break LOCA

For small break LOCAs (SBLOCAs) due to breaks less than 1 ft^2, Westinghouse developed the NOTRUMP computer code (Reference 6-23) to calculate the transient depressurization of the reactor coolant system (RCS) as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 6-24, 6-25, 6-26, and 6-27) was developed and licensed by Westinghouse to determine the RCS response to design basis SBLOCAs, and to address NRC concerns expressed in NUREG-0737, Item II.K.3.30. ^ Insert A

The NRC approved noding scheme for the NOTRUMP Evaluation Model is shown in Reference 6-24, although minor noding changes to facilitate the modeling of broken loop ECCS were instituted and reported to the NRC in Reference 6-28. Peak cladding temperature (PCT) calculations are performed with the LOCTA-IV code (Reference 6-29) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Additional modifications to the LOCTA-IV code to allow the modeling

Insert A

In addition, several model enhancements have been made to the evaluation model and implemented via the 10 CFR 50.46 process. These enhancements or changes were determined to be non-significant as defined by 10 CFR 50.46. Westinghouse reported these enhancements to the NRC in annual notification reports (References 6-22, 6-28 and 6-39) and implemented them on a forward fit basis. Duke did not report these changes in their annual 10 CFR 50.46 reports since the Westinghouse SBLOCA analysis using these enhancements had not been implemented for McGuire and Catawba during this time period. The purpose of identifying these enhancements in this report is to clearly identify the SBLOCA analysis method to be used to support McGuire and Catawba.

6-38 WCAP-10484-P-A Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", September 1993.

6-39 NSD-NRC-99-5839, "1998 Annual Notification of Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46 (a)(3)(ii).

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the RELAP5 model, which is used to model the mass and energy release from LOCAs, are also anticipated. The RETRAN and RELAP5 model changes for the RFA design are not significant enough to require reanalyses. Future reanalyses will incorporate the RFA design model revisions.

6.5 LOCA Analyses

Large and small break LOCA analyses will be performed by Westinghouse using approved versions of the Westinghouse Appendix K LOCA evaluation models. All features employed have been approved by the NRC as required and annual model reports for the evaluation models have been supplied to the NRC, the most recent of which is found in Reference 6-22. Therefore, no NRC review of the evaluation model features is necessary, and only methodology with respect to analyzing McGuire/Catawba will be presented in this section. New LOCA analyses will be performed to support the licensing of McGuire/Catawba during the transition and full core operation of the RFA design.

6.5.1 Small Break LOCA

For small break LOCAs (SBLOCAs) due to breaks less than 1 ft^2, Westinghouse developed the NOTRUMP computer code (Reference 6-23) to calculate the transient depressurization of the reactor coolant system (RCS) as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 6-24, 6-25, 6-26, and 6-27) was developed and licensed by Westinghouse to determine the RCS response to design basis SBLOCAs, and to address NRC concerns expressed in NUREG-0737, Item II.K.3.30.

In addition, several model enhancements have been made to the evaluation model and implemented via the 10 CFR 50.46 process. These enhancements or changes were determined to be non-significant as defined by 10 CFR 50.46. Westinghouse reported these enhancements to the NRC in annual notification reports (References 6-22, 6-28 and 6-39) and implemented them on a forward fit basis. Duke did not report these changes in their annual 10 CFR 50.46 reports since the Westinghouse SBLOCA analysis using these enhancements had not been implemented for McGuire and Catawba during this time period. The purpose of identifying these enhancements in this report is to clearly identify the SBLOCA analysis method to be used to support McGuire and Catawba.

The NRC approved noding scheme for the NOTRUMP Evaluation Model is shown in Reference 6-24, although minor noding changes to facilitate the modeling of broken loop ECCS were instituted and reported to the NRC in Reference 6-28. Peak cladding temperature (PCT) calculations are performed with the LOCTA-IV code (Reference 6-29) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Additional modifications to the LOCTA-IV code to allow the modeling

6-38 WCAP-10484-P-A Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", September 1993.

6-39 NSD-NRC-99-5839, "1998 Annual Notification of Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10CFR50.46 (a)(3)(ii)".

Attachment 6a

McGuire Units 1 and 2 Technical Specifications

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5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in Chapter 16 of the UFSAR and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: and 60 ppm

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit S for Specification 3.1.3,

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor limits for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.14,

11. SHUTDOWN MARGIN for Specification 3.1.~~1~~, and

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary).

2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," (W Proprietary).

3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," (B&W Proprietary).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," ~~March 1990~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

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

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. UFSAR, Chapter 7.

Attachment 6b

Catawba Units 1 and 2 Technical Specifications

Marked Copy

5.6 Reporting Requirements

and 60 ppm

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3.1.3,

s

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.15,

11. SHUTDOWN MARGIN for Specification 3.1.1,

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY" (W Proprietary).

2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE" (W Proprietary).

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2, and

13. Reactor Makeup Water Pumps Combined Flow Rates limit for Specifications 3.3.9 and 3.9.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", March 1990.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", ~~March 1990~~.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Chapter 7.

Attachment 7a

McGuire Units 1 and 2 Technical Specifications

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Remove	Insert
5.6-2	5.6-2
5.6-3	5.6-3
B 3.2.1-11	B 3.2.1-11
B 3.2.3-4	B 3.2.3-4

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in Chapter 16 of the UFSAR and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 60 ppm and 300 ppm surveillance limits for Specification 3.1.3,

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor limits for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.14,

11. SHUTDOWN MARGIN for Specification 3.1.1, and

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary).

2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," (W Proprietary).

3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," (B&W Proprietary).

(continued)

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

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

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. UFSAR, Chapter 7.

Attachment 7b

Catawba Units 1 and 2 Technical Specifications

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Remove	Insert
5.6-3	5.6-3
B 3.2.1-11	B 3.2.1-11
B 3.2.3-4	B 3.2.3-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Moderator Temperature Coefficient BOL and EOL limits and 60 ppm and 300 ppm surveillance limits for Specification 3.1.3,

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.15,

11. SHUTDOWN MARGIN for Specification 3.1.1,

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2, and

13. Reactor Makeup Water Pumps Combined Flow Rates limit for Specifications 3.3.9 and 3.9.2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY" (W Proprietary).

2. WCAP-10266-P-A, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE" (W Proprietary).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Chapter 7.

Section 12.0 (cont.)

- Added - NOTES:
  1. The following steps ensure that no VCT makeup is required during rod swap.
  2. To reduce pump head loss, all additions should occur directly to the suction of the NV pumps (through NV-175 or NV-265), <u>NOT</u> to the top of the VCT (through NV-171). Auto Makeup Limit = 41.4% and Low Level Alarm = 15.7%.
- Added - CAUTION - Ensure that VCT pressure does not exceed 30 psig while performing Step 12.2. This may require batching the additions to allow the VCT pressure controller adequate time to operate. Failure to do so may result in misoperation of the boric acid transfer pump
- Added Step 12.2 - Ensure that the VCT level is sufficient such that makeup will not be required for approximately 4 hours.
- Added Step 12.4 - Verify that drift in the reactivity trace over that last 30 minutes is less than 5 pcm.
- Added NOTE - Temporary signs will be provided for the OATC to assist in designating rod group being withdrawn and rod group being inserted.
- Added Step 12.16 - Any temporary signs provided for the OATC to assist in designating rod group being withdrawn and rod group being inserted should be removed from the Control Room upon completion of this test.

The procedure was updated to follow the current procedure writers guidelines for NOTES, CAUTIONS, IFs, etc. Additionally, the procedure steps were renumbered as needed.

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION
CONTROL ROD WORTH MEASUREMENT: ROD SWAP

1.0 PURPOSE

NOTE:	The reference bank is the bank which has the predicted highest reactivity worth of all control and shutdown banks when inserted into an otherwise unrodded core.

1.1 To determine the worth of all control and shutdown banks, except the reference bank, as inferred from an iso-reactivity interchange with the reference bank.

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 Source Documents:

2.1.1 Rod Bank Worth Measurements Utilizing Bank Exchange, WCAP-9863-A, May 1982.

2.1.2 Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Startup Physics Test Program, April 8 1988.

2.1.3 Operating Experience Program Commitment 1-91-41-001A.

2.1.4 Significant Event Report 90-15.

2.1.5 FSAR Section 14.3.2.3

2.1.6 Technical Specification 3.10.3

2.1.7 NSD 213, "Conduct of Infrequently Performed Tests or Evolutions".

2.1.8 SER for Duke Power Rod Swap Methodology Report for Startup Physics Testing, May 22, 1987.

2.2 Support Documents:

2.2.1 Control Rod Worth Measurement, PT/0/A/4150/11

2.2.2 Post Refueling Controlling Procedure for Criticality, Zero Power Physics Test, and Power Escalation Testing, PT/0/A/4150/21

2.2.3 MNS Technical Specifications

- Surveillance Requirement 3.1.1.1
- Surveillance Requirement 3.10.4
- Surveillance Requirement 3.10.3
- Surveillance Requirement 3.10.2.

2.2.4 Duke Power Company, Startup and Operational Report for appropriate unit and cycle.

2.2.5 RODSWAP computer application User's Guide

2.2.6 PT/0/A/4150/10, Boron Endpoint Measurement

3.0 TIME REQUIRED

3.1 Duration: 8 hours

3.2 Personnel Required: Two Engineers

4.0 PREREQUISITE TESTS

None

5.0 TEST EQUIPMENT

5.1 Westinghouse Digital Reactivity Computer or equivalent, (with flux signal from top and bottom of one power range channel).

5.2 One 2 pen strip chart recorder with reactivity ($\pm$100 pcm scale) and flux signal inputs.

5.3 One strip chart recorder monitoring NC Tave (optional).

6.0 LIMITS AND PRECAUTIONS

6.1 If a stable startup rate of 0.5 DPM is achieved, insert rods to reduce startup rate to less than 0.5 DPM. If the startup rate is greater than or equal to 1.0 DPM, immediately trip the reactor.

6.2 The NC system temperature is 557°F ± 2°F (555°F to 559°F), and controlled preferably by steam dump to the condenser. Temperature may be controlled by other methods as required by system conditions.

6.3 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.

6.4 The rod insertion limit and bank overlap sequence will be violated during this test. The Unit SRO and OATC should be made aware in advance and should anticipate the associated alarms. Technical Specification 3.10.3 allows for this violation.

6.5 Maintain the flux level in the zero power test range established in PT/0/A/4150/21.

6.6 If bank has two groups, both must be at the same position prior to switching rod control selector switch between banks to avoid group misalignment.

6.7 If any unexpected, inadvertent drop of an RCCA(s) or Bank(s) of RCCAs occurs, recommend to Unit SRO immediate initiation of manual reactor shutdown. (OEP Commitment 1-91-41-001A)

6.8 Avoid makeup to VCT during rod swap evolution.

6.9 Keep reactivity between - 50 pcm and 75 pcm during rod swap.

6.10 Adjustments to procedure are required if any bank (other than Bank 8) is worth more than the reference bank.

7.0 REQUIRED UNIT STATUS

Initial

____ Mode 2 with the flux level in the zero power physics test band established in PT/0/A/4150/21.

8.0 PREREQUISITE SYSTEM CONDITIONS

NOTES:		
	1)	The following steps may be signed off in any order.
	2)	See Enclosure 13.1 for an explanation of nomenclature used in this test.
	3)	Banks should be measured in order of increasing predicted worth.
	4)	Step 8.6 may be performed prior to any other Section 8 step.

Initial

____ 8.1 Complete Enclosure 13.2.

____ 8.2 Ensure the reactor is critical with all control and shutdown banks fully withdrawn except the Reference Bank which is < 50 pcm from fully inserted.

____ 8.3 The Rod Control Selector switch is in Bank Select Mode set the Reference Bank.

____ 8.4 Reactor coolant system temperature is 557 ± 2°F (555 °F to 559°F).

____ 8.5 Reactor coolant system pressure is 2235 ± 50 psig (2185 to 2285).

____ 8.6 Complete Enclosure 13.9.

____ 8.7 IF available, start computer application, RODSWAP, as directed by reference 2.2.5.

____ 8.8 Test equipment is setup per section 5.0

____ 8.9 Rod Control System has been checked per Enclosure 13.10

Sections 7.0 and 8.0 Performed By/Date: ______________________________

9.0 TEST METHOD

The bank with the highest predicted value of reactivity worth has been measured using the dilution technique per PT/0/A/4150/11. This bank serves as a reference. The integral worth of the remaining 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 after each bank exchange, recorded on Enclosure 13.3:

- Time
- Critical height of reference bank

10.2 Nuclear design predictions on Enclosure 13.2.

10.3 A copy of the rod positions and rod worths for the reference bank from Enclosure 13.2 of PT/0/A/4150/11.

10.4 The calculated, implied integral worth W_x^I for each RCC bank except the reference bank. List data on Enclosure 13.3 <u>OR</u> from RODSWAP printout attached to Enclosure 13.6.

10.5 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 13.7 <u>OR</u> on RODSWAP printout attached to Enclosure 13.6.

11.0 ACCEPTANCE CRITERIA

NOTES:

1) The appropriate actions for failure of an acceptance or review criteria are as follows:

<u>Acceptance Criteria</u>: G.O. Nuclear Engineering shall:

- Provide concurrence to continue testing.
- Investigate and provide solution within 30 days of the test.
- Submit a report of the findings to the NRC within 45 days of the test.

<u>Review Criteria</u>: G.O. Nuclear Engineering shall:

- Investigate and provide solution within 60 days of the test.
- Submit a report of the findings to the NRC within 75 days of the test.

2) For calculating percent differences, use $\left(\frac{\text{Meas}}{\text{Pred}} - 1\right) \times 100\%$

11.1 Acceptance Criteria

11.1.1 The sum of all banks (ε_2) >90% of predicted.

11.1.2 For all banks other than the reference bank, from Enclosure 13.7 either:

a) $(\varepsilon_1)_x$ is $\pm$30% of predicted for each bank x

<u>OR</u>

b) $(W_x^P - W_x^I)$ is $\pm$ 200 pcm of predicted for each bank x, whichever is greater.

11.1.3 All banks, both control and shutdown banks, are measured.

11.2 Review Criteria:

11.2.1 From Enclosure 13.7, the sum of all banks (ε_2) is $\leq$110%.

11.2.2 For all banks other than the reference bank, from Enclosure 13.7, either:

a) $(\varepsilon_1)_x$ is $\pm$15% of predicted for each bank x

<u>OR</u>

b) $(W_x^P - W_x^I)$ is $\pm$ 100 pcm of predicted for each bank x, whichever is greater.

12.0 PROCEDURE

Initial

NOTE:	All banks except reference bank are referred to by bank number identified on Enclosure 13.2.

____ 12.1. Attach Enclosure 13.2 of PT/0/A/4150/11 and label as Enclosure 13.8.

NOTES:	
1)	Step 12.2 ensures that no VCT makeup is required during rod swap.
2)	To reduce pump head loss, all additions should occur directly to the suction of the NV pumps (through NV-175 or NV-265), NOT to the top of the VCT (through NV-171). Auto Makeup Limit = 41.4% and Low Level Alarm = 15.7%.

CAUTION:	Ensure that VCT pressure does not exceed 30 psig while performing Step 12.2. This may require batching the additions to allow the VCT pressure controller adequate time to operate. Failure to do so may result in misoperation of the boric acid transfer pump.

____ 12.2 Ensure that the VCT level is sufficient such that makeup will not be required for approximately 4 hours.

____ 12.3 Verify that drift in the reactivity trace over that last 30 minutes is less than 5 pcm.

NOTE:	The first assigned bank on Enclosure 13.2 is referred to as Bank 1.

12.4 Measure integral worth of first assigned bank of Enclosure 13.2 as follows:

____ 12.4.1 Record initial critical position of reference bank $(h_x^M)_o$ on Enclosure 13.3.

CAUTION:	
1)	When switching from one bank to another, step counters for both groups, for banks with two groups, must be indicating the same step number to avoid rod misstepping.
2)	During rod exchange, ensure limits and precautions per Step 6.8 are observed.

NOTE:	Temporary signs will be provided for the OATC to assist in designating rod group being withdrawn and rod group being inserted.

____ 12.4.2 Direct Operations to insert bank 1 until indicated reactivity is approximately - 40 pcm.

____ 12.4.3 Direct Operations to withdraw reference bank until indicated reactivity is approximately + 40 pcm.

____ 12.4.4 Repeat Steps 12.4.2 and 12.4.3 until bank 1 is fully inserted maintaining indicated reactivity at approximately ± 40 pcm.

____ 12.4.5 Direct Operations to adjust position of reference bank until reactor is critical.

___ 12.4.6 Record final critical configuration data (h_x^M) on Enclosure 13.3.

12.5 Measure integral worth of remaining assigned banks as follows:

NOTES:		
	1)	The bank being measured is denoted as bank N.
	2)	The previously measured bank is denoted as bank N-1.
	3)	N = 2 for the second assigned bank.

___ 12.5.1 Direct Operations to insert bank N until indicated reactivity is approximately - 40 pcm.

___ 12.5.2 Direct Operations to withdraw bank N-1 until indicated reactivity is approximately + 40 pcm.

___ 12.5.3 Repeat Steps 12.5.1 and 12.5.2 until bank N is fully inserted or bank N-1 is fully withdrawn.

___ 12.5.4 IF bank N is fully inserted before bank N-1 is fully withdrawn, direct Operations to insert reference bank, compensating with withdrawal of bank N-1, maintaining indicated reactivity approximately ± 40 pcm throughout, until critical conditions are achieved with bank N-1 fully withdrawn.

___ 12.5.5 IF bank N-1 is fully withdrawn before bank N is fully inserted, direct Operations to withdraw reference bank, compensating with insertion of bank N, maintaining indicated reactivity approximately ± 40 pcm throughout, until critical conditions are achieved with bank N fully inserted or reference bank is fully withdrawn.

___ 12.5.6 IF bank N is not fully inserted and reference bank is fully withdrawn, mark Steps 12.5.7 and 12.5.8 N/A AND measure bank after others.

___ 12.5.7 Adjust position of reference bank until reactor is critical.

___ 12.5.8 Record final critical configuration data (h_x^M) on Enclosure 13.3.

___ 12.5.9 Repeat Steps 12.5.1 through 12.5.8 using Enclosure 13.4 for step signoffs to measure integral worths of assigned bank N = 3 through 7.

12.6 Measure integral worth of bank 8 as follows:

___ 12.6.1 Direct Operations to insert bank 8 until indicated reactivity is approximately - 40 pcm.

___ 12.6.2 Direct Operations to withdraw bank 7 until indicated reactivity is approximately + 40 pcm.

___ 12.6.3 Repeat Steps 12.6.1 and 12.6.2 until bank 8 is fully inserted OR bank 7 is fully withdrawn.

____ 12.6.4 **IF** bank 8 is fully inserted before bank 7 is fully withdrawn, insert reference bank, compensating with withdrawal of bank 7, maintaining indicated reactivity between ± 40 pcm throughout until critical conditions.

____ 12.6.5 **IF** bank 7 is fully withdrawn before bank 8 is fully inserted, withdraw reference bank compensating with insertion of bank 8 maintaining indicated reactivity between ± 40 pcm throughout, until critical conditions are achieved with bank 8 fully inserted OR reference bank is fully withdrawn.

____ 12.6.6 **IF** bank 8 is fully inserted with reference bank not fully withdrawn, direct Operations to adjust reference bank position to critical and record h_x^M on Enclosure 13.3.

12.6.7 **IF** bank 8 is NOT fully inserted and reference bank is fully withdrawn, perform the following:

____ 12.6.7.1 **IF** remaining worth of bank 8 to be inserted is estimated to be less than approximately 50 pcm, measure remaining worth by inserting bank 8 to 0 steps and measure worth using reactivity computer. Record worth on Enclosure 13.3 in column for $\alpha_x(\Delta\rho_2)_x$.

12.6.7.2 **IF** remaining worth of bank 8 to be inserted is estimated to be greater than approximately 50 pcm, perform the following:

____ a) Swap bank 8 for reference bank until bank 8 is fully withdrawn.

b) Record reference bank inserted, final critical point (h_x^M) final on Enclosure 13.5 and Enclosure 13.3 for bank 7.

____ c) On Enclosure 13.5, mark bank 8 drift as N/A and divide drift by 7 to get drift/bank.

____ d) Swap bank 8 for reference bank until reference bank is fully withdrawn.

NOTE:	It is permissible to insert another bank to maintain the reactor critical.

____ e) Direct Operations to commence a slow NC system dilution and measure remaining worth of bank 8 using reactivity computer.

____ 12.7 Direct Operations to insert reference bank until indicated reactivity is approximately - 40 pcm.

____ 12.8 Direct Operations to withdraw bank 8 until indicated reactivity is approximately + 40 pcm.

___ 12.9 Repeat Steps 12.7 and 12.8 maintaining indicated reactivity approximately ± 40 pcm, until bank 8 is fully withdrawn and critical conditions are achieved.

12.10 IF Step 12.6.7.2 was NOT performed, perform the following:

- Record $(h_x^M)_o$ on Enclosures 13.3 and 13.5.
- ___ Divide through by 8 on Step 13.5.6 of Enclosure 13.5.

___ 12.11 Complete Enclosure 13.5.

NOTE:	If computer application, RODSWAP is used, Step 12.12 and any unused blanks on Enclosures 13.3, 13.5, and 13.6 may be marked N/A.

12.12 Compute inferred worth for each control and shutdown bank (except reference bank) as follows:

12.12.1 Using data from Enclosure 13.3, and worth measurement data for reference bank from Enclosure 13.8, record value of $(\Delta\rho_1)_x$ on Enclosure 13.3.

___ 12.12.2 IF bank being measured has a worth greater than reference bank worth, replace $\alpha_x(\Delta\rho_2)_x$ with worth measured by reactivity computer:

$$\left[W_x^M\right]_{h_x^M}^{FW}$$

___ 12.12.3 Using data from Enclosure 13.3, worth measurement data for reference bank from Enclosure 13.8 and data of Enclosure 13.2, compute value of $\alpha_x(\Delta\rho_2)_x$ as described below and record on Enclosure 13.3:

$$\alpha_x(\Delta\rho_2)_x = \alpha_x\left[W_R^M\right]_{h_x^M}^{FW}$$

where: $\left[W_R^M\right]_{h_x^M}^{FW}$ is the measured integral worth of the reference bank from h_x^M to the fully withdrawn position from Enclosure 13.8. Linearly interpolate if h_x^M does not correspond to the steps on Enclosure 13.8.

h_x^M is the measured critical position of the reference bank after interchange with bank x from Enclosure 13.3.

and

α_x is a correction factor from Enclosure 13.2 to account for the influence of bank x on the worth of the reference bank.

___ 12.12.4 IF bank being measured has a worth greater than the reference bank worth, compute the inferred integral worth of the bank and record on Enclosure 13.3:

$$W_x^I = W_R^M + \left[W_x^M\right]_{h_x^M}^{FW} - (\Delta\rho_1)_x$$

where $\left[W_x^M\right]_{h_x^M}^{FW}$ is given in the column marked $\alpha_x(\Delta\rho_2)_x$ on Enclosure 13.3.

___ 12.12.5 Compute inferred integral worth of each bank x, W_x^I, as described below and record on Enclosure 13.3:

$$W_x^I = W_R^M - (\Delta\rho_1)_x - \alpha_x(\Delta\rho_2)_x$$

where: W_R^M is the measured total integral reference bank worth from Enclosure 13.8.

$(\Delta\rho_1)_x$ is from step 12.12.1.

and

$\alpha_x(\Delta\rho_2)_x$ is from step 12.12.2 or 12.12.3.

___ 12.12.6 Compute difference and percent difference between inferred and predicted worths for each individual RCC bank and the sum of all banks described below.

$$(\varepsilon_1)_x = \left(\frac{W_x^I}{W_x^P} - 1\right) \times 100\%$$

$$\varepsilon_2 = \left(\frac{\sum_{i=1}^{N} W_i^I}{\sum_{i=1}^{N} W_i^P}\right) \times 100\%$$

Fill in all blanks and summarize the calculations on Enclosure 13.6.

___ 12.13 IF computer application, RODSWAP, is used, attach printout to Enclosure 13.6.

___ 12.14 Complete Enclosure 13.7.

12.15 Verify all acceptance and review criteria have been met, or appropriate actions are being taken.

12.16 Any temporary signs provided for the OATC to assist in designating rod group being withdrawn and rod group being inserted should be removed from the Control Room upon completion of this test.

13.0 ENCLOSURES

13.1 Nomenclature
13.2 Nuclear Design Predictions for Rod Exchange Measurements
13.3 Critical Configuration and Worth Calculation Sheet
13.4 Additional Signoffs for Banks 3 through 7
13.5 Reference Bank Drift Evaluation
13.6 Comparison of Inferred Bank Worths with Design Predictions
13.7 Review Criteria Evaluation
13.8 Reference Bank Integral Worth
13.9 Requirements for Infrequently Performed Tests
13.10 Rod Control Cabinet Group Select Light Checkout

ENCLOSURE 13.1
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.	$(\Delta\rho_2)_x$	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.	$\left[W_R^M\right]_o^{(h_x^M)_o}$	The measured integral worth of the reference bank from 0 steps to $(h_x^M)_o$; equivalent to $(\Delta\rho_1)_x$.
9.	$(h_x^M)_o$	Initial critical position of the reference bank before interchange with bank x.
10.	$\left[W_R^M\right]_{h_x^M}^{FW}$	The measured integral worth of the reference bank from h_x^M to the fully withdrawn position.

ENCLOSURE 13.2
NUCLEAR DESIGN PREDICTIONS
FOR ROD EXCHANGE MEASUREMENTS

McGuire Unit _____ Cycle _____

Bank No. (x)	Bank Identity +	W_x^P (pcm)	(b) h_x^P (steps)	(c) α_x
(a) Reference			N/A	N/A
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 bank x in the core.

\+ Control Bank C, Shutdown Bank E, etc.

NOTE: See Enclosure 13.1 for a complete listing of nomenclature used in this test.

Recorded By ______________________ Date ____________________

This data came from (list source and document number): ____________________________

____

McGuire Unit _____ Cycle _____

ENCLOSURE 13.3
CRITICAL CONFIGURATION AND WORTH CALCULATION SHEET

Bank (x)	Date/Time	$(h_x^M)_o$	(h_x^M)	* $(\Delta\rho_1)_x$	* $\alpha_x(\Delta\rho_2)_x$	* W_x^I
No. Ident.	N/A	(steps)	(steps)	(pcm)	(pcm)	(pcm)
1						
2		N/A				
3		N/A				
4		N/A				
5		N/A				
6		N/A				
7						
8						

* NOTE: IF bank being measured has a worth greater than the reference bank worth, these values will be as given by Enclosure 13.5 or Step 12.4.7.2.

Recorded By___________________Date__________

CNG #25
TDR 1/24/96

ENCLOSURE 13.4
ADDITIONAL SIGNOFFS FOR BANKS 3 THROUGH 7

Bank	3	4	5	6	7
Step					
12.5.1	____	____	____	____	____
12.5.2	____	____	____	____	____
12.5.3	____	____	____	____	____
12.5.4	____	____	____	____	____
12.5.5	____	____	____	____	____
12.5.6	____	____	____	____	____
12.5.7	____	____	____	____	____
12.5.8	____	____	____	____	____

Section 12.5 Performed By/Date: ____

ENCLOSURE 13.5
REFERENCE BANK DRIFT EVALUATION

McGuire Unit _____ Cycle _____

Step

13.5.1 Final Reference Bank Critical Position __________ steps

13.5.2 Initial Reference Bank Critical Position __________ steps

13.5.3 Reactivity worth of reference bank from 0 to position of Step 13.5.1 __________ pcm

13.5.4 Reactivity worth of reference bank from 0 to position of Step 13.5.2. __________ pcm

13.5.5 Difference of Step 13.5.3 and 13.5.4 (Circle correct sign).

13.5.3 - 13.5.4 = ________ - ________ = ±__________ pcm

NOTE: Round Step 13.5.6 to the nearest pcm.

13.5.6 Incremental drift for each bank (Circle correct sign and circle either 8 or 7 as appropriate) (See Step 12.4.7.2.c)

Step 13.5.5 / 8 or 7 = __________ / 8 or 7 ±__________ pcm

13.5.7 $(\rho_1)_x$ for banks:

bank 1	Step 13.5.4		__________ pcm
bank 2	Step 13.5.4 + 13.5.6	_______ + _______	__________ pcm
bank 3	bank 2 + 13.5.6	_______ + _______	__________ pcm
bank 4	bank 3 + 13.5.6	_______ + _______	__________ pcm
bank 5	bank 4 + 13.5.6	_______ + _______	__________ pcm
bank 6	bank 5 + 13.5.6	_______ + _______	__________ pcm
bank 7	bank 6 + 13.5.6	_______ + _______	__________ pcm
bank 8	bank 7 + 13.5.6	_______ + _______	__________ pcm

Recorded By ____________________ Date __________

Checked By ____________________ Date __________

ENCLOSURE 13.6
COMPARISON OF INFERRED BANK WORTHS WITH DESIGN PREDICTIONS

McGuire Unit _____ Cycle _____

NOTE: Round rod worth numbers to the nearest pcm.

Bank (x) No.	Ident.	* W_x^P (pcm)	++ W_x^I (pcm)	$(W_x^P - W_x^I)$ (pcm)	$(\varepsilon_1)_x$ (%)
Reference			+		
1					
2					
3					
4					
5					
6					
7					
8					
		$\sum W_x^P$ (pcm)	$\sum W_x^I$ (pcm)		ε_2 (%)

+Record the measured worth of the reference bank here.

*from Enclosure 13.2

++from Enclosure 13.3

Recorded By ______________________ Date______

Checked By ______________________ Date______

ENCLOSURE 13.7
REVIEW CRITERIA EVALUATION

McGuire Unit _____ Cycle _____

NOTE: IF any of the below Review Criteria are checked "No", notify G.O. Nuclear Design by the next working day.

		Yes (√)	No (√)
I.	Review Criteria 11.2.1: sum of all banks (ε_2) from Enclosure 13.6 is $\leq$110%.	___	___
II.	Review Criteria 11.2.2: for each bank x $(\varepsilon_1)_X$ from Enclosure 13.6 is $\pm$15% or $(W_x^P - W_x^I)$ from Enclosure 13.6 is $\pm$100 pcm, whichever is greater.	___	___

Bank x No.	Ident.	Yes (√)	No (√)
1	_____	___	___
2	_____	___	___
3	_____	___	___
4	_____	___	___
5	_____	___	___
6	_____	___	___
7	_____	___	___
8	_____	___	___

Recorded by ______________________________ Date ______________

Checked by ______________________________ Date ______________

ENCLOSURE 13.9
REQUIREMENTS FOR INFREQUENTLY
PERFORMED TESTS

This test, which involves exchanging (swapping) a bank with either the Reference Bank and/or the previous bank to measure its reactivity worth, involves additional requirements and management involvement since it is an infrequently performed test. The guidance in this enclosure establishes an environment that places a high priority on preserving the plant's nuclear safety which is management's prime responsibility.

The Management Designee's responsibility is to ensure management expectations are met and that the evolution is controlled appropriately. The Management Designee can stop the evolution at any point that is deemed necessary or appropriate and provide the Operations Shift Supervisors with guidance for any recovery actions.

The Evolution Coordinator's responsibility is overall coordination of the evolution to ensure it is done in a safe controlled manner. The Evolution Coordinator can stop the evolution at any point that is deemed necessary or appropriate and provide the Operations Shift Supervisor with guidance for any recovery actions. (Reference SOER 91-01)

The Management Designee shall initial and date the steps below when completed.

____ 1,0 Record the following:

Evolution Coordinator ______________________________

Management Designee ______________________________

____ 2.0 A pre-job briefing has been performed by the Management Designee.

ENCLOSURE 13.10
ROD CONTROL CABINET
GROUP SELECT LIGHT CHECKOUT

NOTE: Shutdown and control banks may be done in any order.

(✓)

_____ 1.0 SHUTDOWN BANK A (SDA)

1.1 Have OATC select SDA on "CRD BANK SELECT"

1.2 Verify that only "GRP SELECT" light "C" is illuminated on Rod Control Power Cabinets 1AC and 2AC.

_____ 2.0 SHUTDOWN BANK B (SDB)

2.1 Have OATC to select SDB on "CRD BANK SELECT"

2.2 Verify that only "GRP SELECT" light "C" is illuminated on Rod Control Power Cabinets 1BD and 2BD.

_____ 3.0 SHUTDOWN BANK C (SDC)

3.1 Have OATC to select SDC on "CRD BANK SELECT"

3.2 Verify that only "GRP SELECT" light "A" is illuminated on Rod Control Power Cabinet SCDE .

_____ 4.0 SHUTDOWN BANK D (SDD)

4.1 Have OATC to select SDD on "CRD BANK SELECT"

4.2 Verify that only "GRP SELECT" light "B" is illuminated on Rod Control Power Cabinet SCDE.

_____ 5.0 SHUTDOWN BANK E (SDE)

5.1 Have OATC to select SDE on "CRD BANK SELECT"

5.2 Verify that only "GRP SELECT" light "C" is illuminated on Rod Control Power Cabinet SCDE.

_____ 6.0 CONTROL BANK A (CBA)

6.1 Have OATC to select CBA on "CRD BANK SELECT"

6.2 Verify that only "GRP SELECT" light "A" is illuminated on Rod Control Power Cabinets 1AC and 2AC.

_____ 7.0 CONTROL BANK B (CBB)

7.1 Have OATC to select CBB on "CRD BANK SELECT"

ENCLOSURE 13.10
ROD CONTROL CABINET
GROUP SELECT LIGHT CHECKOUT

7.2 Verify that only "GRP SELECT" light "A" is illuminated on Rod Control Power Cabinets 1BD and 2BD.

_____ 8.0 CONTROL BANK C (CBC)

8.1 Have OATC to select CBC on "CRD BANK SELECT"

8.2 Verify that only "GRP SELECT" light "B" is illuminated on Rod Control Power Cabinets 1AC and 2AC.

_____ 9.0 CONTROL BANK D (CBD)

9.1 Have OATC to select CBD on "CRD BANK SELECT"

9.2 Verify that only "GRP SELECT" light "B" is illuminated on Rod Control Power Cabinet 1BD and 2BD.

_____ 10.0 IF any expected response is not received, contact Work Control Shift Work Manager to have E Work Order generated for troubleshoot/repair.

Performed By ______________________________ Date ______________

Verified By ______________________________ Date ______________

Attachment 5
Responses to NRC Concern on
Topical Report Numbered DPC-NE-2009-P, Revision 1 Westinghouse Fuel Transition Report

NRC Concern: The proposed Revision 1 for Topical Report DPC-NE-2009 added a new reference document designated as Reference 6-39. This document, WCAP-15085, *Model Changes to the Westinghouse Appendix K Small Break LOCA NOTRUMP Evaluation Model: 1988 – 1997*, has not been approved by the NRC. In an NRC/Duke telephone conference held on July 24, 2002, NRC officials expressed a concern with the Duke proposal to reference an unapproved topical report in DPC-NE-2009-P, Revision 1.

Response

WCAP-15085 is a compilation of 10 CFR 50.46 reports related to the Westinghouse SBLOCA evaluation model previously reported individually to the NRC by Westinghouse pursuant to 10 CFR 50.46. The reference to WCAP-15085 has been deleted from Duke's proposed Revision 1 to DPC-NE-2009-P, since it is not totally applicable to McGuire and Catawba. Only those 10 CFR 50.46 reports applicable to McGuire and Catawba (identified as References 6-22, 6-28, and 6-39) are now referenced in the proposed Revision 1 to DPC-NE-2009-P. This is consistent with the current NRC-approved Revision 0 of DPC-NE-2009-P. Appropriate changes have been made to the affected pages of DPC-NE-2009-P and are included within Attachment 5 in both marked and reprinted versions.