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~~PROPRIETARY INFORMATION WITHHOLD UNDER 10 CFR 2.390~~  
UPON REMOVAL OF ATTACHMENTS 4 AND 7 THIS LETTER IS UNCONTROLLED

Serial: RA-16-0023  
May 4, 2016

10 CFR 50.90

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1  
DOCKET NO. 50-400 / RENEWED LICENSE NO. NPF-63

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2  
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

**SUBJECT: SUPPLEMENTAL INFORMATION FOR LICENSE AMENDMENT REQUEST  
REGARDING METHODOLOGY REPORT DPC-NE-1008-P**

**REFERENCES:**

1. Duke Energy letter, *Application to Revise Technical Specifications for Methodology Report DPC-NE-1008-P Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors"*, dated August 19, 2015 (ADAMS Accession No. ML15236A044)
2. Duke Energy letter, *Application to Revise Technical Specifications to Adopt Methodology Reports DPC-NF-2010 Revision 3 "Nuclear Physics Methodology for Reload Design" and DPC-NE-2011-P Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors"*, dated February 3, 2016 (ADAMS Accession No. ML16034A610)
3. Duke Energy letter, *Withdrawal of License Amendment Request Regarding Methodology Reports DPC-NF-2010 and DPC-NE-2011-P*, dated April 7, 2016 (ADAMS Accession No. ML16098A317)
4. NRC letter, *Shearon Harris Nuclear Power Plant, Unit No. 1 and H. B. Robinson Steam Electric Plant, Unit 2 – Withdrawal of Requested Licensing Action Regarding Duke Energy Progress, Inc., Application to Revise Technical Specifications to Adopt Methodology Reports Submitted to NRC for Acceptance Review (CAC Nos. MF7337 AND MF7338)*, dated April 14, 2016 (ADAMS Accession No. ML16098A540)

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UPON REMOVAL OF ATTACHMENTS 4 AND 7 THIS LETTER IS UNCONTROLLED

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Ladies and Gentlemen:

In References 1 and 2, Duke Energy Progress, Inc., referred to henceforth as "Duke Energy", submitted requests for amendments to the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP). Specifically, in Reference 1, Duke Energy requested NRC review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," and adoption of the methodology into the TS for SHNPP and HBRSEP. In Reference 2, Duke Energy requested NRC review and approval of DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and adoption of these two methodologies into the TS for SHNPP and HBRSEP.

References 3 and 4 document the withdrawal of Reference 2, with the intent to resubmit the request as a supplement to Reference 1. The purpose of this submittal is to provide that supplement.

This supplement supersedes Reference 1 in its entirety. The changes that this supplement makes to Reference 1 in order to incorporate Reference 2 include:

1. Revising the attachment "Evaluation of the Proposed Change" to add discussion of the DPC-NF-2010 and DPC-NE-2011-P methodology reports. In addition, the subject reports are not being approved under the topical report process, so the applicable discussions have been removed. Revision bars have been included in the left margin to denote the location of all changes.
2. Revising the attachment "Proposed Technical Specification Changes (Mark-Up)" to add the DPC-NF-2010 and DPC-NE-2011-P methods to the TS and to also reflect that the addition of DPC-NE-2005-P to the TS has been approved by the NRC (ML16049A630). In addition, for the DPC-NE-1008-P-A line item, the "-A" was removed and the Safety Evaluation date included, due to the fact that the report is not being approved under the topical report process.
3. Deleting the attachment "Retyped Technical Specification Pages." Retyped pages will be provided to the NRC Project Manager prior to issuance of the requested amendment.
4. Adding the following attachments
  - DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design" and Technical Justification of Changes
  - DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and Technical Justification of Changes (Proprietary)
  - DPC-NE-2011, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and Technical Justification of Changes (Redacted)
  - Application of the DPC-NE-2004-PA Operating Limits and Maximum Allowable Total Peak Methodology to Harris and Robinson Nuclear Plants
5. Revising the requested NRC approval date from December 31, 2016 to April 30, 2017

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This supplement affects the bases for concluding that the Reference 1 license amendment does not involve a significant hazards consideration; however, the conclusion remains the same, based on evaluation in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c). The bases for this determination are included in Attachment 2. Attachment 2 provides an evaluation of the proposed change. Attachment 3 provides the existing TS pages marked up to show the proposed change.

Attachment 4 contains DPC-NE-1008-P, which includes information that is proprietary to Duke Energy. In accordance with 10 CFR 2.390, Duke Energy requests that Attachment 4 be withheld from public disclosure. An affidavit is included (Attachment 1) attesting to the proprietary nature of the information. A non-proprietary version of Attachment 4 is included in Attachment 5. Attachment 6 contains DPC-NF-2010, showing the changes from Revision 2 to Revision 3, as well as the technical justification for each change. Attachment 7 contains DPC-NE-2011-P, showing the changes from Revision 1 to Revision 2, as well as the technical justification for each change. Attachment 7 also includes information that is proprietary to Duke Energy, is requested to be withheld from public disclosure in accordance with 10 CFR 2.390, and is attested to in Attachment 1. A non-proprietary version of Attachment 7 is included in Attachment 8.

The DPC-NE-2011-P methodology incorporates by reference the thermal-hydraulic methodology described in the NRC approved DPC-NE-2004-PA methodology report. No revisions to this methodology are required for application to SHNPP and HBRSEP. Attachment 9 is provided to clarify the application of the generic thermal-hydraulic methodology to these units.

As stated above, requested approval of the proposed amendment is revised from December 31, 2016 to April 30, 2017. The requested implementation period of 120 days to allow for updating the TS and Facility Operating License is not changed.

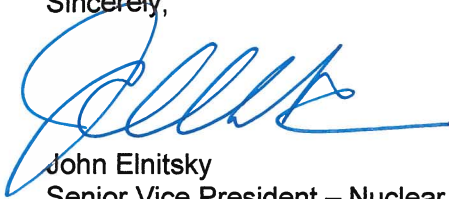
This submittal contains no new regulatory commitments. In accordance with 10 CFR 50.91, Duke Energy is notifying the states of North Carolina and South Carolina of this license amendment request by transmitting a copy of this letter to the designated state officials. Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba, Manager – Nuclear Fleet Licensing, at 980-373-2062.

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on May 4, 2016.

Sincerely,



John Elnitsky  
Senior Vice President – Nuclear Engineering

JBD

- Attachments:
1. Affidavit of John Elnitsky
  2. Evaluation of the Proposed Change
  3. Proposed Technical Specification Changes (Mark-Up)
  4. DPC-NE-1008-P, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors" (Proprietary)
  5. DPC-NE-1008, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors" (Redacted)
  6. DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design" and Technical Justification of Changes
  7. DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and Technical Justification of Changes (Proprietary)
  8. DPC-NE-2011, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and Technical Justification of Changes (Redacted)
  9. Application of the DPC-NE-2004-PA Operating Limits and Maximum Allowable Total Peak Methodology to Harris and Robinson Nuclear Plants

cc: (all with Attachments unless otherwise noted)

W. L. Cox, III, Section Chief, NC DHSR (Without Attachments 4 and 7)  
S. E. Jenkins, Manager, Radioactive and Infectious Waste Management Section (SC)  
(Without Attachments 4 and 7)  
Attorney General (SC) (Without Attachments 4 and 7)  
A. Gantt, Chief, Bureau of Radiological Health (SC) (without Attachments 4 and 7)

Attachment 1  
RA-16-0023

**Attachment 1**  
**Affidavit of John Elnitsky**

AFFIDAVIT of John Elnitsky

1. I am Senior Vice President of Nuclear Engineering, Duke Energy Corporation, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke Energy.
2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy's application for withholding which accompanies this affidavit.
3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential. I am familiar with the Duke Energy information contained in the proprietary version of the Duke methodology reports DPC-NE-1008-P, *Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors*, and DPC-NE-2011-P, *Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*.
4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, 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 Energy and has been held in confidence by Duke Energy and its consultants.
  - (ii) The information is of a type that would customarily be held in confidence by Duke Energy. Information is held in confidence if it falls in one or more of the following categories.
    - (a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by a vendor or consultant, without a license from Duke Energy, would constitute a competitive economic advantage to that vendor or consultant.
    - (b) The information requested to be withheld consist of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage for example by requiring the vendor or consultant to perform test measurements, and process and analyze the measured test data.
    - (c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation assurance of quality or licensing of a similar product.
    - (d) The information requested to be withheld reveals cost or price information, production capacities, budget levels or commercial strategies of Duke Energy or its customers or suppliers.

- (e) The information requested to be withheld reveals aspects of the Duke Energy funded (either wholly or as part of a consortium ) development plans or programs of commercial value to Duke Energy.
- (f) The information requested to be withheld consists of patentable ideas.

The information included in DPC-NE-1008-P and DPC-NE-2011-P is held in confidence for the reasons set forth in paragraphs 4(ii)(a) and 4(ii)(c) above. Rationale for this declaration is the use of this information by Duke Energy provides a competitive advantage to Duke Energy over vendors and consultants, its public disclosure would diminish the information's marketability, and its use by a vendor or consultant would reduce their expenses to duplicate similar information. 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 Energy.

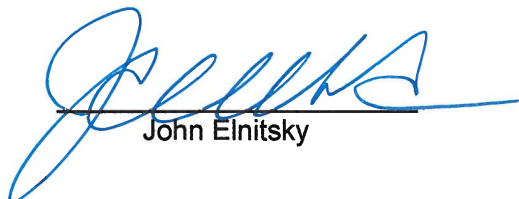
- (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.
  - (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.
  - (v) The proprietary information sought to be withheld is that which is marked in the proprietary version of the Duke methodology reports DPC-NE-1008-P, *Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors*, and DPC-NE-2011-P, *Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*. This information enables Duke Energy to:
    - (a) Support license amendment requests for its Harris and Robinson reactors.
    - (b) Support reload design calculations for Harris and Robinson reactor cores.
  - (vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke Energy.
    - (a) Duke Energy uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.
    - (b) Duke Energy can 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 Energy.
5. Public disclosure of this information is likely to cause harm to Duke Energy because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke Energy to recoup a portion of its expenditures or benefit from the sale of the information.

Attachment 1  
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John Elnitsky 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.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on May 4, 2016.



John Elnitsky



## Attachment 2

### EVALUATION OF THE PROPOSED CHANGE

Subject: APPLICATION TO REVISE TECHNICAL SPECIFICATIONS FOR METHODOLOGY REPORTS DPC-NE-1008-P REVISION 0, "NUCLEAR DESIGN METHODOLOGY USING CASMO-5/SIMULATE-3 FOR WESTINGHOUSE REACTORS," DPC-NF-2010 REVISION 3, "NUCLEAR PHYSICS METHODOLOGY FOR RELOAD DESIGN," AND DPC-NE-2011-P REVISION 2, "NUCLEAR DESIGN METHODOLOGY REPORT FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS"

- 1.0 SUMMARY DESCRIPTION
- 2.0 DETAILED DESCRIPTION
- 3.0 TECHNICAL EVALUATION
- 4.0 REGULATORY EVALUATION
  - 4.1 Applicable Regulatory Requirements/Criteria
  - 4.2 Precedent
  - 4.3 No Significant Hazards Consideration Determination
  - 4.4 Conclusions
- 5.0 ENVIRONMENTAL CONSIDERATION
- 6.0 REFERENCES

## 1.0 SUMMARY DESCRIPTION

Duke Energy requests amendments to Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP) Technical Specifications (TSs) pursuant to 10 CFR 50.90, to support the allowance of Duke Energy to perform nuclear physics calculations and reload design analysis as described in DPC-NE-1008-P, DPC-NF-2010, and DPC-NE-2011. AREVA currently performs the subject analyses for SHNPP and HBRSEP. The proposed change requests review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and subsequent inclusion of the three methodologies into the SHNPP and HBRSEP Technical Specifications.

## 2.0 DETAILED DESCRIPTION

### DPC-NE-1008-P

AREVA currently performs core physics reload design analyses for SHNPP and HBRSEP using the CASMO-3 and PRISM codes. Duke Energy currently performs core physics reload design analysis for McGuire and Catawba nuclear stations with the methodology described in DPC-NE-1005-P-A, approved in References 1 and 2. This methodology uses CASMO-4/SIMULATE-3 analytical models. The intent of this proposed change is to extend the CASMO/SIMULATE reload design capability to HBRSEP and SHNPP. In addition, the DPC-NE-1008-P report provides the basis for the future upgrade of the currently licensed CASMO-4/SIMULATE-3 methodology used at McGuire and Catawba.

As part of a continuous effort to improve design methods, Duke Energy is seeking to use CASMO-5 based SIMULATE-3 models in lieu of CASMO-4 based models for HBRSEP and SHNPP nuclear analyses. Therefore, DPC-NE-1008-P, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," is presented to seek NRC approval of the CASMO-5/SIMULATE-3 code system for performing Duke Energy reload design calculations. This report demonstrates the acceptability of the CASMO-5/SIMULATE-3 code system for performing reload calculations for the three-loop and four-loop reactor cores characteristic of the Harris/Robinson and Catawba/McGuire nuclear steam supply systems. Reactor fuel encompassing both 15x15 and 17x17 lattice geometries are analyzed. In addition, reactor cores containing fuel with B4C discrete burnable absorbers, zirconium diboride integral fuel rod burnable absorbers (IFBA), and gadolinia integral fuel burnable absorbers are analyzed. This diverse set of benchmarks is used to demonstrate the accuracy of the CASMO-5/SIMULATE-3 models for use in calculating reload physics parameters, core reactivity and power distributions for use in reload design analyses at all four Duke Energy Westinghouse reactor sites. However, request for approval of implementation is limited to the TSs for HBRSEP and SHNPP at this time.

A detailed description of the CASMO-5/SIMULATE-3 computer code system is presented in Section 2 of the attached DPC-NE-1008-P report.

Upon NRC approval, DPC-NE-1008-P, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," will be added to HBRSEP TS Section 5.6.5.b and SHNPP TS Section 6.9.1.6.2, as shown in Attachment 3.

The CASMO-5/SIMULATE-3 methodology in DPC-NE-1008-P will be used to perform core physics analyses and confirmation of thermal limits as a portion of the overall Duke Energy methodology for cycle reload safety analyses. There are additional methodology reports and analyses related to the application of the CASMO-5/SIMULATE-3 methodology. These reports address transient analytical models and reload methods along with methods for establishing operational Axial Flux Difference (AFD) limits. DPC-NF-2010 and DPC-NE-2011-P are two such reports that address application of the CASMO-5/SIMULATE-3 methodology. Subsequent submittals will be made for staff approval. Therefore, the appropriate SHNPP Final Safety Analysis Report (FSAR) and HBRSEP Updated Final Safety Analysis Report (UFSAR) changes will be processed once core designs using the methodology addressed by this LAR (and the methodologies which will be the subject of subsequent LARs) are implemented.

#### DPC-NF-2010 and DPC-NE-2011-P

DPC-NF-2010, "Nuclear Physics Methodology for Reload Design," describes Duke Energy's reload design process. Key elements of the report include an overview of the reload design process, the methodology for calculating nuclear physics parameters used to confirm safety analysis assumptions and to support the startup and operation of a reactor core, and the types of comparisons performed to qualify a nuclear code. A summary of previously approved statistical methods for developing peaking factor uncertainties is also included. This methodology is only approved for use at McGuire and Catawba Nuclear Stations, and therefore requires NRC approval prior to being used for performing reload analyses at other Duke Energy Westinghouse reactors.

DPC-NE-2011-P, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," describes the analysis methodology used to generate transient power distributions. These power distributions are subsequently used to develop core operational axial flux difference (AFD) limits, rod insertion limits (RILs) and  $f(\Delta I)$  limits for the over-power delta-temperature (OP $\Delta T$ ) and over-temperature delta-temperature (OT $\Delta T$ ) reactor protection system (RPS) trip functions. DPC-NE-2011-P also describes the types of  $F_Q$  and  $F_{\Delta H}$  power distribution surveillances that should be performed. This methodology, like DPC-NF-2010, is only approved for use at McGuire and Catawba Nuclear Stations, and requires NRC approval prior to being used for other Duke Energy Westinghouse reactors.

Upon NRC approval, DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design" and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" will both be added to HBRSEP TS Section 5.6.5.b and SHNPP TS Section 6.9.1.6.2, as shown in Attachment 3.

DPC-NF-2010 describes the reload design process and DPC-NE-2011-P describes the core operating limits methodology. These reports represent two elements of the overall Duke Energy methodology for cycle reload safety analyses. There are additional methodology reports and analyses which will be submitted for staff approval. Some reports have already been submitted to the staff for approval, others will be provided in the future. Therefore, the appropriate SHNPP Final Safety Analysis Report (FSAR) and HBRSEP Updated Final Safety Analysis Report (UFSAR) changes will be processed once core designs using the methodology addressed by this LAR (and the methodologies addressed in the additional LARs) are implemented.

### 3.0 TECHNICAL EVALUATION

#### DPC-NE-1008-P

The qualification of the CASMO-5/SIMULATE-3 code system is accomplished through the same types of benchmark calculations used in DPC-NE-1005-P-A.

The Harris and Robinson benchmarks are for a 3-loop 157 fuel assembly core with AREVA's high thermal performance (HTP) fuel design with gadolinia integral burnable absorbers. Differences between the Harris and Robinson core designs primarily relate to differences in the fuel lattice design, rated thermal power level and operating temperature. A large range of gadolinia concentrations and absorber patterns are contained in the Harris and Robinson benchmark cycles. The McGuire benchmark calculations are for a 4-loop 193 fuel assembly reactor core with Westinghouse's 17x17 Robust Fuel Assembly (RFA) design. The core designs evaluated employ a variety of IFBA patterns in combination with discrete burnable absorbers for reactivity and peaking control. The McGuire benchmarks are characteristic in fuel management strategy and fuel type used at all of Duke Energy's four-loop reactors (McGuire Units 1 and 2, and Catawba Units 1 and 2), and are therefore representative of this reactor type. The McGuire benchmarks demonstrate the accuracy of the core models for fuel with discrete and zirconium diboride integral fuel rod burnable absorbers, which provides flexibility of the methodology if a fuel design change is ever sought at SHNPP or HBRSEP.

Operational benchmarks against measured reactivity and power distribution data from McGuire, Harris and Robinson demonstrate the ability of the methodology to predict core physics parameters and power distributions. Section 3 of the attached DPC-NE-1008-P report presents these benchmarks. The power distribution comparisons are also used to determine assembly uncertainty factors, or Observed Nuclear Reliability Factors (ONRFs) for  $F\Delta H$ ,  $F_q$  and  $F_z$ . The benchmark results presented demonstrate the ability of the CASMO-5/SIMULATE-3 code package to adequately model the behavior of these reactor cores.

Qualification of the CASMO-5/SIMULATE-3 methodology for predicting pin power distributions in low-enriched uranium (LEU) fuel lattices with and without gadolinia is based on comparison of predicted pin power distributions against measured data from the B&W Urania Gadolinia critical experiments, as well as power distribution comparisons of CASMO-5 and SIMULATE-3 for gadolinia fuel pins. This work is detailed in Section 4 of DPC-NE-1008-P, which also quantifies the pin power distribution uncertainty. The results show that CASMO-5 and SIMULATE-3 accurately predict pin power distributions in fuel lattices with and without gadolinia.

In Section 5 of DPC-NE-1008-P, the assembly uncertainties and pin uncertainties are statistically combined to calculate the  $F\Delta H$ ,  $F_q$  and  $F_z$  peaking factor uncertainties applied in reload design analyses, with separate values for fuel pins with and without gadolinia burnable absorbers.

In summary, DPC-NE-1008-P concludes that the CASMO-5/SIMULATE-3 methodology is acceptable for use in performing reactivity and power distribution reactor core calculations for input into safety-related reload design analyses for reactor cores containing LEU fuel.

DPC-NF-2010 and DPC-NE-2011-P

Revision 3 of DPC-NF-2010 (Attachment 6) includes changes necessary to extend the applicability of this methodology to SHNPP and HBRSEP. The majority of the changes are editorial and clarification type changes to add descriptions of the Harris and Robinson Nuclear Plants to the report. The CASMO-5/SIMULATE-3 nuclear analysis methodology is also incorporated into the report by reference for application to Harris and Robinson reload designs.

The primary focus of Revision 2 of DPC-NE-2011-P (Attachment 7) is to extend the core operating limits methodology to the Harris and Robinson Nuclear Plants. The majority of changes made for this purpose are clarification and editorial type changes. Several methodology changes are also proposed based on experience with using the methodology and to address a unique aspect of the Harris reactor protection system. These changes include:

- Revising the methodology for establishing initial condition xenon distributions to exclude xenon shapes that operationally cannot occur
- Adding an option to adjust the over-temperature delta-temperature ( $OT\Delta T$ )  $f_1(\Delta I)$  trip reset function breakpoints if the  $F_Q$  centerline fuel melt surveillance limit is exceeded
- Removing the quadrant power tilt allowance from the loss of coolant accident (LOCA)  $F_Q$ , centerline fuel melt (CFM)  $F_Q$ , and loss of flow accident (LOFA)  $F\Delta H$  surveillance equations.

The base load operational mode option was also removed from the methodology.

The methodology for establishing initial condition xenon distributions for initiating a xenon transient is being revised because the original methodology allowed generation of core axial power distributions that were significantly wider than the final operational axial flux difference space and operationally could not occur. The option to adjust the breakpoints of the  $OT\Delta T$   $f_1(\Delta I)$  trip reset penalty function is added because the  $OP\Delta T$   $f_2(\Delta I)$  trip reset penalty function at Harris is not active. The  $OP\Delta T$   $f_2(\Delta I)$  trip reset penalty function is available at McGuire, Catawba and Robinson.

Removal of the quadrant power tilt allowance from the LOCA  $F_Q$ , CFM  $F_Q$  and LOFA  $F\Delta H$  surveillance equations is being made because any quadrant power tilt present in the reactor core is implicit in the power distribution measurement. Continuous monitoring of plant parameters (i.e. quadrant power tilt ratio and rod insertion) available to reactor operators and their associated alarms would provide indication of a gross change in the radial power distribution prior to the next power distribution measurement. This available indication, along with conservatisms in the power distribution methodology and peaking margin between steady state and transient conditions, provides assurance that  $F_Q$  and  $F\Delta H$  power peaking limits will not be exceeded prior to the next power distribution measurement. Design analyses which establish the limits for AFD, RIL, and  $f(\Delta I)$  will continue to include a quadrant power tilt penalty which accounts for a change in the quadrant power tilt ratio of 1.02 in the verification of LOCA, departure from nucleate boiling (DNB), and CFM thermal limits. This change would also bring the DPC-NE-2011-P surveillance methodology in line with the current Harris and Robinson  $F\Delta H$  and  $F_Q$  surveillance methodology, and standard technical specifications for Westinghouse plants (NUREG-1431, "Standard Technical Specifications for Westinghouse Plants") where a quadrant power tilt penalty is not applied

in heat flux hot channel factor  $F_Q$  and nuclear enthalpy rise hot channel factor  $F_{\Delta H}$  power distribution surveillances.

DPC-NE-2011-P (Attachment 7), sections 4.3 and 4.4, references the thermal-hydraulics methodology described in DPC-NE-2004-PA and DPC-NE-2005-P. DPC-NE-2004-PA, "Core Thermal-Hydraulic Methodology Using VIPRE-01," describes Duke Energy's NRC-approved steady state thermal-hydraulics analysis methodology for McGuire and Catawba Nuclear Stations using the VIPRE-01 computer code. It also includes the thermal-hydraulic methodology used to develop allowable operating limits that provide DNB protection. DPC-NE-2005-P, "Duke Energy Thermal-Hydraulic Statistical Core Design Methodology," describes the code inputs and correlations and inputs used to develop the Harris and Robinson VIPRE-01 models. The statistical DNBR SCD limit is also included in this report for fuel in operation at Harris and Robinson. Note that NRC approval of the DPC-NE-2005-P methodology and adoption into the SHNPP and HBRSEP TSs was received on March 8, 2016 (ML16049A630). However, this amendment resubmits the DPC-NE-2011-P report that was submitted to the NRC on February 3, 2016 (ML16034A610) and subsequently withdrawn. Thus, the attached DPC-NE-2011-P report continues to refer to DPC-NE-2005-P as though it is still under NRC review.

The thermal-hydraulics methodology from DPC-NE-2004-PA is used to develop the loss of flow accident and RPS Maximum Allowed Total Peak curves. These curves are used in the DPC-NE-2011-P methodology to verify positive DNB margins exists to the applicable limits. Attachment 9 is provided to clarify the application of the generic thermal-hydraulic methodology described in Section 5 of DPC-NE-2004-PA for Harris and Robinson.

Detailed technical justifications supporting DPC-NF-2010 and DPC-NE-2011-P are included in Attachments 6 and 7.

#### 4.0 REGULATORY EVALUATION

##### 4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criterion (GDC) 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. SHNPP is licensed to GDC 10 and this proposed change will not affect the SHNPP conformance to GDC 10.

HBRSEP was not licensed to the current 10 CFR 50, Appendix A, GDC. Per the HBRSEP UFSAR, it was evaluated against the proposed Appendix A to 10 CFR 50, General Design Criteria for Nuclear Power Plants, published in the Federal Register on July 11, 1967.

Criterion 6, "Reactor Core Design," of the July 11, 1967 proposed Appendix A requires that:

"The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power."

This proposed change will not affect the HBRSEP conformance to the July 11, 1967 proposed Appendix A Criterion 6.

#### 4.2 Precedent

##### DPC-NE-1008-P

The current Duke Energy nuclear design methodology report for McGuire and Catawba nuclear stations is DPC-NE-1005-P-A, "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," Revision 1, and for Oconee nuclear station is DPC-NE-1006-P-A, "Oconee Nuclear Design Methodology using CASMO-4 / SIMULATE-3." The methods used for code qualification and calculating assembly and pin power uncertainties are the same in DPC-NE-1008-P as in both of these previously approved reports. The principal differences are the substitution of CASMO-5 for CASMO-4 and the plants which are modeled and analyzed. These methods were approved by the NRC in References 1, 2, and 3.

##### DPC-NF-2010 and DPC-NE-2011-P

DPC-NF-2010 and DPC-NE-2011-P are the methodologies approved for use at McGuire and Catawba Nuclear Stations. DPC-NF-2010 Revisions 0, 1, and 2 were approved by the NRC in Reference 4 (Revision 0), References 5 and 6 (Revision 1), and Reference 7 (Revision 2). DPC-NE-2011-P Revisions 0 and 1 were approved by the NRC in Reference 8 (Revision 0) and References 5 and 6 (Revision 1). Note that this amendment does not affect McGuire and Catawba Nuclear Stations.

Extension of the previously approved DPC-NF-2010 and DPC-NE-2011-P methodologies to SHNPP and HBRSEP involves application of the previously approved methods in whole with several modifications made to improve the method based on experience with using the methodology, and as the result of as-built plant differences. The modifications are described in Attachments 6 and 7.

#### 4.3 No Significant Hazards Consideration Determination

Duke Energy Progress, Inc., referred to henceforth as "Duke Energy", requests NRC review and approval of methodology reports DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" and adoption of these three methodologies into the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP).

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change requests review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," to be applied to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). The CASMO-5 and SIMULATE-3 codes are not used in the operation of any plant equipment. The benchmark calculations performed confirm the accuracy of the codes and develop a methodology for calculating power distribution uncertainties for use in reload design calculations. The use of power distribution uncertainties in conjunction with predicted peaking factors ensures that thermal accident acceptance criteria are satisfied. The proposed use of this methodology does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

The proposed change also requests review and approval of DPC NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" to be applied to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). The proposed change supports the use of revised McGuire and Catawba reload design methodologies for performance of reload design analyses at Harris and Robinson Nuclear Plants. Implementation of the methodologies will occur following approval by the NRC. The proposed amendments will have no impact upon the probability of occurrence of any design basis accident, nor will they affect the performance of any plant equipment used to mitigate the consequences of an analyzed accident. There will be no significant impact on the source term or pathways assumed in accidents previously evaluated. No analysis assumptions will be violated and there will be no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change requests review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," to be applied to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). It does not change any system functions or maintenance activities. The change does not involve physical alteration of the plant, that is, no new or different type of equipment will be installed. The software is not installed in any plant equipment, and therefore the software is incapable of initiating an equipment malfunction that would result in a new or different type of accident from any previously



evaluated. The change does not alter assumptions made in the safety analyses but ensures that the core will operate within safe limits. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

The proposed change also requests review and approval of DPC NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" to be applied to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). The proposed amendments do not change the methods used for normal plant operation, nor are the methods used to respond to plant transients modified. Use of the DPC-NF-2010 and DPC-NE-2011-P methodologies does not result in a new or different type of accident from any previously evaluated. There are no changes to any system functions or maintenance activities. The change does not physically alter the plant, that is, no new or different type of equipment will be installed. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed change requests review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," to be applied to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). As with the existing methodology, the qualification of the methods therein and the use of power distribution uncertainties ensure the acceptability of analytical limits under normal, transient, and accident conditions. The use of the proposed methodology revision once it has been approved by the NRC will ensure that all applicable design and safety limits are satisfied such that the fission product barriers will continue to perform their design functions.

The proposed change also requests review and approval of DPC NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" to be applied to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). Application of the DPC NF-2010 and DPC-NE-2011-P methodologies will assure the acceptability of thermal limits assumed in the cycle reload safety analyses. As with the existing methodology, the Duke Energy methodology will continue to ensure (a) the acceptability of analytical limits under normal, transient, and accident conditions, and (b) that all applicable design and safety limits are satisfied such that the fission product barriers will continue to perform their design functions.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

#### 4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

#### 5.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

#### 6.0 REFERENCES

1. NRC letter, *McGuire Nuclear Station, Units 1 and 2 Issuance of Amendments Regarding Revision 1 to DPC-NE-1005-P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX (TAC Nos. MD7409 and MD7410)*, dated November 12, 2008 (ADAMS Accession No. ML082820015)
2. NRC letter, *Catawba Nuclear Station, Units 1 and 2 Issuance of Amendments Regarding Revision 1 to DPC-NE-1005-P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX (TAC Nos. MD7407 and MD7408)*, dated November 12, 2008 (ADAMS Accession No. ML082820047)
3. NRC letter, *Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding the Use of CASMO-4/SIMULATE-3 Methodology for Reactor Cores Containing Gadolinia Bearing Fuel (TAC Nos. ME4646, ME4647, and ME4648)*, dated August 2, 2011 (ADAMS Accession No. ML101580106)
4. NRC letter, *Topical Report on Physics Methodology for Reloads: McGuire and Catawba Nuclear Station*, dated March 13, 1985
5. NRC letter, *McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB3222 and MB3223)*, dated October 1, 2002 (ADAMS Accession No. ML022740527)
6. NRC letter, *Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB3343 and MB3344)*, dated October 1, 2002 (ADAMS Accession No. ML022740677)

7. NRC letter, *Catawba Nuclear Station, Units 1 and 2 and McGuire Nuclear Station, Units 1 and 2, Re: Topical Report DPC-NF-2010, Nuclear Physics Methodology for Reload Design, Revision 2*, dated June 24, 2003 (ADAMS Accession No. ML031760728)
8. NRC letter, *Acceptance for Referencing of Topical Report DPC-NE-2011-P, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors"*, dated January 24, 1990

**Attachment 3**

**Proposed Technical Specification Changes (Mark-up)**

## 5.6 Reporting Requirements

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### 5.6.2 Annual Radiological Environmental Operating Report (continued)

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

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted 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 the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

### 5.6.4 DELETED

### 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. Shutdown Margin (SDM) for Specification 3.1.1;
  2. Moderator Temperature Coefficient limits for Specification 3.1.3;
  3. Shutdown Bank Insertion Limits for Specification 3.1.5;
  4. Control Bank Insertion Limits for Specification 3.1.6;
  5. Heat Flux Hot Channel Factor ( $F_Q(Z)$ ) limit for Specification 3.2.1;
  6. Nuclear Enthalpy Rise Hot Channel Factor ( $F_{\Delta H}^N$ ) limit for Specification 3.2.2;

(continued)

5.6 Reporting Requirements (continued)

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. Axial Flux Difference (AFD) limits for Specification 3.2.3; and
8. Boron Concentration limit for Specification 3.9.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The approved version shall be identified in the COLR. These methods are those specifically described in the following documents:
  1. Deleted
  2. XN-NF-84-73(P), "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," approved version as specified in the COLR.
  3. XN-NF-82-21(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.
  4. Deleted
  5. XN-75-32(A), "Computational Procedure for Evaluating Rod Bow," approved version as specified in the COLR.
  6. Deleted.
  7. Deleted
  8. XN-NF-78-44(A), "Generic Control Rod Ejection Analysis," approved version as specified in the COLR.
  9. XN-NF-621(A), "XNB Critical Heat Flux Correlation," approved version as specified in the COLR.
  10. Deleted
  11. XN-NF-82-06(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.
  12. Deleted
  13. Deleted.

(continued)

5.6 Reporting Requirements (continued)

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. Deleted
15. Deleted
16. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.
17. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," approved version as specified in the COLR.
18. ANF-89-151(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.
19. EMF-92-081(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.
20. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.
21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.
22. EMF-96-029(P)(A), "Reactor Analysis System for PWRs," approved version as specified in the COLR.
23. EMF-92-116, "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.
24. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.

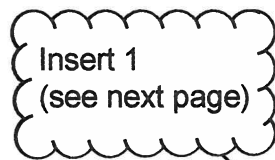
(continued)

5.6 Reporting Requirements (continued)

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

25. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.
26. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods," approved version as specified in the COLR.
27. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," approved version as specified in the COLR.
28. DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," approved version as specified in the COLR.



- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status,

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(continued)



Insert 1:

29. DPC-NE-1008-P, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," as approved by NRC Safety Evaluation dated [Month xx, xxxx].
30. DPC-NF-2010, "Nuclear Physics Methodology for Reload Design," as approved by NRC Safety Evaluation dated [Month xx, xxxx].
31. DPC-NE-2011-P, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" as approved by NRC Safety Evaluation dated [Month xx, xxxx].

## ADMINISTRATIVE CONTROLS

### 6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

- a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.2.
- b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.
- c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.
- d. Control Bank Insertion Limits for Specification 3/4.1.3.6.
- e. Axial Flux Difference Limits for Specification 3/4.2.1.
- f. Heat Flux Hot Channel Factor,  $F_{\Delta H}^{RTP}$ ,  $K(Z)$ , and  $V(Z)$  for Specification 3/4.2.2.
- g. Enthalpy Rise Hot Channel Factor,  $F_{\Delta H}^{RTP}$ , and Power Factor Multiplier,  $PF_{\Delta H}$  for Specification 3/4.2.3.
- h. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

- a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

- b. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

- c. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

No changes to this page. Included for information only.

## ADMINISTRATIVE CONTROLS

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### 6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.  
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.  
(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- f. ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, as approved by NRC Safety Evaluation dated May 30, 2012.  
(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.  
(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

## ADMINISTRATIVE CONTROLS

### 6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.  
(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).
- i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.  
(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.  
(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- k. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods."  
(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).
- l. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.  
(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).
- m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.  
(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- n. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", approved version as specified in the COLR.

## ADMINISTRATIVE CONTROLS

### 6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

p. DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

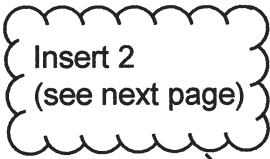
6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

### 6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.I. The report shall include:

- a. The scope of inspections performed on each SG,
- b. Degradation mechanisms found,
- c. Nondestructive examination techniques utilized for each degradation mechanism,



Insert 2:

- q. DPC-NE-1008-P, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," as approved by NRC Safety Evaluation dated [Month xx, xxxx].  
  
(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4 and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 – Boron Concentration).
- r. DPC-NF-2010, "Nuclear Physics Methodology for Reload Design," as approved by NRC Safety Evaluation dated [Month xx, xxxx].  
  
(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4 and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, and 3.9.1 – Boron Concentration).
- s. DPC-NE-2011-P, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" as approved by NRC Safety Evaluation dated [Month xx, xxxx].  
  
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Attachment 5  
RA-16-0023

**Attachment 5**

**DPC-NE-1008, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for  
Westinghouse Reactors” (Redacted)**

**Duke Energy Carolinas  
Nuclear Design Methodology  
Using CASMO-5/SIMULATE-3 for Westinghouse Reactors**

**DPC-NE-1008  
Revision 0**

**May 2015**

**Non-Proprietary Version**

Nuclear Fuel Engineering Division  
Nuclear Generation Department  
Duke Energy Carolinas, LLC





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## **Abstract**

The CASMO-5/SIMULATE-3 software package consists of a set of computer codes for performing nuclear analysis of pressurized water reactor cores. The primary codes in this software package are CASMO-5 and SIMULATE-3. Benchmark comparisons are presented to operating data from Harris, Robinson and McGuire fuel cycles, and measured data from the B&W Urania Gadolinia Critical Experiments. These benchmarks demonstrate the fidelity of the core models for analyzing reactor cores containing low enriched uranium fuel with integral and discrete burnable absorbers. The Harris and Robinson benchmarks demonstrate the accuracy of the core models for fuel with gadolinia integral fuel burnable absorbers, while the McGuire benchmarks demonstrate the accuracy of the core models for fuel with discrete and zirconium diboride ( $\text{ZrB}_2$ ) integral fuel rod burnable absorbers (referred to as IFBA). Results from the benchmark calculations are used to develop appropriate biases and uncertainty factors for use in reload calculations supporting the design and operation of reload cores. Biases and uncertainty factors can be updated if necessary using the methodology described in this report as new operating data is collected from subsequent operation.

The models and methods described in this report will be used to perform nuclear design analyses for Duke's fleet of Westinghouse pressurized water reactors which includes Catawba, McGuire, Harris and Robinson nuclear units. The CASMO-5/SIMULATE-3 methodology presented qualifies the code package for analysis of low enriched uranium fuel with both integral and discrete burnable absorbers.



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### Acronym Definitions

Acronym	Definition
a/o	Atom percent
ARO	All rods out
B&W	Babcock & Wilcox
B4C	Boron Carbide
BOC	Beginning of cycle
BP	Burnable poison
BWR	Boiling water reactor
CFM	Centerline fuel melt
CRAM	Chebyshev rational approximation method
Duke	Duke Energy Carolinas
DRWM	Dynamic rod worth measurement
EFPD	Effective full power day
eV	Electron volt
FΔH	Assembly/pin radial power – Average relative power in each fuel assembly or pin
Fq	Assembly/Pin maximum local power (largest relative power in each fuel assembly or pin)
FP	Full power
Fz	Assembly axial peak ( $Fz = Fq/F\Delta H$ )
Gad	Gadolinia oxide integral burnable absorber – $Gd_2O_3 + UO_2$
GWD	Gigawatt days
HFP	Hot full power
HTP	AREVA's high thermal performance fuel assembly design
HZP	Hot zero power
IFBA	Integral fuel burnable absorber
ITC	Isothermal temperature coefficient
LBP	Lumped burnable poison
LEU	Low-enriched uranium
MeV	Million electron volts

### Acronym Definitions

Acronym	Definition
MTU	Metric tonne uranium
MWt	Megawatts thermal
NRC	Nuclear Regulatory Commission
ONRF	Observed nuclear reliability factor
pcm	Percent mille
ppm	Parts per million
ppmb	Parts per million boron
PWR	Pressurized water reactor
QPANDA	Quartic polynomial analytic nonlinear diffusion accelerated
RCCA	Rod cluster control assembly
RFA	Westinghouse's robust fuel assembly design
RVLIS	Reactor vessel level indication system
SCUF	Statistically combined uncertainty factor
SWD	Steps withdrawn
w/o	Weight percent
WABA	Wet annular burnable absorber

## 1.0 Introduction

The design of a commercial pressurized water reactor core determines the characteristics of a specific number of fuel assemblies which are generally similar in design but differ in the amount of fissile material content. The refueling of a reactor core involves removing some of the fuel assemblies and replacing them with fresh fuel and possibly previously burned fuel assemblies. In a reload core the fuel enrichment, burnup, and burnable absorber content may be different for each fuel assembly in the core. In general, the neutronic and operating parameters of the new core are different from the previous core. The reload design analysis defines the characteristics of the new core and confirms that it can be operated safely while meeting design power generation requirements.

Neutronic analyses are performed to define the number of feed assemblies, their enrichment, burnable poison loading, and the arrangement of fuel and control components within the reactor. Calculations are performed which verify core safety parameters, determine reactor protection system setpoints, and provide necessary startup and operational information. This report presents a state-of-the-art package of analytical models which may be used to develop these analyses. The fidelity of the CASMO-5/SIMULATE-3 analytical models is demonstrated by comparison of calculated nuclear parameters to available measurements from power reactor operation and laboratory experiments, and through code-to-code comparisons.

Duke Energy currently performs reload design analysis for the McGuire and Catawba nuclear stations with methodologies defined by References 1 through 8. Reference 1 describes the overall reload design methodology. Reference 2 describes the current core physics methodology which uses CASMO-4/ SIMULATE-3 analytical models. Reference 3 describes Duke's current Nuclear Regulatory Commission (NRC)-approved methodology for performing dynamic rod worth measurements and Reference 8 describes Duke's Rod Swap methodology. References 4 through 7 address other specific aspects of the reload design process required to support operational and safety analyses for the McGuire and Catawba nuclear units. The intent is to extend the reload design capability described in the aforementioned methodology reports to the Harris and Robinson nuclear units. This report extends Duke's CASMO based SIMULATE-3 core physics methodology to the Harris and Robinson reactors, and also provides the basis for the future upgrade of the currently licensed CASMO-4/SIMULATE-3 methodology used at McGuire and Catawba.

As part of a continuous effort to improve design methods, qualification of the CASMO-5/SIMULATE-3 code system for performing reload design calculations is presented in this report. This report demonstrates the acceptability of the CASMO-5/SIMULATE-3 code system for performing reload calculations for Westinghouse three-loop and four-loop reactor cores characteristic of the Harris/Robinson and Catawba/McGuire nuclear steam supply systems. Reactor fuel encompassing both 15x15 and 17x17 lattice geometries are analyzed. In addition, reactor cores containing fuel with B4C discrete burnable absorbers, zirconium diboride (ZrB<sub>2</sub>) integral fuel rod burnable absorbers (IFBA), and gadolinia integral fuel burnable absorbers are analyzed. This diverse set of benchmarks is used to demonstrate the accuracy of the CASMO-5/SIMULATE-3 models for use in calculating reload physics parameters, core reactivity and power distributions for use in reload design analyses at all four Duke Westinghouse reactor sites.

Qualification of the CASMO-5/SIMULATE-3 code package is accomplished through performance of a series of benchmark calculations using measured data from four contemporary McGuire Unit 1 fuel cycles, five Harris fuel cycles and five Robinson fuel cycles. The benchmark cycles selected encompass both 15x15 and 17x17 fuel lattice designs, varying cycle lengths, fuel enrichments, burnable absorber types and fuel management strategies. Coupled with differences in reactor core size, rated thermal power level and operating temperature associated with each of the benchmark units, the cycles analyzed provide a diverse set of fuel design and core operating conditions for code validation. The McGuire benchmark calculations are for a 4-loop 193 fuel assembly reactor core with core designs developed using Westinghouse's 17x17 Robust Fuel Assembly (RFA) design. The core designs evaluated employ a variety of IFBA patterns ranging from 16 to 128 IFBA rods per fuel assembly in combination with discrete burnable absorbers for reactivity and peaking control. The McGuire benchmarks are characteristic of the fuel management strategy and fuel type used at all of Duke's four loop reactors (McGuire Units 1 and 2, and Catawba Units 1 and 2), and are therefore representative of this reactor type. The Harris and Robinson benchmarks are for a 3-loop 157 fuel assembly core with core designs developed with AREVA's high thermal performance (HTP) fuel design with gadolinia integral burnable absorbers. Differences between the Harris and Robinson core designs primarily relate to differences in the fuel lattice design, rated thermal power level and operating temperature. A large range of gadolinia concentrations and absorber patterns are contained in the Harris and Robinson benchmark cycles. Additional information describing the plant, fuel design and fuel cycles evaluated are presented in Section 3 of this report.

Qualification of the CASMO-5 code for predicting pin power distributions in LEU fuel lattices with and without gadolinia is based on comparison of predicted pin power distributions against measured data from the B&W Urania Gadolinia critical experiments. Additional qualification of the CASMO-5 code is performed by the code vendor, Studsvik Scandpower, in Reference 11. These qualifications consist of comparisons against measurements and higher order calculations.

Section 2 presents a description of the computer codes that are used in this reload design methodology.

Section 3 presents the operational benchmarks against measured reactivity and power distribution data from McGuire, Harris and Robinson that demonstrates the ability of the methodology to predict core physics parameters and power distributions. Comparisons between predicted and measured critical boron concentrations, control bank worths, isothermal temperature coefficients and assembly power distributions are presented. A statistical analysis of the measured to predicted power distributions was also performed to determine  $F_{\Delta H}$ ,  $F_q$  and  $F_z$  assembly uncertainty factors.

Section 4 presents benchmarks of the CASMO-5/SIMULATE-3 nuclear design methodology against the B&W Urania Gadolinia critical experiment data and demonstrates the ability of the methodology to predict relative fuel pin power in all-LEU lattices as well as lattices containing gadolinia fuel pins. A statistical analysis of predicted and measured pin power distributions as well as CASMO-5 and SIMULATE-3 power distribution comparisons for gadolinia fuel pins were performed to develop LEU and gadolinia pin power uncertainties.

Section 5 describes the development of statistically combined power distribution uncertainty factors for use in reload design power distribution analyses.

Section 6 summarizes the results and conclusion of this report.



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## 2.0 Computer Codes and Models

As part of the reload design process, reactor physics calculations are performed on a cycle-specific basis to develop the core nuclear design and ensure safety. The cycle design is set by specifying the number and enrichment(s) of the feed assemblies and burnable absorbers, as well as the core locations of all feed and reinserted assemblies. Collectively, this information defines the core loading pattern. Calculations are performed to verify core safety parameters and to generate operational and reactor protection system (RPS) limits. Calculations are also performed to support startup testing, including rod worth measurement, and for core follow activities during reactor operation. Details of these calculations have previously been described for McGuire and Catawba in References 1, 3, 4, 5, and 8. Duke intends to extend the methodology described in these references to the Harris and Robinson reactors. The extension of the McGuire/Catawba reload analysis methodology to Harris and Robinson, or any modification to this methodology, will be the subject of this and future NRC submittals.

This section provides a brief description of the CASMO-5/SIMULATE-3 computer codes and the supporting programs that are used to perform the above calculations. The NRC has approved Duke's current reactor physics calculation methodology, which includes the use of CASMO-4/SIMULATE-3 MOX computer codes. The methodology described in this report is fundamentally the same as Duke's current methodology with replacement of the CASMO-4 code with the CASMO-5 code. The methodology is comprised of the following four computer codes:

- CASMO-5
- CMS-LINK
- SIMULATE-3
- SIMULATE-3K

These computer codes were developed by Studsvik Scandpower Incorporated. Various forms of these codes have a long history of utilization in both the United States and international nuclear community. In the calculation sequence, CASMO-5 is used to generate nuclear data for each unique fuel assembly lattice. CMS-LINK collects this data into a single library for use by SIMULATE-3 or SIMULATE-3K. SIMULATE-3 is used to deplete the fuel cycle and to predict critical boron concentrations, rod worths, reactivity coefficients and core power distribution. It is also used for 3-D analysis for generation of operational and reactor protection system limits, and

for confirmation of accident analysis key input parameters and thermal limits. SIMULATE-3K is used to model core transients and support the dynamic rod worth measurement technique.

## 2.1 CASMO-5

CASMO-5 is a multi-group two-dimensional characteristics based transport theory code for burnup calculations on Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) assemblies or simple pin cells. A description of the code's methodology and input is provided in References 9 and 10, respectively. The code accommodates a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for absorber-loaded fuel rods, integral fuel burnable absorber (IFBA), burnable absorber rods, cluster control rods, in-core instrument channels, and water gaps. Reflector and baffle calculations can also be performed with CASMO-5. Multi-group calculations can be performed using as many as 586 neutron and 18 gamma group data from ENDF/B-VII.1.

CASMO-5 incorporates the direct microscopic depletion of burnable absorbers into the main calculation and a fully heterogeneous model is used for the two-dimensional transport calculation. The accurate depletion of burnable absorbers such as gadolinia is modeled by subdividing the fuel region into small annular rings. CASMO-5 also tracks the gadolinium nuclides separately from heavy nuclides and fission products. This allows for a gadolinium specific depletion model to be used to accurately account for the depletion of gadolinium without the need for extremely small depletion steps. CASMO-5 calculations are typically performed in single fuel assembly geometry, however, the capability exists to perform larger multi-assembly calculations. Multi-assembly calculations use the same transport theory methodology solution (Method of Characteristics) as the single assembly calculations.

Some characteristics of CASMO-5 are:

- The two-dimensional transport solution is based upon the Method of Characteristics (MoC) with a linear source approximation.

- Multigroup energy discretization can be carried out in a number of different energy group structures, with the finest structure using 586 groups and the default for the MoC calculation using 19 groups for UO<sub>2</sub> fuel and 35 groups for MOX.
- Nuclear data for CASMO-5 are collected in a library containing microscopic cross sections in 586 energy groups. Neutron energies cover the range from 10<sup>-5</sup> eV to 20 MeV.
- CASMO-5 can accommodate non-symmetric fuel assemblies. However, because most fuel designs are symmetric, single fuel assembly calculations are typically performed with octant symmetry.
- Effective resonance cross sections are calculated individually for each fuel pin.
- Microscopic depletion is calculated in each fuel and burnable absorber pin.
- The neutron transport and burnup calculations are coupled via a predictor-corrector approach which greatly improves accuracy. This is particularly important when burnable poison rods are involved. A special quadratic depletion model is used for lattices containing gadolinium. The burnup equations are solved with the Chebyshev Rational Approximation Method (CRAM).
- Reflector calculations are performed and discontinuity factors are calculated at the assembly boundaries and for reflector regions.
- CASMO-5 generates output edits of few-group cross sections and reaction rates for any region of the assembly. An ASCII card image file (CI-file) containing few group nuclear data and discontinuity factors is created for linking to various diffusion theory core analysis programs such as CMS-LINK/SIMULATE-3.

CASMO-5 cases are executed for each unique fuel assembly lattice configuration. A typical case set characterizes the effect of fuel burnup, moderator temperature, fuel temperature, soluble boron concentration, shutdown cooling time, and control rod presence. For reflector regions the impact of changes in moderator temperature and soluble boron concentration are typically modeled.

## 2.2 CMS - LINK

CMS-LINK processes data generated by CASMO-5 and produces a nuclear data library for input to the SIMULATE-3 core model as described in Reference 12. The code collects the following data for each unique fuel lattice configuration.

- Macroscopic cross sections in two energy groups
- Discontinuity factors at fuel assembly boundaries in two energy groups
- Yields and microscopic cross sections for important fission products
- Incore detector constants
- Kinetics data
- Pin by pin power distributions

Data is collected into multi-dimensional tables that characterize the effect of both instantaneous and integrated perturbations to local core conditions. The precise functionalization of the data varies depending on the type of data and the amount that a given data type changes as core conditions change.

## 2.3 SIMULATE-3

SIMULATE-3 is a three-dimensional nodal diffusion theory reactor core simulator described in Reference 13. The program calculates core wide power distribution and fuel depletion with macroscopic cross sections in two energy groups. Homogenized cross sections and discontinuity factors are used on a coarse mesh nodal basis to solve the two group diffusion equation using the Quartic Polynomial Analytic Nonlinear Diffusion Accelerated (QPANDA) neutronics model. Polynomial expansions are used to represent the fast and thermal intra-nodal flux solution. A thermal hydraulics model is used to calculate coolant density and fuel temperature distributions to provide both fuel and moderator temperature feedback. Calculations can be performed in eighth-core, quarter-core, half-core and full-core geometries.

The nodal solution is performed on a geometric mesh of either one or four nodes per assembly in the radial plane and an appropriate axial mesh in the active fuel column. A transverse integration procedure is used to reduce the multi-dimensional diffusion equations to a set of coupled one-

dimensional equations. The one-dimensional equations are solved by representing the flux with fourth-order polynomial expansions in both the fast and thermal groups, and then using weighted residual methods to determine the coefficients for each of the two energy groups. Explicit models of top, bottom, and radial reflector regions allow analytic solutions for flux and leakage at the core boundary. A microscopic depletion model is used to track iodine, xenon, promethium, and samarium during anticipated core transients. The macroscopic effects of fission product isotopic decay after shutdown can also be included on nodal basis. Pin power distributions are constructed by synthesizing results of the nodal mesh solution with heterogeneous lattice solutions extracted from CASMO-5.

The fuel temperature data used in SIMULATE-3 is developed from data derived from the fuel rod thermal model within SIMULATE-3K. As additional industry data becomes available, fuel temperature inputs may be updated to improve model performance.

SIMULATE-3 is used to calculate cycle-specific core reactivity, reactivity coefficients and power distribution peaking factors input assumptions used in the safety analysis and to provide data to support the startup and operation of the reactor.

## **2.4 SIMULATE-3K**

The SIMULATE-3K code is a three-dimensional transient neutronic version of the SIMULATE-3 code as described in References 14 and 15. SIMULATE-3K uses the same full two-group nodal spatial model developed in SIMULATE-3, with the addition of six delayed neutron groups. The program solves the transient neutron diffusion equation incorporating effects of delayed neutrons, spontaneous fission in fuel, alpha-neutron interactions from actinide decay, and gamma-neutron interactions from long term fission product decay. The thermal-hydraulics module consists of a fuel pin heat conduction model, fission product decay heat generation model, and a 5-equation hydraulic channel model.

The fuel pin conduction model calculates the radial temperature distribution and the fuel pin surface heat flux using a finite difference model of the nonlinear cylindrical heat conduction equation. An explicit fuel pin conduction calculation is performed for the average fuel pin in each nodal mesh, and optionally for the hot pin in each fuel assembly to determine local maxima.

The axial nodalization of the fuel pin conduction solution is identical to that of the neutronics model. Fuel, gap, and clad thermal properties are treated as functions of node-averaged fuel pin burnup and local temperature. Convective heat transfer coefficients are computed using regime-dependent correlations. The coupling between the pin conduction calculation and the heat transfer coefficient calculation is fully resolved at each time step by nonlinear iteration.

An explicit hydraulic calculation is performed for each nodal mesh, using the average fuel pin heat flux and hydraulic characteristics of the node. The axial nodalization of the hydraulic solution is identical to that of the neutronics model. For pressurized water reactor (PWR) applications, SIMULATE-3K utilizes a fully-implicit, five-equation hydraulics model (liquid mass and energy, vapor mass and energy, and mixture momentum).

The SIMULATE-3K neutronics model uses the same nuclear data library as SIMULATE-3. The thermal and hydraulic models are coupled to the neutronics model via the fuel pin heat generation rate which is directly determined from the calculated neutron power. In turn, the thermal hydraulics module provides the nuclear calculation with the appropriate hydraulic data to permit nuclear feedback with local thermal conditions. Boundary conditions for the hydraulic calculations are defined by moderator core inlet conditions and upper plenum pressure.

SIMULATE-3K is capable of modeling core transients initiated by changes in soluble boron concentration, control rod placement, moderator temperature, moderator flow, and/or system pressure. Incore and excore instrumentation may be modeled for the purpose of driving the reactor control system and allowing realistic comparison to actual core transients. SIMULATE-3K is a best estimate model, however conservatism may be applied via individual scalar multipliers to important parameters such as fuel conductivity, specific heat, gap conductance, convective heat transfer, fuel temperature, moderator temperature, void fraction, delayed neutron yields, and control rod worths.

### 3.0 Power Reactor Benchmark Analyses

This section compares measured core physics parameters and power distributions from McGuire Unit 1, Harris Unit 1 and Robinson Unit 2 fuel cycles to SIMULATE-3 predictions.

Comparisons are made for the following recent operating cycles:

McGuire Unit 1 Cycles 20, 21, 22 and 23

Harris Unit 1 Cycles 14, 15, 16, 17 and 18

Robinson Unit 2 Cycles 24, 25, 26, 27 and 28

The benchmark units and cycles were selected because they encompass a diverse set of fuel designs, lattice design (15x15 and 17x17), burnable absorber types, fuel management strategies and reactor types (4-loop and 3-loop reactor designs). Key characteristics of the benchmark cycles are summarized in Table 3-1. The diverse nature of the benchmark cycles selected allows for a thorough evaluation of the acceptability of the CASMO-5 based SIMULATE-3 models and qualification of their accuracy. Results from the predicted to measured reactivity and power distribution comparisons demonstrate the acceptability of the CASMO-5/SIMULATE-3 code system for performing reactivity and power distribution calculations at Catawba, McGuire, Harris and Robinson nuclear units.

Measurements of critical boron concentrations, control bank worths, and isothermal temperature coefficients are made during the initial startup of each fuel cycle at hot zero power (HZP) conditions. Critical boron concentration and core wide power distribution measurements are also made throughout the depletion of each fuel cycle at or near nominal hot full power (HFP) rated thermal power conditions. A statistical analysis of the power distribution data is also performed to determine the fuel assembly  $F_{\Delta H}$ ,  $F_q$ , and  $F_z$  peaking factor uncertainties in Section 3.3.

For each parameter compared, a brief description of the measurement technique is included, and where applicable, the mean and standard deviations of the differences or relative differences between measurements and predictions are provided.



### 3.1 Description of Reactors

The McGuire and Catawba reactors are operated by Duke Energy Carolinas, a division of Duke Energy. Both reactors sites are located near Charlotte, North Carolina. Each reactor is a Westinghouse designed four loop pressurized water reactor containing 193 fuel assemblies and 53 control rod clusters. Both McGuire units are currently rated at 3469 megawatts thermal (MWt), while each Catawba unit is currently rated at their initial design thermal power level of 3411 MWt. The McGuire Unit 1 benchmark cycles operated at the pre-uprate thermal power level of 3411 MWt. The average moderator temperature at HFP conditions is approximately 585 °F. The general core layout of instrumented and control rod core locations is shown in Figure 3-1. Each fuel assembly is comprised of a 17x17 square lattice having 264 fuel pins, 24 guide tubes, and a central instrument tube. For all fuel cycles analyzed, an 18 month low leakage fuel management strategy was employed.

Harris is a single unit plant operated by Duke Energy Progress, a division of Duke Energy, and located near Raleigh, North Carolina. The reactor is based on Westinghouse's three loop pressurized water reactor design containing 157 fuel assemblies and 52 control rod clusters. Harris is currently rated at 2948 megawatts thermal. It was uprated in Cycle 18 from 2900 MWt. The average moderator temperature at HFP conditions is approximately 589 °F. The general core layout of instrumented and control rod core locations is shown in Figure 3-2. Each fuel assembly is comprised of a 17x17 square lattice having 264 fuel pins, 24 guide tubes, and a central instrument tube. An 18 month low leakage fuel management strategy is employed.

Robinson is a single unit plant operated by Duke Energy Progress, a division of Duke Energy, and located near Hartsville, South Carolina. The reactor, like Harris, is based on Westinghouse's three loop pressurized water reactor design containing 157 fuel assemblies, but differs in that it has only 45 control rod clusters. Robinson's rated thermal power level is 2339 megawatts thermal. The average moderator temperature at HFP is approximately 576 °F. The general core layout of instrumented and control rod core locations is shown in Figure 3-3. Each fuel assembly is comprised of a 15x15 square lattice having 204 fuel pins, 20 guide tubes, and a central instrument tube. An 18 month low leakage fuel management strategy is employed.

### 3.2 Measured to Predicted Reactivity Comparisons

Section 3.2 presents comparisons of measured to predicted critical boron concentrations, isothermal temperature coefficients, and control rod bank worths. Measured data is compiled from measurements performed as part of the physics testing program performed following each refueling outage, and from required reactivity measurements performed during routine power operation.

#### 3.2.1 Critical Boron Concentration Comparisons

Critical boron concentrations are measured during initial cycle start-up physics testing performed at the beginning of each fuel cycle, and throughout cycle operation by an acid-base titration of a reactor coolant sample. Critical boron concentration measurements made during initial startup physics testing correspond to beginning of cycle (BOC), peak samarium, no xenon, zero power conditions, and are corrected to the all rods out (ARO) condition. Measurements performed during power operation are at or near hot full power (HFP) nominal operating conditions since Harris, McGuire and Robinson are operated as base load units. Natural boron, with a B<sup>10</sup> concentration of 19.76 a/o was used during the generation of all predicted boron concentrations and as the basis for all comparisons. Measured boron concentrations were corrected for B<sup>10</sup> depletion effects to ensure both the measured and predicted values were based on the same B<sup>10</sup> abundance.

Table 3-2 compares measured to predicted BOC hot zero power (HWP) ARO critical boron concentrations for the benchmark cycles. The mean measured minus predicted deviation is [ ]<sup>a,c</sup> with a standard deviation of [ ]<sup>a,c</sup>. Figure 3-4 graphically presents deviations between measured and predicted HWP ARO critical boron concentrations as a function of core average burnup.

Table 3-3 compares measured and predicted HFP equilibrium xenon critical boron concentrations for each of the benchmark cycles. The mean measured minus predicted deviation is [ ]<sup>a,c</sup> with a standard deviation of [ ]<sup>a,c</sup>. Figures 3-5 through 3-7 graphically illustrate the deviations between measured and predicted HFP critical boron concentrations as a function of cycle burnup. Figure 3-8 shows the distribution of measured to predicted deviations. The

calculated results with CASMO-5/SIMULATE-3 core models show excellent overall reactivity agreement between predicted and measured soluble boron concentrations. The accuracy of the CASMO-5/SIMULATE-3 core models is similar to the accuracy of previously approved Duke methodologies (References 2 and 22).

### **3.2.2 Isothermal Temperature Coefficient Comparisons**

Isothermal temperature coefficients (ITC) are measured at BOC, HZP, near ARO conditions during startup physics testing for each fuel cycle. The ITC is determined by changing the average moderator temperature and measuring the corresponding change in reactivity with a reactivity computer.

Table 3-4 compares measured to predicted BOC, HZP, isothermal temperature coefficients for each benchmark cycle. The mean measured minus predicted deviation is [ ]<sup>a,c</sup> with a standard deviation of [ ]<sup>a,c</sup>.

Figure 3-9 presents measured minus predicted BOC, HZP, isothermal temperature coefficient deviations as a function of core average burnup. The results show that the accuracy of the CASMO-5/SIMULATE-3 core model is consistent with previously approved Duke methodologies (References 2 and 22).

### **3.2.3 Control Rod Worth Comparisons**

Control bank worth measurements are performed at BOC, HZP, peak samarium, no xenon conditions during startup physics testing following each cycle's refueling outage. Each of the reactor sites employs a different measurement technique. The boron dilution technique is used for all Robinson control rod measurements. The Rod Swap measurement technique is used for all Harris measurements, and the dynamic rod worth measurement (DRWM) technique is used for all McGuire control rod measurements. A description of each measurement technique is provided followed by results from measured to predicted control rod worth comparisons. In the following discussions, the term rod and bank are used synonymously.

All control and shutdown banks are measured at Harris and McGuire where the Rod Swap and DRWM measurement techniques are employed. However, only regulating control banks D and C that may be inserted during power operation are measured during startup physics testing at Robinson.

***Boron Dilution Measurement Technique:***

The boron dilution technique is based on a continuous decrease in reactor coolant system boron concentration which is compensated for by incrementally inserting control rod banks in sequence, as required, in small, discrete steps. The change in reactivity due to each insertion is determined from the reactivity computer before and after each insertion. The worth of each bank is determined by the sum of all differential reactivity changes for that bank.

***Rod Swap Measurement Technique:***

The test is initiated by measuring the control or shutdown bank with the highest predicted worth (referred to as the Reference Bank) using the boron dilution technique as described above. Reference bank differential rod worths determined from the reactivity computer before and after each bank insertion are integrated to define a Reference Bank Worth curve as a function of bank position. For the initial test bank measurement, the test bank is inserted and its worth is offset by withdrawing the Reference Bank. For the remaining test bank measurements, the next test bank is exchanged with the previously inserted test bank, and then the Reference Bank. A critical configuration is established with the test bank fully inserted after each measurement. The worth of individual test banks is inferred from the amount the Reference Bank is withdrawn compensating for any changes in reactor conditions during the test.

***Dynamic Rod Worth Measurement Technique:***

The dynamic rod worth measurement (DRWM) technique is a relatively fast method for measuring the reactivity worth of individual control rod banks. The technique used by Duke is based on the methodology developed by Westinghouse in Reference 16 and demonstrated in Reference 3. Rod worth measurements are performed by inserting and withdrawing an individual rod bank at maximum stepping speed without changing boron concentration. Excore detector

signals are recorded during the rod insertion and then processed by a reactivity computer which solves the inverse point kinetics equation with proper analytical compensation to account for space-time effects that occur during control rod insertion. Static and dynamic spatial corrections used in the methodology are calculated with SIMULATE-3 and SIMULATE-3K, respectively.

### ***Predicted and Measured Results***

Table 3-5 presents comparisons between measured and predicted control rod bank worths for each cycle. The mean of the control rod bank worth percent difference for all measured individual control rod banks is [            ]<sup>a,c</sup> with a standard deviation of [            ]<sup>a,c</sup>, where percent difference is defined as the measured minus predicted bank worth divided by the predicted bank worth. The mean percent difference in total bank worth for all cycles is [            ]<sup>a,c</sup> with a standard deviation of [            ]<sup>a,c</sup>.

Figure 3-10 shows the distribution of individual bank worth errors. Figure 3-11 presents the percent difference between measured and predicted individual control bank worths. The bank worth comparisons show the performance of the CASMO-5/SIMULATE-3 core models are similar in accuracy to previously approved Duke methodologies (References 2 and 22). In addition, the results show the test review criteria of +/- 100 pcm or +/- 15%, whichever is greater for the rod swap, boron dilution and DRWM results, is satisfied.

### **3.3 Fuel Assembly Power Distribution Analysis and Uncertainty Factors**

Core power distributions are measured at regular intervals during operation of each fuel cycle. Measured power distributions are derived from electrical signals produced by moveable incore fission chambers as they pass through the instrument guide tube of individual fuel assemblies during reactor operation.

The incore system at Harris and Robinson uses five fission chambers to take measurements at instrumented core locations distributed among the 157 fuel assemblies in the core. The Harris incore system has 50 instrumented locations while the Robinson incore system has one less as the result of the instrument at core location L-05 being converted to a reactor vessel level instrument. The McGuire and Catawba incore systems are similar to the Harris and Robinson systems, but

they use six fission chambers to take measurements in 58 instrumented core locations distributed throughout the 193 fuel assembly core. Figures 3-1, 3-2, and 3-3 show the instrumented core locations for the McGuire/Catawba, Harris and Robinson reactors, respectively. Measurements (signals) are obtained by moving the fission chamber from the top of the fuel assembly to the bottom at a constant rate. A total of 61 individual signals are recorded at equally spaced intervals as the detector passes through each instrumented fuel assembly. These signals are aligned, adjusted for background and then normalized to account for differences in individual fission chamber performance, and changes in reactor power level that may have occurred during data acquisition. The normalized signals are converted to normalized relative power by applying signal-to-power conversion factors that were derived from cycle-specific core models. These conversion factors are dependent upon core location, burnup, and control rod presence. Signal-to-power conversion factors based on CASMO-5/SIMULATE-3 models were used to process all measured signals.

The culmination of the process described above is a full core, assembly mesh, three-dimensional measured relative power distribution. These data are used to calculate radial, total, and axial peaking factors that characterize the important attributes of the power distribution. The assembly radial power, or assembly  $F\Delta H$ , is simply the average relative power in each fuel assembly. The assembly maximum power, or assembly  $F_q$ , is the largest relative power in each fuel assembly. The assembly axial peak power, or assembly  $F_z$ , is the assembly  $F_q$  normalized to the assembly average power for each fuel assembly ( $F_z = F_q/F\Delta H$ ). Measured assembly  $F\Delta H$ ,  $F_q$ , and  $F_z$  peaking factors are compared directly against equivalent peaking factors generated by SIMULATE-3 to determine assembly uncertainty factors.

SIMULATE-3 is used to model the reactor conditions for 264 power distribution measurements performed during operation of recent Harris, Robinson and McGuire fuel cycles. These measurements are distributed as follows among the three units:

- 98 power distribution measurements that were performed during the operation of Harris Unit 1 Cycles 14 through 18,
- 94 power distribution measurements that were performed during the operation of Robinson Unit 2 Cycles 24 through 28, and
- 72 power distribution measurements that were performed during operation of McGuire Unit 1 Cycles 20 through 23.

Comparisons of predicted and measured assembly  $F\Delta H$ , maximum assembly  $F_q$  and assembly axial peak  $F_z$  peaking factors are performed to define the relative error in the predicted value for each core location for each power distribution measurement. The relative error in the predicted value is defined by equation 3-1.

$$Relative\ error = \frac{(P - M)}{M} * 100 \quad \text{eq. 3-1}$$

where,

$P$  = SIMULATE-3 predicted power

$M$  = Measured power

Comparisons of calculated and measured assembly average powers for Harris, Robinson and McGuire for the last cycle evaluated for each reactor site are presented in Figures 3-12 through 3-14, respectively. These results are representative of the typical performance of the SIMULATE-3 core models predictive capability relative to measurement. Relative errors in predicted values produced from comparisons of predicted and measured power distributions are used to develop one-sided upper tolerance limit uncertainties to ensure with a 95% confidence level that 95% of local power predictions, when augmented by the uncertainty factor, are equal to or larger than the measured value. This statistical method requires that the data set pass a test for normality which is performed at a 1% level of significance. If a given data set fails this normality test, uncertainty factors are determined by a non-parametric evaluation of the data. These statistical methods are described in References 17 through 20.

Assembly uncertainty factors or Observed Nuclear Reliability Factors (ONRFs) for  $F\Delta H$ ,  $F_q$  and  $F_z$  are derived from comparisons of measured assembly powers from Harris, Robinson, and McGuire Unit 1 power distribution measurements against equivalent peaking factors generated with SIMULATE-3. Core locations with normalized assembly average ( $F\Delta H$ ) powers less than 1.0 are not considered in the ONRF calculation. Equation 3-2 is used to define each uncertainty factor.

$$ONRF = 1 - Bias + K_a \sigma_a \quad \text{eq. 3-2}$$

The bias term in the above equation is defined as the mean relative error in the predicted peaking factor for all power distribution measurements in the data set being evaluated. The term is defined in equation 3-3:

$$Bias = \frac{1}{n} \sum_i^n \frac{(P_i - M_i)}{M_i} \quad \text{eq. 3-3}$$

where,

$P_i$  = the  $i^{\text{th}}$  predicted or calculated value

$M_i$  = the  $i^{\text{th}}$  measured value

$n$  = data set size

The  $K_a \sigma_a$  term is equivalent to the 95/95 one-sided upper tolerance uncertainty relative to the bias. The subscript “a” denotes that these factors are assembly parameters. Definitions for  $\sigma_a$  and  $K_a$  are:

$\sigma_a$  = the standard deviation of the relative error between predicted and measured powers

$K_a$  = 95/95 one-sided upper tolerance factor based on sample size n (Ref. 19)

The D-prime test for normality is used to test the distribution for normality (Ref. 20). If a given data set fails this normality test (the data set is not normal), then the  $K_a \sigma_a$  term is determined from a non-parametric analysis. In this instance, the total uncertainty factor, or ONRF, is defined by equation 3-4.

$$ONRF = 1 - E_{mth} \quad (\text{non-parametric}) \quad \text{eq. 3-4}$$

The  $E_{mth}$  term is the  $m^{\text{th}}$  smallest relative error (negative errors indicate that the measured power is greater than the predicted power) for a sample size of n such that there is a 95% confidence level that at least 95% of the population has a relative error greater than this value. The  $K_a \sigma_a$  term in this instance is defined by equation 3-5.

$$K_a \sigma_a = Bias - E_{mth} \quad \text{eq. 3-5}$$



where,

$Bias$  = the mean relative error defined by eq. 3-3

$E_{mth}$  = the relative error corresponding to the  $m^{th}$  smallest relative error that defines a 95/95 one-sided tolerance uncertainty based on non-parametric statistics.

D-prime test results and Observed Nuclear Reliability Factors or assembly uncertainties for  $F\Delta H$ ,  $F_q$  and  $F_z$  are summarized in Table 3-6. These assembly uncertainties are combined with pin uncertainties (developed in section 4) to calculate the  $F\Delta H$ ,  $F_q$  and  $F_z$  peaking factor uncertainties applied in reload design analyses.

### 3.4 Summary Comparison of Benchmark Results

A summary of the number of observations, the average deviation between measured and predicted values and the associated standard deviation for each of the four reactor physics parameters evaluated (HZP critical boron concentration, HFP boron concentration, control rod worth, and isothermal temperature coefficient) are shown in Table 3-7. These results are based on the benchmark calculations performed against measured data from Harris, Robinson and McGuire. The results demonstrate that the performance of the CASMO-5/SIMULATE-3 core models are acceptable for modeling reactor cores containing 15x15 and 17x17 fuel with IFBA, gadolinia and discrete burnable absorbers.

Assembly uncertainty factors (or ONRFs) for  $F\Delta H$ ,  $F_q$ , and  $F_z$  that were developed from comparisons of the McGuire, Harris and Robinson measured power distribution data and CASMO-5/SIMULATE-3 predicted values are summarized in Table 3-6. Excellent agreement between predicted and measured power distributions was observed. As new operating data are collected from subsequent fuel cycle operation these values may be updated if necessary using the methodology described in this report.

In summary, the CASMO-5/SIMULATE-3 benchmark results presented demonstrate the ability of the CASMO-5/SIMULATE-3 code package to adequately model the behavior of reactor cores

containing low enriched uranium fuel. Accordingly, the results and conclusions drawn from the benchmarks are considered applicable to future McGuire, Catawba, Harris and Robinson core designs using integral (IFBA and gadolinium) and discrete burnable absorbers for reactivity and power peaking control.

**Table 3-1  
Core Characteristics**

<b>Cycle</b>	<b>Cycle Length (EFPD)</b>	<b>Power Level (MWt)</b>	<b># Feed Assemblies</b>	<b>Feed Enrichments (w/o)</b>	<b># Gad/IFBA Rods in Fresh</b>	<b>Gadolinia Concentrations (w/o Gd)</b>	<b># Removable BP Rods</b>
R2C24	522.3	2339	56	4.25, 4.45, 4.60	960 Gad	2, 8	None
R2C25	495.3	2339	57	4.10, 4.40, 4.90	860 Gad	2, 4, 8	None
R2C26	483.6	2339	56	4.10, 4.60	896 Gad	2, 8	None
R2C27	488.6	2339	56	4.10, 4.45	864 Gad	2, 4, 8	None
R2C28	530.5	2339	76	4.60, 4.95, 2.00	1200 Gad	2, 4, 8	None
H1C14	494.1	2900	68	4.25, 4.75	1312 Gad	4, 6, 8	None
H1C15	528.1	2900	69	4.45, 4.95	1336 Gad	4, 6, 8	None
H1C16	501.8	2900	68	4.40, 4.95	1200 Gad	4, 6, 8	None
H1C17	518.9	2900	69	4.45, 4.85	1236 Gad	2, 4, 6, 8	None
H1C18	489.3	2948	72	4.55, 4.95	1312 Gad	4, 6, 8	None
M1C20	482.8	3411	68	4.00, 4.28, 4.85	6816 IFBA	None (IFBA)	192
M1C21	502.8	3411	73	3.60, 4.60, 4.95	7240 IFBA	None (IFBA)	528
M1C22	509.9	3411	72	4.21, 4.90	7616 IFBA	None (IFBA)	64
M1C23	503.7	3411	73	4.15, 4.40, 4.95	7520 IFBA	None (IFBA)	32

**Table 3-2**  
**BOC, HZP, ARO Critical Soluble Boron Concentration Comparisons**

Cycle	Measured (ppm)	Predicted (ppm)	Difference (M – P)

a-c

Table 3-3  
HFP Critical Soluble Boron Concentration Comparisons

Harris Unit 1 Cycle 14				Harris Unit 1 Cycle 15				Harris Unit 1 Cycle 16			
	Meas.	Pred.	Diff		Meas.	Pred.	Diff		Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)

a-c

Table 3-3 cont'd  
HFP Critical Soluble Boron Concentration Comparisons

Harris Unit 1 Cycle 17			
	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

Harris Unit 1 Cycle 18			
	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

a-c

Table 3-3 cont'd  
HFP Critical Soluble Boron Concentration Comparisons

Robinson Unit 2 Cycle 24				Robinson Unit 2 Cycle 25				Robinson Unit 2 Cycle 26			
	Meas.	Pred.	Diff		Meas.	Pred.	Diff		Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)

a-c

Table 3-3 cont'd  
HFP Critical Soluble Boron Concentration Comparisons

Robinson Unit 2 Cycle 27				Robinson Unit 2 Cycle 28			
	Meas.	Pred.	Diff		Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)

a-c



Table 3-3 cont'd  
HFP Critical Soluble Boron Concentration Comparisons

McGuire Unit 1 Cycle 20				McGuire Unit 1 Cycle 21			
	Meas.	Pred.	Diff		Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)

a-c

Table 3-3 cont'd  
HFP Critical Soluble Boron Concentration Comparisons

McGuire Unit 1 Cycle 22				McGuire Unit 1 Cycle 23			
	Meas.	Pred.	Diff		Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)	EFPD	(ppm)	(ppm)	(M – P)

a-c

**Table 3-4**  
**BOC, HZP Isothermal Temperature Coefficient Comparisons**

Cycle	Measured (pcm/°F)	Predicted (pcm/°F)	Difference (M – P)

a-c

**Table 3-5**  
**BOC, HZP Individual Control Bank Worth Comparisons**

Cycle		Control Rod Bank									Total Worth
		CBD	CBC	CBB	CBA	SBE	SBD	SBC	SBB	SBA	

a-c

**Table 3-6**  
**D-Prime Test Results and Assembly Uncertainty Factors**

**D-Prime Test For Normality Results**

Parameter	FΔH	Fq	Fz
N	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
D' (P = 0.005)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
D'	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
D' (P = 0.995)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
Evaluation	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

**Assembly Uncertainty**

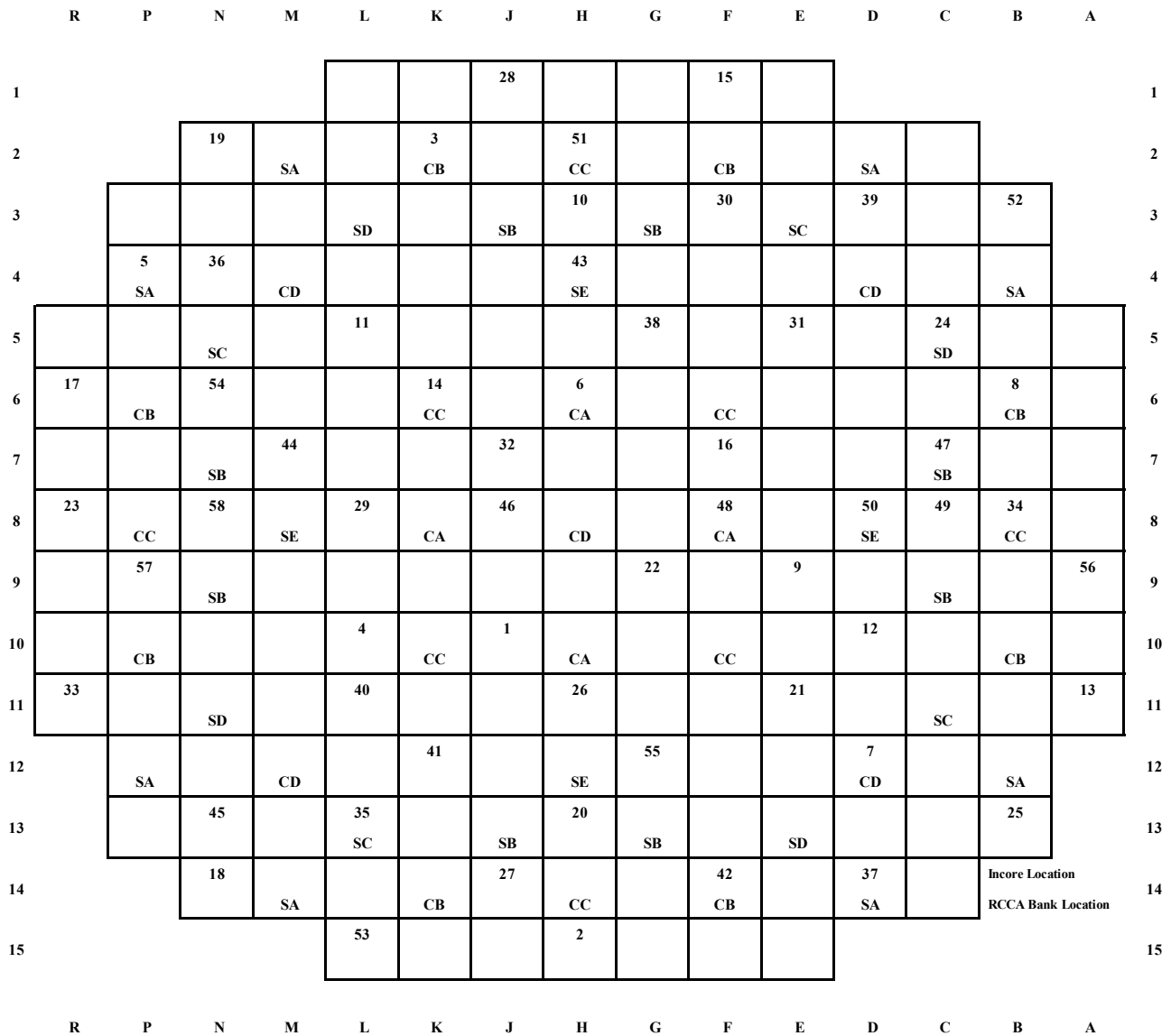
Parameter	Bias	Statistical Deviation ( $K_a \sigma_a$ )	Assembly Uncertainty Factor (ONRF)
FΔH	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
Fq	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
Fz	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

Non Parametric Assembly Uncertainty = -  $E_{mth}$   
( $E_{mth}$  is the  $m^{th}$  smallest relative error for a sample size of N)

**Table 3-7**  
**Summary Comparison of Benchmark Results**

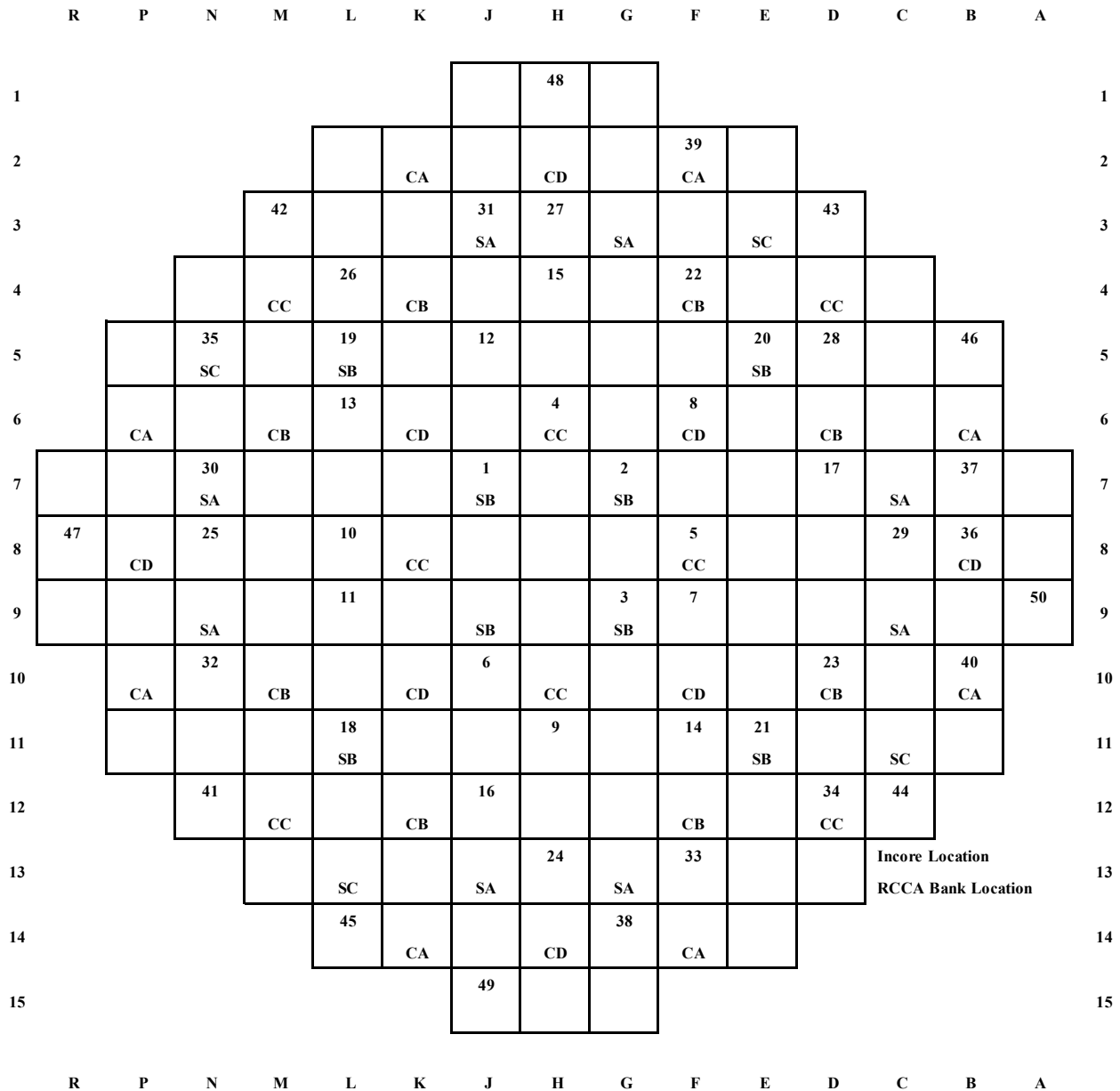
Parameter	N	Mean	Standard Deviation
BOC HZP ARO Critical Soluble Boron (ppmb)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
HFP Critical Soluble Boron (ppmb)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
BOC HZP Isothermal Temp. Coefficient (pcm/°F)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
BOC HZP Individual Control Rod Worth (%)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

**Figure 3-1**  
**McGuire Incore Instrumented and RCCA Bank Core Locations**



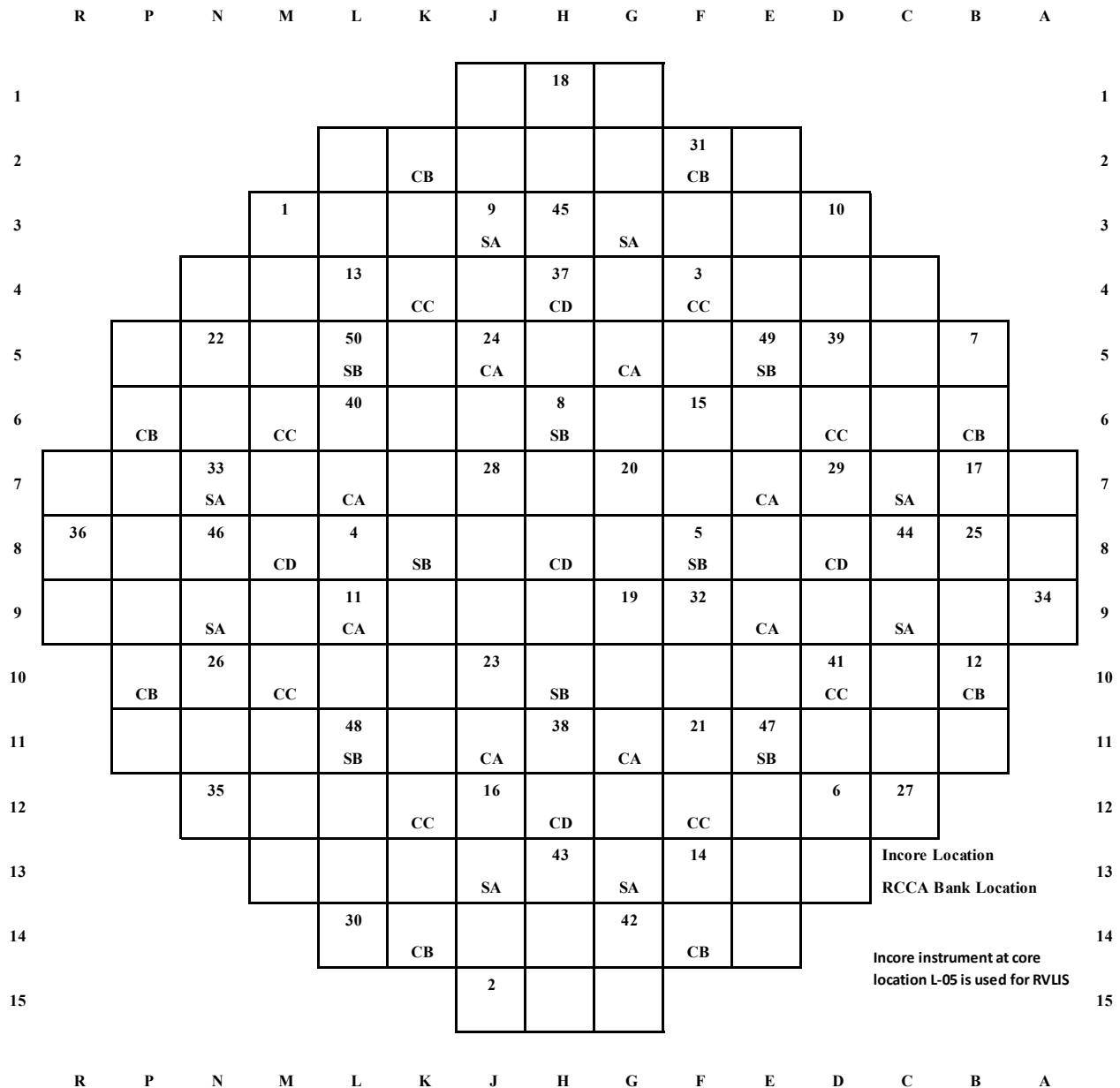
Bank	Number Of Banks
Control D	5
Control C	8
Control B	8
Control A	4
Shutdown A	8
Shutdown B	8
Shutdown C	4
Shutdown D	4
Shutdown E	4
Total	53

**Figure 3-2**  
**Harris Incore Instrumented and RCCA Bank Core Locations**



Bank	Number Of Banks
Control D	8
Control C	8
Control B	8
Control A	8
Shutdown C	4
Shutdown B	8
Shutdown A	8
Total	52

**Figure 3-3**  
**Robinson Incore Instrumented and RCCA Bank Core Locations**



Bank	No. Of Banks
Control D	5
Control C	8
Control B	8
Control A	8
Shutdown A	8
Shutdown B	8
Total	45



**Figure 3-4**  
**Measured Minus Predicted**  
**BOC, HZP, ARO Critical Boron Concentration Deviations**

a,c

**Figure 3-5**  
**Harris Measured Minus Predicted**  
**HFP Critical Boron Concentration Deviations**

a,c

**Figure 3-6**  
**Robinson Measured Minus Predicted**  
**HFP Critical Boron Concentration Deviations**



**Figure 3-7**  
**McGuire Measured Minus Predicted**  
**HFP Critical Boron Concentration Deviations**



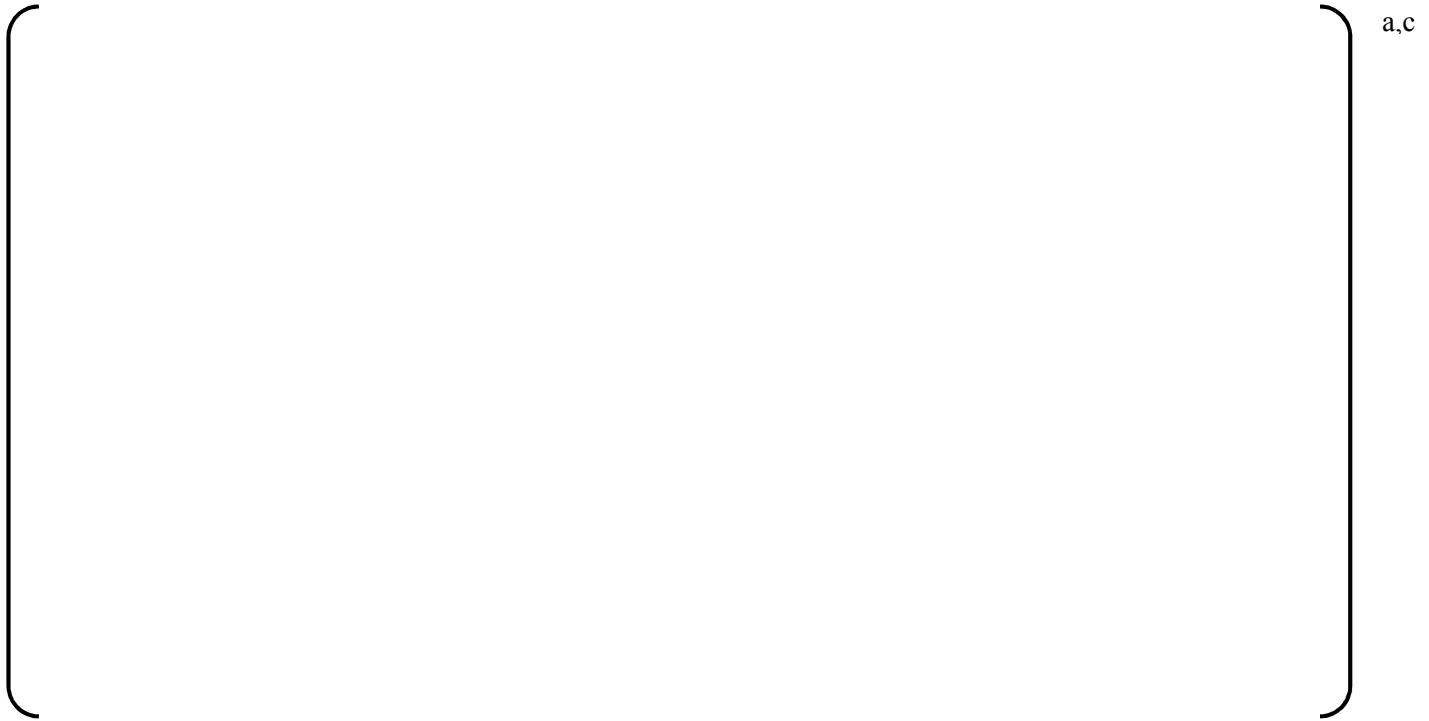
**Figure 3-8**  
**Distribution of HFP Critical**  
**Boron Concentration Deviations**

a,c

**Figure 3-9**  
**Measured Minus Predicted Deviation**  
**In Isothermal Temperature Coefficients**

a,c

**Figure 3-10**  
**Distribution of Individual Bank Worth Deviations**



**Figure 3-11**  
**Percent Difference in Individual Bank Worths**



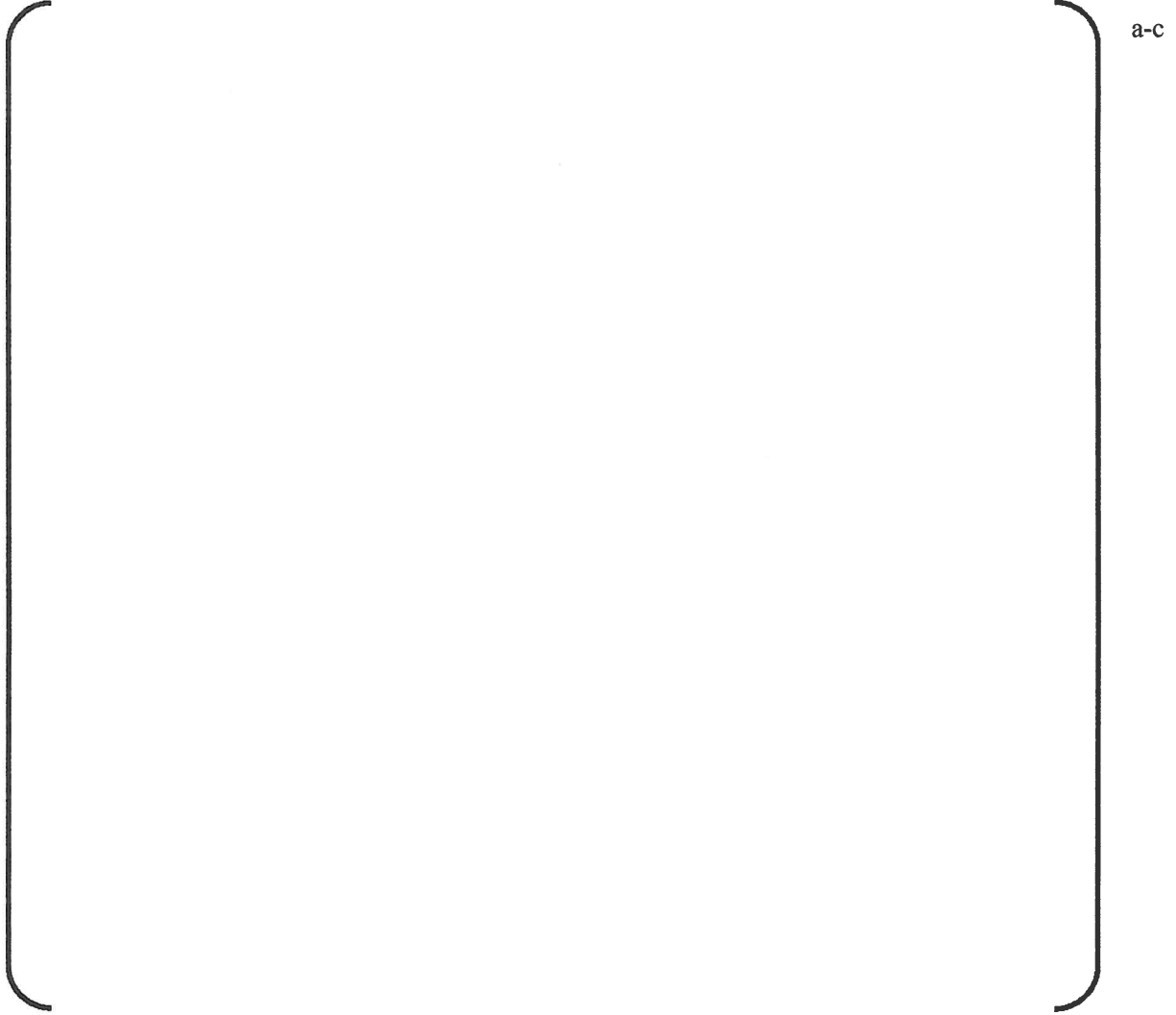
**Figure 3-12**  
**Harris Unit 1 Cycle 18**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 4 EFPD, Control D at 218 SWD**

a-c

**Figure 3-12 cont'd**  
**Harris Unit 1 Cycle 18**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 113 EFPD, Control D at 218 SWD**

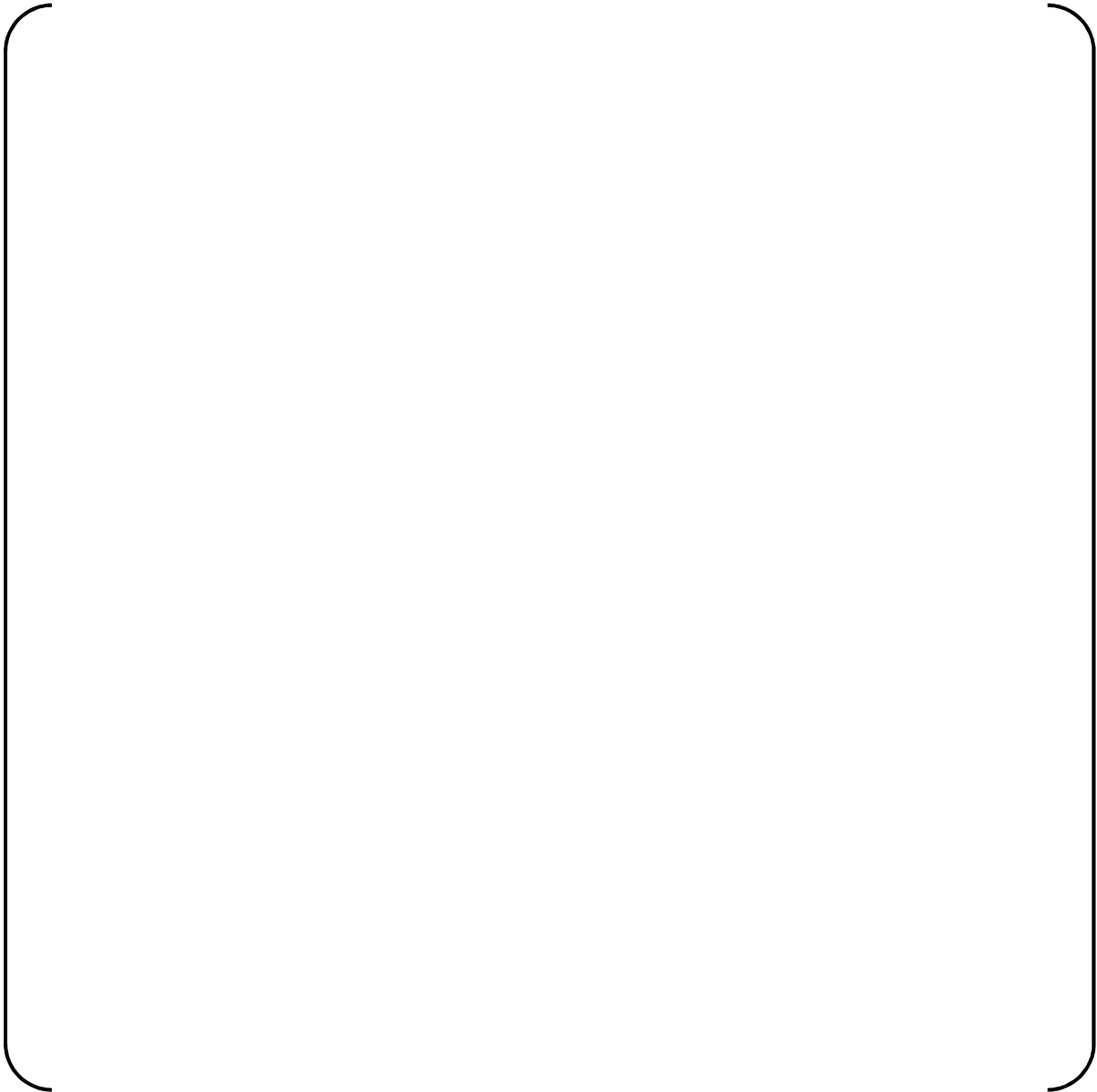


**Figure 3-12 cont'd**  
**Harris Unit 1 Cycle 18**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 192 EFPD, Control D at 218 SWD**



a-c

**Figure 3-12 cont'd**  
**Harris Unit 1 Cycle 18**  
**Assembly Average Power Distribution Comparisons**  
**99.4% FP, 309 EFPD, Control D at 215 SWD**



a-c



**Figure 3-12 cont'd**  
**Harris Unit 1 Cycle 18**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 394 EFPD, Control D at 218 SWD**

a-c

**Figure 3-12 cont'd**  
**Harris Unit 1 Cycle 18**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 485 EFPD, Control D at 218 SWD**



**Figure 3-13**  
**Robinson Unit 2 Cycle 28**  
**Assembly Average Power Distribution Comparisons**  
**99.7% FP, 7 EFPD, Control D at 218 SWD**



**Figure 3-13 cont'd**  
**Robinson Unit 2 Cycle 28**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 103 EFPD, Control D at 218 SWD**



a-c

**Figure 3-13 cont'd**  
**Robinson Unit 2 Cycle 28**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 186 EFPD, Control D at 218 SWD**



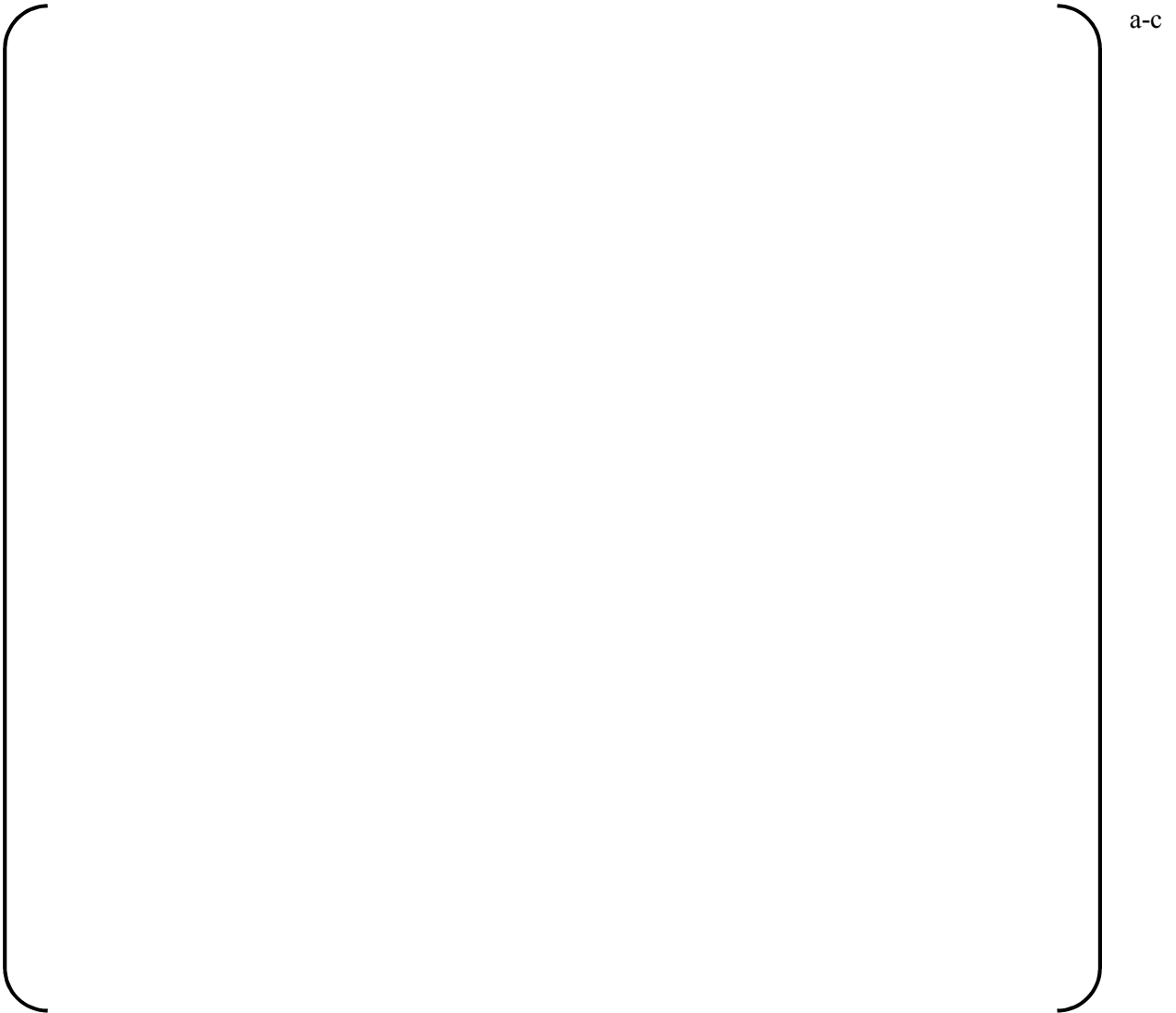
a-c

**Figure 3-13 cont'd**  
**Robinson Unit 2 Cycle 28**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 298 EFPD, Control D at 218 SWD**



a-c

**Figure 3-13 cont'd**  
**Robinson Unit 2 Cycle 28**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 407 EFPD, Control D at 218 SWD**



**Figure 3-13 cont'd**  
**Robinson Unit 2 Cycle 28**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 520 EFPD, Control D at 218 SWD**



a-c



**Figure 3-14**  
**McGuire Unit 1 Cycle 23**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 4 EFPD, Control D at 215 SWD**



**Figure 3-14 cont'd**  
**McGuire Unit 1 Cycle 23**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 90 EFPD, Control D at 215 SWD**



**Figure 3-14 cont'd**  
**McGuire Unit 1 Cycle 23**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 202 EFPD, Control D at 215 SWD**



a-c

**Figure 3-14 cont'd**  
**McGuire Unit 1 Cycle 23**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 311 EFPD, Control D at 215 SWD**



a-c

**Figure 3-14 cont'd**  
**McGuire Unit 1 Cycle 23**  
**Assembly Average Power Distribution Comparisons**  
**100.0% FP, 402 EFPD, Control D at 215 SWD**



a-c

**Figure 3-14 cont'd**  
**McGuire Unit 1 Cycle 23**  
**Assembly Average Power Distribution Comparisons**  
**99.9% FP, 493 EFPD, Control D at 215 SWD**



a-c

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## 4.0 Pin Power Uncertainty Factor Analysis

This section describes the development of the pin power distribution uncertainty factors for low enriched uranium fuel pins and gadolinia fuel pins using CASMO-5/SIMULATE-3. A statistical analysis of predicted-to-measured pin power distribution comparisons against the Babcock and Wilcox (B&W) Urania Gadolinia critical experiments and code-to-code multi-assembly predicted pin power distribution comparisons is used to develop LEU and gadolinia fuel pin power uncertainty factors.

### 4.1 Methodology

Pin power uncertainty factors for low enriched uranium fuel pins and gadolinia fuel pins are developed based on comparisons of predicted-to-measured pin power distributions against measured data from the B&W Urania Gadolinia critical experiments performed in collaboration with the Department of Energy (Ref. 21). Pin power comparisons are performed against fuel lattices containing LEU fuel pins and gadolinia fuel pins to demonstrate the ability of the methodology to predict relative pin powers for LEU and gadolinia fuel pins. A pin power uncertainty factor for low enriched fuel (LEU) pins is developed based on direct comparison of SIMULATE-3 predicted pin power distributions against measured pin power distributions from the B&W Urania Gadolinia critical experiments. This uncertainty is calculated using a 95/95 statistical method consistent with that described in Section 3.3. The derivation of this uncertainty is described in Section 4.5.

An alternate approach was used to calculate the pin power uncertainty for gadolinia fuel pins. This alternate approach was adopted because the large thermal absorption cross sections associated with gadolinia results in the power density of gadolinia fuel rods being significantly lower than that of LEU fuel rods at, or near, BOC conditions. Consequently, gadolinia fuel rods are non-limiting at BOC conditions and a direct calculation of a pin power uncertainty from the B&W critical experiments at this condition would not be meaningful. For this reason, a multi-step process is used to calculate the gadolinia pin power uncertainty consistent with the process previously approved in DPC-NE-1005-P, Rev. 1 (Ref. 2). First, an uncertainty of CASMO-5 predicted to measured differences is determined from the evaluation of the B&W critical experiments. Next, the CASMO-5 predicted to measured difference uncertainty is converted to a relative uncertainty by assuming a conservative power density at which the gadolinia fuel rods



could potentially become limiting. Finally, this uncertainty is combined with the SIMULATE-3 to CASMO-5 pin power reconstruction uncertainty derived from a series of 2x2 colorset calculations to develop the gadolinia pin power uncertainty used in reload applications for the analysis of gadolinia fuel pins. Sections 4.6 and 4.7 describe the derivation of this uncertainty.

## **4.2 CASMO-5 Code Validation**

The B&W critical experiments were evaluated to demonstrate the CASMO-5 nuclear models are capable of accurately predicting the local pin power distributions for LEU fuel with and without gadolinia burnable absorbers. Results from the critical experiment benchmarks are described in Section 4.4.

The code vendor, Studsvik Scandpower (SSP), has also performed extensive benchmarking of CASMO-5 to validate the code's analytical models and methods. A description of the computational methods and models used in CASMO-5, and the benchmarks performed, are presented in the report titled: "CASMO5 PWR Methods and Validation Report" (Ref. 11). Validation of the CASMO-5 methods and models by SSP is accomplished by performance of a series of benchmark calculations against physical measurements and higher order computer codes (i.e. MCNP). Collectively, these benchmarks quantify the predictive accuracy of CASMO-5 with respect to reactivity, burnup (isotopics) and reaction rates (power). The higher order code calculations and computational benchmarks are used to study the mechanics of the code where measurements were unavailable. These computational benchmarks are also used to ensure proper method implementation over a full range of possible reactor conditions.

The benchmark calculations performed to quantify the predictive accuracy of the code with respect to reactivity, reaction rate and local power include comparisons against reactivity and power distribution measurements performed for the Babcock & Wilcox Urania Gadolinia critical experiments, Babcock & Wilcox 1484 critical experiments, KRITZ-3 critical experiments, and AEA Winfrith DIMPLE criticals. Qualification of the depletion models performance is provided by comparison against the Yankee Rowe isotopic measurements, and JAERI PWR isotopic benchmarks. Computational benchmarks included evaluation of the OECD C5G7 MOX lattice benchmark and MCNP6 lattice reactivity comparisons. The benchmark results show good agreement between prediction and measurement and higher order calculations.

### 4.3 B&W Critical Experiments

The Babcock & Wilcox (B&W) Urania Gadolinia critical experiments are described in Reference 21. They were performed to develop experimental data for the purpose of verifying the predictive capability of analytical computer models such as CASMO-5 and SIMULATE-3 for calculating pin power distributions for fuel lattices containing all LEU fuel, and fuel with uranium-gadolinia ( $\text{UO}_2+\text{Gd}_2\text{O}_3$ ) burnable absorbers. A series of 23 critical experiments were performed. Power distribution measurements were only performed for six of the critical experiments. Reactivity measurements for  $\text{UO}_2+\text{Gd}_2\text{O}_3$ , Ag-In-Cd and  $\text{B}_4\text{C}$  rods were performed for the remaining critical experiments. Since power distribution measurements were not available for these cores, they were not modeled by Duke. These experiments were however modeled by SSP, and the results of these reactivity calculations can be found in Reference 11.

Predicted-to-measured pin power uncertainties using CASMO-5 and SIMULATE-3 predicted data were calculated by analyzing measured power distributions for critical core configurations 1, 5, 12, 14, 18, and 20. The critical core configurations analyzed simulated B&W's 15x15 fuel assembly lattice and Combustion Engineering's 16x16 fuel assembly lattice with 2x2 water holes. Three of the critical core configurations contain all LEU fuel, and three contain both LEU and gadolinia fuel. A description of each core configuration is provided in Table 4-1. Cores 1, 5, 12 and 14 simulate B&W's 15x15 fuel assembly lattice, and cores 18 and 20 simulate Combustion Engineering's 16x16 fuel assembly lattice. Fuel enrichments range from 2.46 w/o to 4.02 w/o U-235 fuel. Table 4-2 shows the critical conditions of each experiment. Cores 1, 12 and 18 contain all LEU fuel, and cores 5, 14 and 20 contain both LEU and gadolinia fuel. Figures 4-1 through 4-3 show the location of LEU fuel pins, gadolinia fuel pins and empty water holes in each core configuration. The LEU and gadolinia fuel enrichment and gadolinia loadings are also provided.

Power distribution measurements were performed by counting the fission product gamma radiation produced from each fuel pin following irradiation. Each fuel rod was measured three times and the results averaged, and then normalized to an average relative power of 1.0. All measurements were performed at the point on the fuel rod corresponding to the mid-plane of the experiment.

#### 4.4 CASMO-5 and SIMULATE-3 Critical Experiment Benchmark

All six of the B&W critical experiments where pin power measurements were performed were evaluated to assess the capability of the CASMO-5 and SIMULATE-3 models to accurately calculate pin power distributions. The CASMO-5 input specification is general enough to model the asbuilt critical core configuration, and calculate the pin power distributions directly. However, small changes to the SIMULATE-3 model were required to model the critical core configurations. These changes are discussed below.

The SIMULATE-3 model is based on cross sections and assembly discontinuity factors developed using CASMO-5 for each critical configuration. Adjustments to the asbuilt core configuration were confined to relocating a small number of peripheral fuel pins to better model the partial fuel assemblies located at the exterior of the experiment. This modification was required because SIMULATE-3 was not designed to model partial fuel assemblies with few fuel pins, and because the reflector region in SIMULATE-3 must be void of fissionable material. Since all changes were confined to the core periphery, any impact to predicted powers in the core central region where the power measurements were performed was minimized.

Comparisons between measured and predicted pin power distributions for cores 1, 5, 12, 14, 18 and 20 were performed with both CASMO-5 and SIMULATE-3. Results from these comparisons are shown in Figures 4-4 through 4-9.

The CASMO-5 predicted-to-measured comparisons were performed to validate the code and to provide data for derivation of the gadolinia pin power uncertainty. SIMULATE-3 predicted-to-measured comparisons were performed to develop the LEU pin power distribution uncertainty factor for this code. The predicted power distributions have been normalized to an assembly average of 1.0 to provide for a consistent comparison with the measured power distribution. The results show that both CASMO-5 and SIMULATE-3 accurately predict the pin power distributions for fuel lattices with and without gadolinia. The predicted minus measured relative error for gadolinia fuel is larger than comparable LEU relative errors because of the low power density of the gadolinia fuel pins. The relative error definition used in all predicted-to-measured comparisons is defined below.

$$Relative\ Error = \frac{(P - M)}{M} * 100 \quad \text{eq. 4-1}$$

where,

$P$  = CASMO-5 or SIMULATE-3 predicted pin power

$M$  = Measured pin power

Results from a statistical evaluation between predicted and measured power distributions for both CASMO-5 and SIMULATE-3 are shown in Tables 4-3 and 4-4. Information presented includes means and standard deviations of the relative error for each core for LEU and gadolinia fuel pins, respectively. The mean relative error is calculated based on equation 4-2.

$$\text{Mean Relative Error} = \frac{1}{n} \sum_i^n \frac{(P_i - M_i)}{M_i} \quad \text{eq. 4-2}$$

where,

$P_i$  = the  $i^{\text{th}}$  predicted CASMO-5 or SIMULATE-3 pin power

$M_i$  = the  $i^{\text{th}}$  measured pin power

$n$  = sample size

Both CASMO-5 and SIMULATE-3 show a slight over prediction in the LEU pin powers, and a slight under-prediction in gadolinia pin powers relative to measurement. The percent deviations for gadolinia fuel are skewed because the average measured relative power density of the gadolinia fuel is only 0.158. Overall, the results show both CASMO-5 and SIMULATE-3 accurately predict pin power distributions in fuel lattices with and without gadolinia.

#### 4.5 SIMULATE-3 LEU Pin Power Uncertainty

Separate pin power uncertainty factors are developed for low enriched uranium fuel pins and gadolinia fuel pins. The approach used to calculate the LEU fuel pin uncertainty is described below. The approach used to calculate the gadolinia fuel pin power uncertainty is described in Sections 4.6 and 4.7

SIMULATE-3 predicted pin powers are compared against measured pin powers from each of the six B&W Urania Gadolinia critical experiments where measurements were performed to define the relative error in the predicted pin power. A one-sided upper tolerance limit uncertainty is developed to ensure with a 95% confidence level that 95% of the pin power predictions are equal

to or larger than the measured values. This statistical method requires that the data set pass a test for normality which is performed at a 1% significance level. The test for normality is performed using the D-Agostino (D-Prime) test defined in Reference 20. If a given data set fails this normality test, the uncertainty is determined by a non-parametric evaluation of the data. The statistical methods used are described in References 17 through 20.

A one-sided upper tolerance LEU pin uncertainty is developed for SIMULATE-3 using equation 4-3 where the  $K_p\sigma_p$  term is equal to the 95/95 one-sided upper tolerance uncertainty relative to the bias. A one-sided upper tolerance LEU pin uncertainty for CASMO-5 is also developed to show the fidelity of this transport theory model and for comparison purposes.

$$\text{LEU Pin Uncertainty} = -\text{Bias} + K_p\sigma_p \quad \text{eq. 4-3}$$

where,

$K_p$  = 95/95 one sided tolerance factor for a sample size of n (Ref. 19)

$\sigma_p$  = standard deviation of the predicted to measured pin power relative error

The subscript “p” denotes that these factors are pin parameters. The bias term is defined in equation 4-4 as the mean relative error between CASMO-5 or SIMULATE-3 and measured pin powers. All LEU fuel pins from critical experiments 1, 5, 12, 14, 18 and 20 are included. All gadolinia fuel pins are excluded.

$$\text{Bias} = \frac{1}{n} \sum_i^n \frac{(P_i - M_i)}{M_i} \quad \text{eq. 4-4}$$

where,

$n$  = sample size

$P_i$  = the  $i^{\text{th}}$  predicted pin power

$M_i$  = the  $i^{\text{th}}$  measured pin power

The CASMO-5 and SIMULATE-3 95/95 LEU pin power uncertainties are [ ]<sup>a,c</sup>, respectively. D-prime test results and the calculation of the pin uncertainties are shown Table 4-5. The results show both codes accurately predict LEU pin power distributions in fuel lattices containing all LEU fuel, and in fuel lattices with a mixture of LEU and gadolinia fuel pins. These comparisons also demonstrate the accuracy of the SIMULATE-3 pin power reconstruction methodology.

#### 4.6 Gadolinia Criticals CASMO-5 Pin Power Uncertainty

The approach used to calculate the gadolinia pin power uncertainty is consistent with that previously approved in DPC-NE-1005-PA (Ref. 2) and in DPC-NE-1006-PA (Ref. 22). The evaluation of the gadolinia critical experiments shows that CASMO-5 can accurately predict the pin power distribution for a fuel lattice with gadolinia. The results demonstrate the excellent predictive capability of CASMO-5 given the large flux gradients that exist between non-gadolinia and gadolinia fuel at the beginning of cycle (BOC) conditions represented by the critical experiments. Gadolinia pin power distribution comparisons show a predicted-to-measured bias of [ ]<sup>a,c</sup> with a standard deviation of [ ]<sup>a,c</sup>. These results are larger than the LEU bias and standard deviation because the average relative power of the gadolinia fuel pins is 0.158. The power density of the gadolinia fuel at or near BOC is a factor of 4 to 5 below that required to be potentially limiting because of the large thermal absorption cross sections associated with gadolinia. As a result, gadolinia fuel rods are non-limiting at BOC conditions, and a direct calculation of a pin power uncertainty from the B&W critical experiments at this condition would not be meaningful. It is not until the gadolinia fuel depletes and the power density of the gadolinia fuel approaches that of a LEU fuel pin power that a gadolinia fuel rod could potentially become limiting. Once the gadolinia depletes, the characteristics of the fuel rod are similar to that of a non-gadolinia fuel rod. Therefore, for the point in life at which a gadolinia fuel rod can become potentially limiting, its make-up is essentially that of a non-gadolinia rod, and the gadolinia pin power uncertainty should approach that of a non-gadolinia fuel rod. A representative gadolinia pin power uncertainty at this condition is what is required for use in safety related calculations. For this reason, the uncertainty for gadolinia fuel is developed using a conservatively assumed measured gadolinia pin power of [ ]<sup>a,c</sup> with predicted minus measured gadolinia pin power differences calculated from evaluation of critical experiments 5, 14, and 20. This formulation is shown below in equation 4-5.

$$\text{Gad Critical Experiment Pin Uncertainty} = - \frac{\text{Bias}}{\bar{M}} + \frac{K\sigma}{\bar{M}} \quad \text{eq. 4-5}$$

where,

Bias = mean difference of the population (Predicted - Measured)

$\bar{M}$  = Average of measured powers [ ]<sup>a,c</sup>

K = 95/95 one sided tolerance factor based on the sample size from Reference 19

$\sigma$  = standard deviation of the difference

An average gadolinia pin power of [ ]<sup>a,c</sup> was chosen instead of 1.0 in the above calculation because the presence of Gd<sub>2</sub>O<sub>3</sub> in fuel results in a lower thermal conductivity and heat capacity relative to standard UO<sub>2</sub> fuel rods. The reduced thermal conductivity of the gadolinia fuel results in an increase in the initial fuel temperature of gadolinia fuel rods relative to UO<sub>2</sub> fuel rods. This temperature increase has a negligible impact on DNB because the decrease in fuel thermal conductivity impedes the transfer of heat out of the fuel pellet. However, the increase in initial fuel temperature of gadolinia fuel relative to UO<sub>2</sub> fuel results in centerline fuel melt (CFM) limits for gadolinia fuel being challenged at lower power densities than UO<sub>2</sub> fuel. Duke has determined that the fractional decrease in CFM linear heat rate limits is approximately [ ]<sup>a,c</sup> depending upon the time in life. The limiting value was used as the basis for lowering the mean pin power for determining the gadolinia pin power uncertainty from 1.0 to [ ]<sup>a,c</sup> to account for the decrease in thermal conductivity and heat capacity associated with gadolinia fuel rods.

The W test (Ref. 20) for normality was used to confirm the assumption of normality at a 1% level of significance for gadolinia predicted minus measured differences. This test was used in lieu of the D-prime test because of the small sample size. Results from the W test are shown in Table 4-6. The distribution was found to be nearly normal, and the assumption of normality was applied. The mean and standard deviation of the CASMO-5 predicted to measured gadolinia pin power differences are also shown in Table 4-6. This information is used to calculate a CASMO-5 gadolinia pin power uncertainty using equation 4-5 with a measured power [ ]<sup>a,c</sup> below the assembly average power ( $\bar{M} = [ ]^{\text{a,c}}$ ). This results in a CASMO-5 pin power uncertainty for gadolinia fuel of [ ]<sup>a,c</sup> based on a K value of [ ]<sup>a,c</sup> from Reference 19 [ ]<sup>a,c</sup>. This uncertainty is combined with the SIMULATE-3 predicted uncertainty relative to CASMO-5 for gadolinia fuel rods determined from 2x2 colorset pin power distribution comparisons between SIMULATE-3 and CASMO-5.

#### 4.7 Theoretical 2x2 Calculation and SIMULATE-3 Gadolinia Pin Power Uncertainty

A series of theoretical infinite lattice 2x2 colorset cases were developed and evaluated with both CASMO-5 and SIMULATE-3 for the purpose of demonstrating the fidelity of the SIMULATE-3 model's pin power reconstruction methodology for a diverse set of fuel lattices, and to develop a SIMULATE-3 to CASMO-5 pin power uncertainty for gadolinia fuel pins. The 2x2 cases

evaluated are for representative fuel lattice configurations expected in a typical reload core. They encompass expected core geometries between fresh and burned fuel with different combinations of fuel enrichment, and the number and enrichment of the gadolinia burnable absorber rods. Feed fuel face-adjacent and feed fuel checkerboard patterns next to burned fuel assemblies were considered along with fuel rod diameter differences associated with 15x15 and 17x17 fuel lattice designs. The case matrix evaluated represents a diverse combination of reactivity and neutron spectrum differences between adjacent fuel assemblies including depletion effects. Gadolinia concentrations from 2.0 to 8.0 w/o of  $Gd_2O_3$  were considered.

Figure 4-10 shows the 2x2 configurations evaluated including lattice geometry, fuel enrichment, assembly burnup, and the number and gadolinia loading present in each fuel lattice. CASMO-5 single fuel assembly executions were performed to develop the input for the SIMULATE-3 model for each fuel assembly type modeled in the 2x2 calculations. The setup of the 2x2 SIMULATE-3 model was consistent with how a reload core would be set up. Each colorset was depleted for 20 GWD/MTU with once-burned reinserted fuel reaching assembly average burnups of approximately 40 GWD/MTU. Separate CASMO-5 models for each 2x2 case shown in Figure 4-10 were set up and run to develop power distribution information for comparison to the SIMULATE-3 models' results.

Comparisons between SIMULATE-3 and CASMO-5 two-dimensional pin-by-pin power distributions were performed at several burnups to validate the ability of SIMULATE-3 to accurately reproduce CASMO-5 pin power distributions for varying reload combinations of fresh and feed fuel placement, fuel enrichment, burnup, and integral and lumped burnable poison rod type. Quantitative results from the 2x2 theoretical benchmark comparisons are shown in Table 4-9 for each 2x2 configuration. The statistical results presented include the standard deviations calculated for all fuel pins, LEU fuel pins, and gadolinia fuel pins. The comparisons demonstrate the accuracy of the SIMULATE-3 pin power reconstruction methodology and show that SIMULATE-3 can reproduce the CASMO-5 predicted pin power distributions with similar accuracy for diverse combinations of fuel enrichment, fuel types, burnup, and depletion of integral and lumped burnable absorbers.

The SIMULATE-3 pin power reconstruction uncertainty factor for gadolinia fuel is developed from comparisons between CASMO-5 and SIMULATE-3 power distribution data for gadolinia fuel pins from the 2x2 benchmark cases. The data set was tested for normality using the D-prime test for normality at a 1% level of significance as described in Reference 20. The normality test



results are shown in Table 4-7. The data set was not normal, and as a result a non-parametric evaluation of the data was performed as described in References 17 and 18 to determine the 95/95 one sided tolerance uncertainty. The 95/95 uncertainty of a distribution is the  $m^{\text{th}}$  worst comparison where  $m$  is a function of the number of comparisons. Values of  $m$  were derived from Reference 17. The 95/95 SIMULATE-3 pin power reconstruction pin power uncertainty calculated based on the theoretical benchmark comparisons is [ ]<sup>a,c</sup> with a bias of [ ]<sup>a,c</sup>. Refer to Table 4-8. This uncertainty includes depletion effects, reactivity, and neutron spectrum differences caused by dissimilar fuel types adjacent to one another.

The SIMULATE-3 gadolinia pin power uncertainty is determined by combining the SIMULATE-3 pin power reconstruction uncertainty with the CASMO-5 uncertainty developed from the CASMO-5 evaluation of the gadolinia critical experiments (section 4.6). The statistical combination of these uncertainties is performed as follows:

*SIMULATE Gadolinia Pin Power Uncertainty*

$$= -(Bias_c + Bias_{pr}) + \sqrt{(K_c \sigma_c)^2 + (K_{pr} \sigma_{pr})^2}$$

$$= [ ]^{a,c}$$

where, bias = [ ]<sup>a,c</sup> for Gad fuel, and

$$\sqrt{(K_c \sigma_c)^2 + (K_{pr} \sigma_{pr})^2} = [ ]^{a,c}$$

The resulting SIMULATE-3 gadolinia pin power uncertainty is [ ]<sup>a,c</sup>. An uncertainty value greater than or equal to this value will be used in reload applications. As new data becomes available, the gadolinia pin power uncertainty can be updated using the methodology described in this report.

#### 4.8 LEU and Gadolinia Fuel Pin Uncertainties

The LEU fuel pin and gadolinia fuel pin uncertainties to be applied in reload application are [ ]<sup>a,c</sup>, respectively.

**Table 4-1**  
**Critical Experiment Characteristics**

Core	Enrichment	Gad Fuel	Description
1	2.46 w/o U-235	None	15x15 B&W lattice. Entire core consists of 2.46 w/o fuel.
5	2.46 w/o U-235	4.0 w/o Gd <sub>2</sub> O <sub>3</sub> /1.94 w/o U-235 enriched fuel	15x15 B&W lattice with 12 Gad pins. Entire core consists of 2.46 w/o fuel.
12	4.02 w/o U-235 (Inner) 2.46 w/o U-235 (Outer)	None	15x15 B&W lattice. The central region consists of 4.02 w/o fuel with the remainder of the core consisting of 2.46 w/o fuel.
14	4.02 w/o U-235 (Inner) 2.46 w/o U-235 (Outer)	4.0 w/o Gd <sub>2</sub> O <sub>3</sub> /1.94 w/o U-235 enriched fuel	15x15 B&W lattice with 12 Gad pins. The central region consists of 4.02 w/o fuel with the remainder of the core consisting of 2.46 w/o fuel.
18	4.02 w/o U-235 (Inner) 2.46 w/o U-235 (Outer)	None	16x16 CE lattice. The central region consists of 4.02 w/o fuel with the remainder of the core consisting of 2.46 w/o fuel.
20	4.02 w/o U-235 (Inner) 2.46 w/o U-235 (Outer)	4.0 w/o Gd <sub>2</sub> O <sub>3</sub> /1.94 w/o U-235 enriched fuel	16x16 CE lattice with 16 Gad pins. The central region consists of 4.02 w/o fuel with the remainder of the core consisting of 2.46 w/o fuel.

**Table 4-2**  
**Critical Experiment Configuration and Critical Conditions**

Core	Configuration	Water Temperature (°F)	Boron Concentration (ppm)
1	15x15 2.46 w/o U-235 enriched fuel	77	1338
5	15x15 2.46 w/o U-235 enriched fuel 12 4.0 w/o Gd <sub>2</sub> O <sub>3</sub> /1.94 w/o U-235 pins	77	1208
12	15x15 4.02 w/o U-235 fuel (central) 2.46 w/o U-235 fuel (peripheral)	77	1899
14	15x15 4.02 w/o U-235 fuel (central) 2.46 w/o U-235 fuel (peripheral) 12 4.0 w/o Gd <sub>2</sub> O <sub>3</sub> /1.94 w/o U-235 pins	77	1654
18	16x16 4.02 w/o U-235 fuel (central) 2.46 w/o U-235 fuel (peripheral)	77	1777
20	16x16 4.02 w/o U-235 fuel (central) 2.46 w/o U-235 fuel (peripheral) 16 4.0 w/o Gd <sub>2</sub> O <sub>3</sub> /1.94 w/o U-235 pins	77	1499

**Table 4-3**  
**B&W Criticals Statistical Summary for LEU Fuel Pins**  
**Mean and Standard Deviation of the Relative Error**

Core	N	CASMO-5 Mean	CASMO-5 Std. Deviation	SIMULATE-3 Mean	SIMULATE-3 Std. Deviation
1	32	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
5	29	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
12	32	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
14	29	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
18	32	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
20	29	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
All	183	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c

**Table 4-4**  
**B&W Criticals Statistical Summary for Gadolinia Fuel Pins**  
**Mean and Standard Deviation of the Relative Error**

Core	N	CASMO-5 Mean	CASMO-5 Std. Deviation	SIMULATE-3 Mean	SIMULATE-3 Std. Deviation
5	5	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
14	5	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
20	5	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c
All	15	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c

**Table 4-5**  
**B&W Criticals D-Prime Test For Normality**  
**and Pin Power Uncertainty for LEU Fuel Pins**

**LEU Fuel D-Prime Test For Normality Results**

Parameter	CASMO-5	SIMULATE-3
N	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
D' (P = 0.005)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
D'	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
D' (P = 0.995)	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
Evaluation	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

**LEU Fuel Pin Power Uncertainty Calculation Results**

Core	N	Bias	Standard Deviation	K	Uncertainty
CASMO-5	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>
SIMULATE-3	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

**Table 4-6**  
**B&W Criticals CASMO-W Test For Normality**  
**and Pin Power Uncertainty for Gadolinia Fuel Pins**

**Mean and Standard Deviation**  
**Of the Predicted to Measured Difference**

Difference (P – M)	
Mean	Standard Deviation
[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

**Normality Test Using the W Test for Normality**

Parameter	Value
N	[ ] <sup>a,c</sup>
W(P = 0.01)	[ ] <sup>a,c</sup>
W of Distribution	[ ] <sup>a,c</sup>
Evaluation	[ ] <sup>a,c</sup>

**CASMO-5 Relative Error Uncertainty Calculation Results for Gadolinia Fuel Pins**

N	$\bar{M}$	Bias	Standard Deviation (%)	K	Uncertainty <sup>+</sup>
[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

+ Based on Equation 4-5 where,

$$Bias = Mean / \bar{M} * 100$$

$$Stdev(\%) = Stdev / \bar{M} * 100$$

**Table 4-7**  
**SIMULATE-3 2x2 Colorset Evaluation**  
**Pin Power Reconstruction D-Prime Normality Test Results For Gadolinia Fuel Pins**

**D-Prime Normality Test Results**

Parameter	Value
N	[ ] <sup>a,c</sup>
D' (P = 0.005)	[ ] <sup>a,c</sup>
D'	[ ] <sup>a,c</sup>
D' (P = 0.995)	[ ] <sup>a,c</sup>
Evaluation	[ ] <sup>a,c</sup>

**Table 4-8**  
**SIMULATE-3 2x2 Colorset Evaluation**  
**Pin Power Reconstruction Uncertainty For Gadolinia Fuel Pins**

N	Bias	Standard Deviation (%)	m	Non-Parametric Uncertainty
[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>	[ ] <sup>a,c</sup>

Non Parametric Uncertainty = - E<sub>mth</sub>  
(E<sub>mth</sub> is the m<sup>th</sup> smallest relative error for a sample size of N)

**Table 4-9**  
**SIMULATE-3 to CASMO-5 Theoretical 2x2 Colorset Comparison Results**

Case #	FA Exposure (GWD/MTU)	PIN POWER COMPARISONS					
		ALL Pins		UO2 Only Pins		Gd Pins	
		Mean	Std. Dev.	Mean	Std. Dev.	Mean	Std. Dev.
1a							
1b							
2a							
2b							
3a							
3b							
4a							
4b							
5a							
5b							
6a							
6b							
7a							
7b							

a,c

**Table 4-9 cont'd**  
**SIMULATE-3 to CASMO-5 Theoretical 2x2 Colorset Comparison Results**

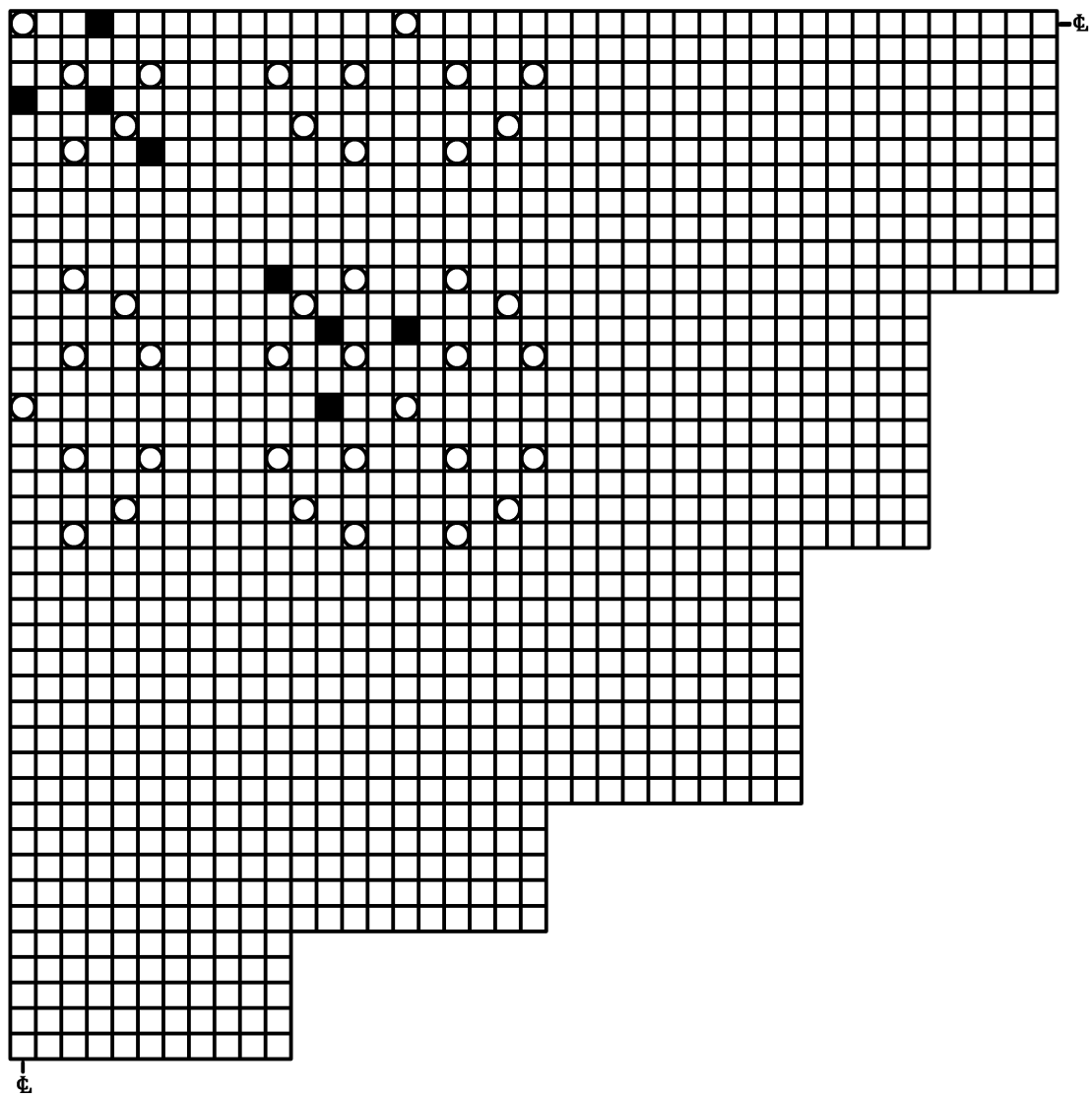
Case #	FA Exposure (GWD/MTU)	PIN POWER COMPARISONS					
		ALL Pins		UO2 Only Pins		Gd Pins	
		Mean	Std. Dev.	Mean	Std. Dev.	Mean	Std. Dev.
8a							
8b							
9a							
9b							
10a							
10b							
11a							
11b							

a,c




$$\text{Mean} = (\text{SIMULATE-3} - \text{CASMO-5}) / \text{CASMO-5} * 100$$






**Figure 4-1**  
**B&W Critical Experiments**  
**Core 1 and 5 Loading Plans**



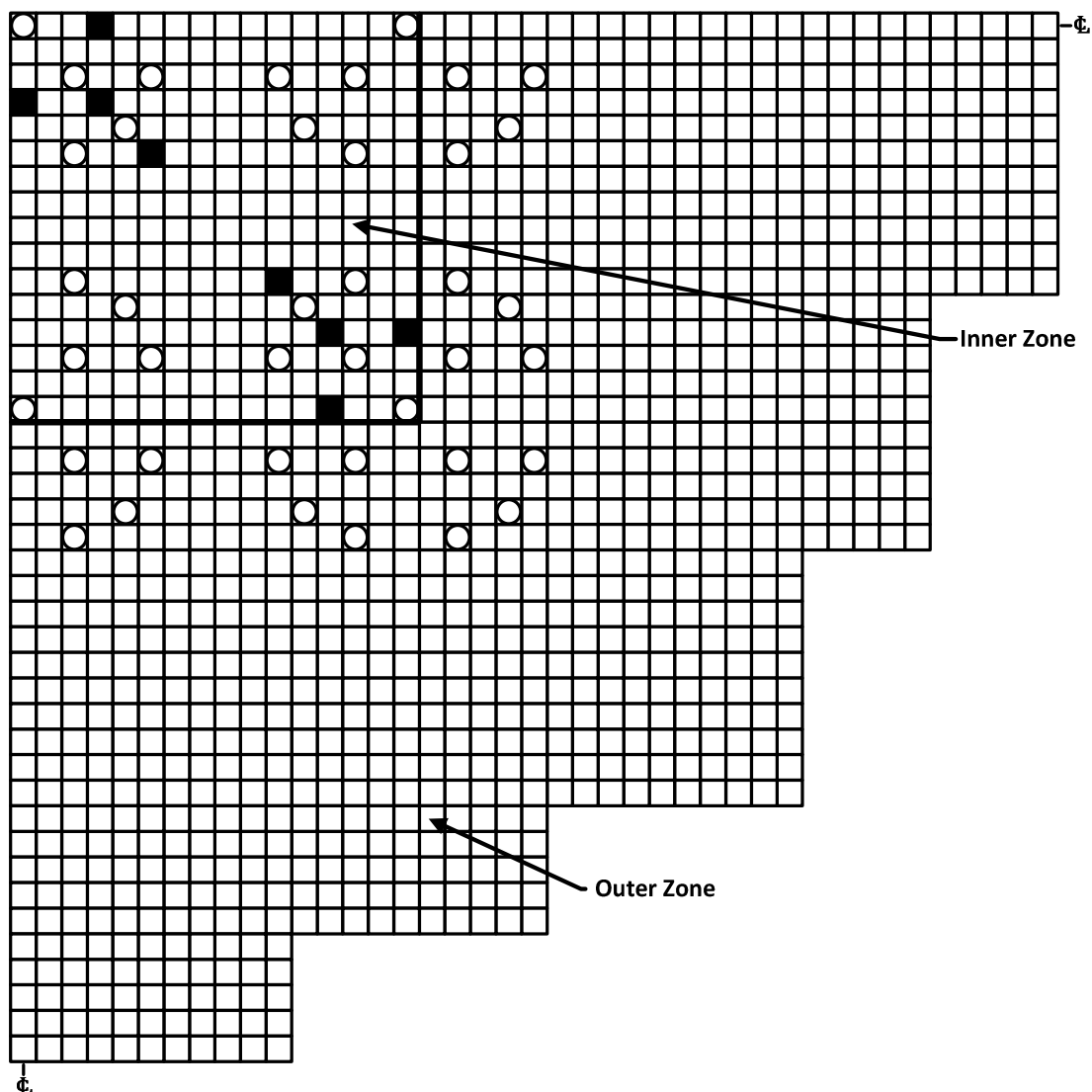
**Core 1**

-  VACANT WATER-FILLED POSITION
-  2.46 wt % U-235 ENRICHED FUEL
-  2.46 wt % U-235 ENRICHED FUEL

**Core 5**

-  VACANT WATER-FILLED POSITION
-  2.46 wt % U-235 ENRICHED FUEL
-  4.00 wt % Gd<sub>2</sub>O<sub>3</sub>/1.94 wt % U-235 ENRICHED FUEL

**Figure 4-2**  
**B&W Critical Experiments**  
**Core 12 and 14 Loading Plans**



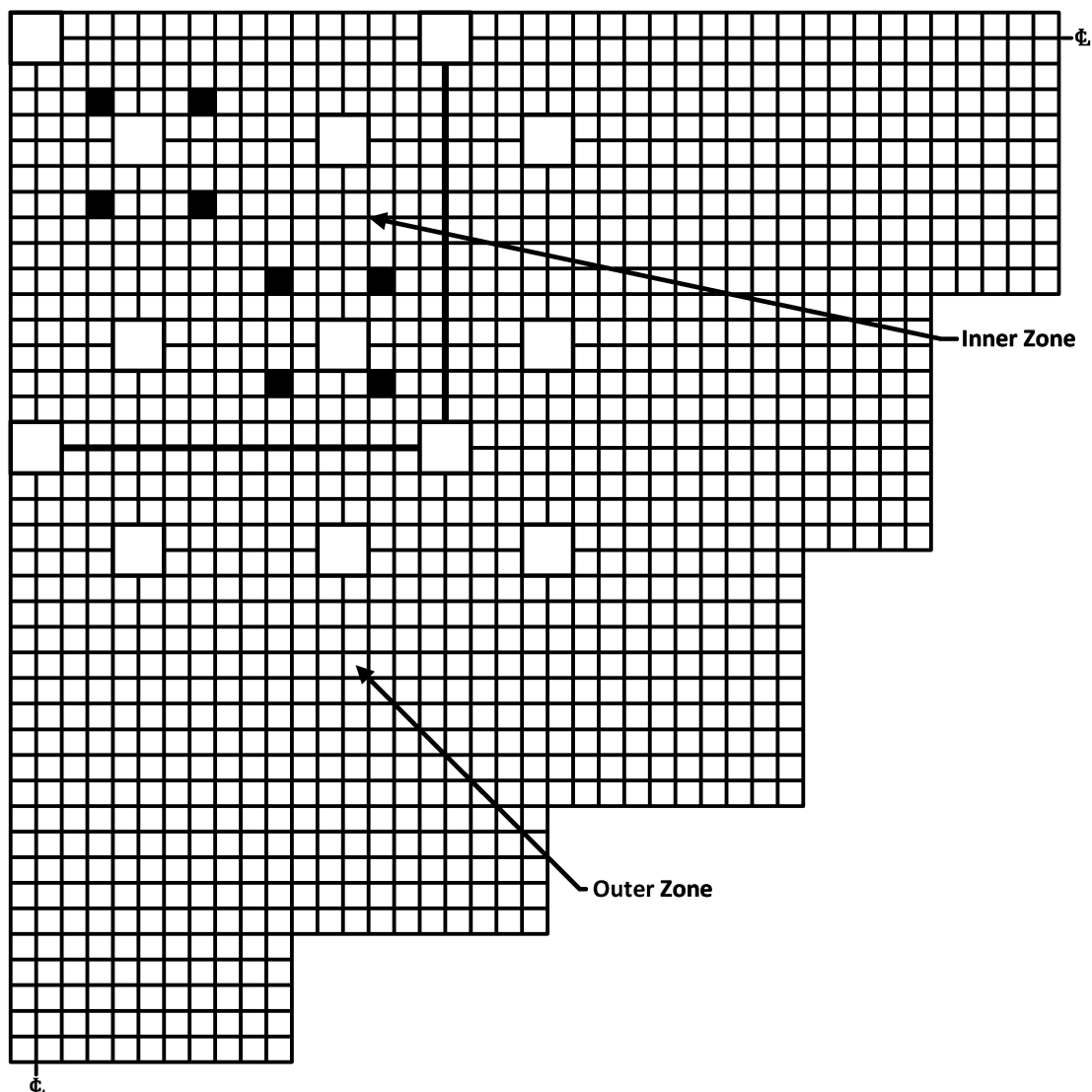
**Core 12**

- ☒ VACANT WATER-FILLED POSITION
- ☐ 2.46 wt % U-235 ENRICHED FUEL: OUTER ZONE
- ☐ 4.02 wt % U-235 ENRICHED FUEL: INNER ZONE
- ☒ 4.02 wt % U-235 ENRICHED FUEL: INNER ZONE

**Core 14**

- ☒ VACANT WATER-FILLED POSITION
- ☐ 2.46 wt % U-235 ENRICHED FUEL: OUTER ZONE
- ☐ 4.02 wt % U-235 ENRICHED FUEL: INNER ZONE
- ☒ 4.00 wt %  $Gd_2O_3$ /1.94 wt % U-235 ENRICHED FUEL

**Figure 4-3**  
**B&W Critical Experiments**  
**Core 18 and 20 Loading Plans**



**Core 18**

	VACANT WATER-FILLED POSITION
	2.46 wt % U-235 ENRICHED FUEL: OUTER ZONE
	4.02 wt % U-235 ENRICHED FUEL: INNER ZONE
	4.02 wt % U-235 ENRICHED FUEL: INNER ZONE

**Core 20**

	VACANT WATER-FILLED POSITION
	2.46 wt % U-235 ENRICHED FUEL: OUTER ZONE
	4.02 wt % U-235 ENRICHED FUEL: INNER ZONE
	4.00 wt % Gd <sub>2</sub> O <sub>3</sub> /1.94 wt % U-235 ENRICHED FUEL

**Figure 4-4**  
**B&W Urania Gadolinia Critical Experiments – Core 1**  
**CASMO-5 and SIMULATE-3 Calculated Versus Measured Pin Power Distributions**

**CASMO-5 Versus Measured Pin Power Distribution**

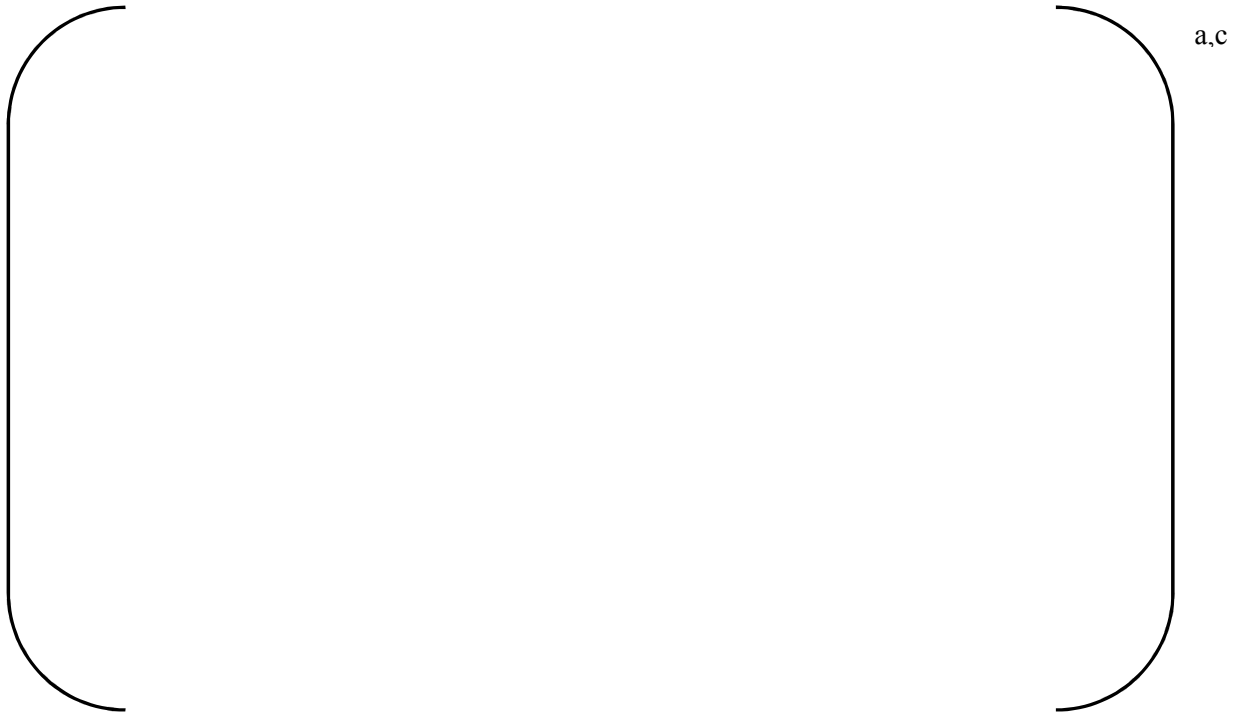


**SIMULATE-3 Versus Measured Pin Power Distribution**



**Figure 4-5**  
**B&W Urania Gadolinia Critical Experiments – Core 5**  
**CASMO-5 and SIMULATE-3 Calculated Versus Measured Pin Power Distributions**

**CASMO-5 Versus Measured Pin Power Distribution**



**SIMULATE-3 Versus Measured Pin Power Distribution**



**Figure 4-6**  
**B&W Urania Gadolinia Critical Experiments – Core 12**  
**CASMO-5 and SIMULATE-3 Calculated Versus Measured Pin Power Distributions**

**CASMO-5 Versus Measured Pin Power Distribution**



**SIMULATE-3 Versus Measured Pin Power Distribution**



**Figure 4-7**  
**B&W Urania Gadolinia Critical Experiments – Core 14**  
**CASMO-5 and SIMULATE-3 Calculated Versus Measured Pin Power Distributions**

**CASMO-5 Versus Measured Pin Power Distribution**



**SIMULATE-3 Versus Measured Pin Power Distribution**



**Figure 4-8**  
**B&W Urania Gadolinia Critical Experiments – Core 18**  
**CASMO-5 and SIMULATE-3 Calculated Versus Measured Pin Power Distributions**

**CASMO-5 Versus Measured Pin Power Distribution**



**SIMULATE-3 Versus Measured Pin Power Distribution**





**Figure 4-9**  
**B&W Urania Gadolinia Critical Experiments – Core 20**  
**CASMO-5 and SIMULATE-3 Calculated Versus Measured Pin Power Distributions**

**CASMO-5 Versus Measured Pin Power Distribution**



**SIMULATE-3 Versus Measured Pin Power Distribution**



This image shows a single sheet of white paper with rounded corners. The paper is ruled with thin, light blue horizontal lines spaced evenly apart. There are no margins, text, or other markings on the page.

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## 5.0 Statistically Combined Power Distribution Uncertainty Factors

Power distribution uncertainty factors are applied in both the design of reload cores and the surveillance of an operating fuel cycle. In each case the uncertainty factor is applied to power distribution peaking factors to ensure a conservative comparison to thermal design limits on fuel pin performance. Because a direct measurement of individual pin power distribution is not available during power reactor operation, the complete uncertainty in the core model's ability to predict pin power distributions must be constructed from a synthesis of power reactor and critical experiment benchmark results. In its generic form, this synthesis can be expressed mathematically as:

$$SCUF = 1 - \sum_{i=1}^n Bias_i + \sqrt{\sum_{i=1}^n (K_i \sigma_i)^2}$$

Where,

$SCUF$  is the statistically combined uncertainty factor,

$\sum_{i=1}^n Bias_i$  is the sum of individual biases (both assembly and pin),

$\sqrt{\sum_{i=1}^n (K_i \sigma_i)^2}$  is the combination by square root, sum of the squares of the individual 95/95 statistical uncertainties contributing to the total uncertainty, and

$n$  is the number of factors combined.

For the data sets that are shown to be normally distributed,  $K_i \sigma_i$  is determined directly from the product of the 95/95 one-sided upper tolerance factor ( $K_i$ ) times the standard deviation ( $\sigma_i$ ) of the data set. For data sets that do not pass a test for normality,  $K_i \sigma_i$  is determined by the non-parametric evaluation of the data as described in Section 3.3.

Power distribution uncertainty factors for application to LEU fuel and gadolinia fuel are developed in Sections 5.1 and 5.2.

## 5.1 LEU Fuel Uncertainty Factor

The statistically combined uncertainty factor (SCUF) for a LEU fuel pin, in a LEU lattice or a LEU/Gadolinia fuel lattice, is determined from equation 5-1:

$$SCUF = 1 - (Bias_a + Bias_p) + \sqrt{(K_a\sigma_a)^2 + (K_p\sigma_p)^2} \quad \text{eq. 5-1}$$

where terms  $Bias_a$  and  $K_a\sigma_a$  represent the bias and statistical deviation in the comparisons between measured and predicted inter-assembly power distributions, and the  $Bias_p$  and  $K_p\sigma_p$  terms are from comparisons between measured and predicted intra-assembly LEU pin power distributions. The  $Bias_a$  and  $K_a\sigma_a$  terms are derived from the power distribution analysis results in Section 3.3, and the  $Bias_p$  and  $K_p\sigma_p$  terms are from the LEU pin uncertainty analysis from the SIMULATE-3 modeling of the B&W critical experiments in Section 4.5. The calculated SIMULATE-3 SCUFs for a LEU fuel pin are shown in Table 5-1.

## 5.2 Gadolinia Fuel Uncertainty Factor

The statically combined uncertainty for a gadolinia fuel pin is calculated using equation 5-2:

$$SCUF = 1 - (Bias_a + Bias_{pg}) + \sqrt{(K_a\sigma_a)^2 + (K_{pg}\sigma_{pg})^2} \quad \text{eq. 5-2}$$

where terms  $Bias_a$  and  $K_a\sigma_a$  represent the bias and statistical deviation in the comparisons between measured and predicted inter-assembly power distributions, and the terms  $Bias_{pg}$  and  $K_{pg}\sigma_{pg}$  are the equivalent terms from the comparison between measured and predicted intra-assembly gadolinia pin power distributions. The assembly  $Bias_a$  and  $K_a\sigma_a$  terms are derived from the power distribution analysis results from Section 3.3, and are the same terms used for derivation of the LEU fuel pin uncertainty. The  $K_{pg}\sigma_{pg}$  term is the SIMULATE-3 gadolinia pin power uncertainty from Section 4.7. The gadolinia pin power distribution uncertainty contains two components. The first component was derived from comparisons of the CASMO-5 calculated pin powers to measured gadolinia pin powers from the B&W gadolinia critical experiments evaluated in Section 4.6. The second component is the SIMULATE-3 pin power

reconstruction uncertainty derived from 2x2 SIMULATE-3 to CASMO-5 colorset power distribution comparisons performed in Section 4.7. These two terms were combined to develop a total pin power uncertainty for gadolinia fuel of [ ]<sup>a,c</sup> in Section 4.7. This uncertainty is composed of a [ ]<sup>a,c</sup> bias term and [ ]<sup>a,c</sup> uncertainty term ( $K_{pg}\sigma_{pg}$  term). The bias terms from the inter-assembly and pin calculations are combined. The calculated SIMULATE-3 SCUFs for a gadolinia fuel pin are shown in Table 5-1.

**Table 5-1**  
**LEU and Gadolinia Statistically Combined Uncertainty Factors**

**LEU Fuel**

Parameter	Assembly Bias	Pin Bias	Assembly Uncertainty ( $K_a\sigma_a$ )	Pin Uncertainty ( $K_p\sigma_p$ )	SCUF
FΔH	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Table 4-5)	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Table 4-5)	[            ] <sup>a,c</sup>
Fq	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Table 4-5)	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Table 4-5)	[            ] <sup>a,c</sup>
Fz	[            ] <sup>a,c</sup> (Table 3-6)	N/A	[            ] <sup>a,c</sup> (Table 3-6)	N/A	[            ] <sup>a,c</sup>

**Gadolinia Fuel**

Parameter	Assembly Bias	Pin Bias	Assembly Uncertainty ( $K_a\sigma_a$ )	Pin Uncertainty ( $K_{pg}\sigma_{pg}$ )	SCUF
FΔH	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Section 4.7)	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Section 4.7)	[            ] <sup>a,c</sup>
Fq	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Section 4.7)	[            ] <sup>a,c</sup> (Table 3-6)	[            ] <sup>a,c</sup> (Section 4.7)	[            ] <sup>a,c</sup>
Fz	[            ] <sup>a,c</sup> (Table 3-6)	N/A	[            ] <sup>a,c</sup> (Table 3-6)	N/A	[            ] <sup>a,c</sup>

## 6.0 Conclusion

This report justifies the use of the CASMO-5/SIMULATE-3 nuclear design methodology for performing reload design analyses for Duke's Harris, Robinson, McGuire, and Catawba nuclear plants. Benchmark calculations using the CASMO-5/SIMULATE-3 code system were presented that demonstrate the CASMO-5/SIMULATE-3 methodology can accurately predict the core behavior for reactor cores containing LEU fuel with both discrete and integral burnable absorbers. The benchmark calculations presented consisted of comparisons between predicted and measured parameters for fourteen Harris, Robinson, and McGuire fuel cycles. Collectively, the code qualification included benchmark calculations encompassing both 3-loop 157 assembly and 4-loop 193 assembly reactor core designs, 15x15 and 17x17 fuel lattice geometries, and fuel containing B4C discrete burnable absorbers, zirconium diboride integral fuel rod burnable absorbers (IFBA), and gadolinia integral fuel burnable absorbers. Comparisons between predicted and measured beginning-of-cycle hot zero power startup physics testing and HFP operating condition data were presented and used to determine the accuracy of the CASMO-5 based SIMULATE-3 core models. The comparisons presented demonstrate the excellent reactivity and power distribution predictive capability of the CASMO-5 based SIMULATE-3 core models for reload analysis of reactor cores containing LEU fuel with gadolinia and zirconium diboride integral fuel burnable absorbers and discrete burnable absorbers.

Fuel assembly peaking factor ( $F_{\Delta H}$ ,  $F_q$  and  $F_z$ ) Observed Nuclear Reliability Factors (ONRFs) were developed from comparisons between calculated and measured power distributions obtained during normal operation. The reliability factors calculated are applicable to reactor cores containing all LEU and LEU/gadolinia fuel lattices.

Pin power distribution uncertainties were developed based on the evaluation of the B&W Urania Gadolinia Critical experiments and the CASMO-5/SIMULATE-3 theoretical infinite lattice 2x2 colorset benchmark problems. Separate uncertainties were developed for low enriched uranium (LEU) fuel pins in an all LEU fuel lattice or a LEU/gadolinia fuel lattice, and for gadolinia fuel pins. The benchmark results are used as the basis for the pin power uncertainties included in the calculation of the statistically combined uncertainty factors for LEU and gadolinia fuel pins.



The report also presents the statistically combined  $F_{\Delta H}$ ,  $F_q$  and  $F_z$  uncertainty factors for LEU and gadolinia fuel for application in reload design safety-related analyses. These values are provided in Table 5-1.

In summary, the CASMO-5/SIMULATE-3 methodology is acceptable for use in performing reactivity and power distribution reactor core calculations for input into safety-related reload design analyses for reactor cores containing all LEU fuel, or a mixture of LEU and gadolinia fuel. CASMO-5/SIMULATE-3 core models are developed for each core design in accordance with the methodology described in this report. Each model explicitly models the fuel types present in the reactor core designs. This process, coupled with code and model quality assurance practices required by Duke's quality assurance program, and continued assessment of model performance, provides assurance that future changes to core, fuel and burnable poison designs will be modeled with accuracy and appropriate conservatism. In addition, it is anticipated that as additional model insights are gained through continued benchmark of the CASMO-5/SIMULATE-3 model against measured data, CASMO-5/SIMULATE-3 improvements can be made and applied to future core designs.

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**Attachment 6**

**DPC-NF-2010, Revision 3, “Nuclear Physics Methodology for Reload Design” and  
Technical Justification of Changes**

~~McGuire Nuclear Station~~  
~~Catawba Nuclear Station~~

Nuclear Physics Methodology  
For Reload Design

DPC-NF-2010-A  
Revision ~~2a~~**3**

~~December 2009~~ **January 2016**

Nuclear **Fuels** Engineering ~~Division~~  
Nuclear Generation Department  
Duke Energy Carolinas, LLC  
**and**  
**Duke Energy Progress, INC**

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### Revision History

Revision	Description
DPC-NF-2010, Original Issue	Originally submitted to the NRC for approval in July 1984. Additional information was submitted to the NRC supplying responses to a request for additional information.
DPC-NF-2010-A, Original Issue	NRC approved version issued in May 1985.
DPC-NF-2010, Revision 1	<p>Submitted to the NRC for approval in August 2001.</p> <p>This revision updates the report for completeness to indicate the use of NRC approved methods approved subsequent to the implementation of the original issue including the use of CASMO-3/SIMULATE-3 reactor physics methods.</p> <p>This revision also removes unnecessary data including items not reviewed by the NRC in the original version and general fuel data not necessary for these methods.</p> <p>Also various editorial changes are made, including updating the table of contents.</p> <p><del>Changes associated with this revision are denoted by revision bars, except format changes.</del></p>
DPC-NF-2010-A, Revision 1	NRC approved. SER issued October 1, 2002.
DPC-NF-2010, Revision 2	<p>Submitted to the NRC for approval in November 2002. This revision removed the description of the source of fuel temperature data for the reactor physics code.</p> <p><del>Changes associated with this revision are denoted by revision bars, and previous revision bars are removed.</del></p>
DPC-NF-2010-A Revision 2	NRC approved. SER issued June 24, 2003.
DPC-NF-2010-A Revision 2a	<p>The primary purpose of Revision 2a is to remove obsolete information that has been superseded by newer NRC methodology reports. The report is also restructured for ease of use. The major elements of this revision are:</p> <ul style="list-style-type: none"> <li>• Removal of EPRI-ARMP PDQ07 and EPRI-NODE-P methodology</li> <li>• Replacement of the CASMO-3/SIMULATE-3 methodology with the CASMO-4/SIMULATE-3 methodology</li> </ul>

### **Revision History cont'd**

DPC-NF-2010-A Revision 2a cont'd	<ul style="list-style-type: none"><li>• Deletion of benchmark results</li><li>• Consolidation of calculation methods for physics parameters and reactivity coefficients into Section 5</li><li>• Editorial changes, error corrections and clarifications</li></ul>
DPC-NF-2010-A Revision 3	<p><b>Extends the applicability of the Duke Energy Carolina's nuclear design methodology to the Duke Energy Progress Harris and Robinson nuclear units.</b></p> <p><b>Incorporates by reference the CASMO-5/SIMULATE-3 nuclear analysis methodology as a methodology that can be used for reload design analyses.</b></p>

## ABSTRACT

This report describes Duke Energy's Carolinas, LLC Nuclear Design Methodology for the McGuire, and Catawba, Nuclear Station **Harris and Robinson nuclear units**. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology Duke Energy intends to use to perform design calculations, to provide operational support, and to develop statistical reliability factors.

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## 1. INTRODUCTION

### 1.1 General Nuclear Design Description

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

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

In arriving at the final nuclear design, the designer tries to meet the requirements imposed by the operational considerations, fuel economics considerations, and safety considerations. These requirements are called nuclear design criteria and some of the key criteria are as follows:

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

2. The fuel assemblies to be discharged at the end of the fuel cycle will attain maximum permissible burnup so that maximum energy extraction consistent with the fuel mechanical integrity criteria is achieved.
3. Values of important core parameters (moderator temperature coefficient, Doppler coefficient, ejected rod worth, boron worth, control rod worth, maximum linear heat rate of the fuel pin at various elevations in the core, and shutdown margin) predicted for the cycle are conservative with respect to the values assumed in the safety analysis of various postulated accidents. If they are not conservative, acceptable reevaluation or reanalysis of applicable accidents is performed, or the core is redesigned.
4. The power distributions within the reactor core during normal operation as permitted by Technical Specifications will not lead to exceeding the thermal design criteria of the fuel or exceeding initial condition LOCA peaking factors during postulated Condition I and II type transients. For transients where fuel failures may occur, the number of fuel failures will be less than applicable dose limits.
5. Fuel management will produce fuel rod power and burnup consistent with the mechanical integrity analysis of the fuel rod.

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

The nuclear design calculations described in this report are covered by the Duke Quality Assurance program (Reference 1).

## **1.2 Definition of Terms**

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

<del>a/o</del>	<del>atom percent</del>
ARI	all rods in
ARO	all rods out

axial offset (AO)	$\frac{P_T - P_B}{P_T + P_B}$ , where $P_T$ is the integrated power in the top half of the core, and $P_B$ is the integrated power in the bottom half of the core
<b>AFD</b>	<b>flux difference between the top and bottom halves of the core</b>
$\beta_i$	delayed neutron fraction for group i
$\beta_{\text{eff}}$	effective delayed neutron fraction in core
<b>BOLC</b>	<b>beginning of life cycle</b>
BP	burnable poison
<b>BTU</b>	<b>British thermal unit</b>
BU	fuel burnup
$C_B$	chemical shim boron concentration in the coolant
CZP	cold zero power
COLR	core operating limits report
<b>DRWM</b>	<b>dynamic rod worth measurement technique</b>
<b>EOLC</b>	<b>end of life cycle</b>
<del>EQXE</del>	<del>equilibrium xenon condition</del>
<b>FFCD</b>	<b>final fuel cycle design</b>
GWD/MTU	Gigawatt days per metric ton of initial uranium metal, 1 GWD/MTU is 1000 MWD/MTU
HFP	hot full power
HZP	hot zero power
<del>I</del>	<del>delayed neutron importance factor</del>
IFBA	<b>zirconium diboride</b> integral fuel burnable absorber

<b>ITC</b>	<b>isothermal temperature coefficient</b>
$\Delta I$	<del>flux difference between the top and bottom halves of the core; in this report, <math>\Delta I</math> is a calculated value, rather than the difference between measured signals from the excore detectors</del>
$K(z)$	<del>The <math>F_Q</math> limit assumed in the LOCA analysis normalized to the maximum value allowed at any core height</del>
<b>KW</b>	<b>kilowatt</b>
<b>LOCA</b>	<b>loss of coolant accident</b>
<b>LOFA</b>	<b>loss of flow accident</b>
$l^*$	prompt neutron lifetime
<b>MTC</b>	<b>moderator temperature coefficient</b>
<b>MO<del>L</del>C</b>	middle of life <b>cycle</b>
<b>MWD/MTU</b>	<del>measure of energy extracted per unit weight of initial uranium metal fuel; is equal to 1 megawatt times 1 day, divided by 1 metric ton of uranium</del>
<b>OPDT</b>	<b>over-power delta-T trip function</b>
<b>OTDT</b>	<b>over-temperature delta-T trip function</b>
pcm	percent mille (a reactivity change that equals $10^{-5} \Delta\rho$ )
ppm	parts per million by weight; which specifies the amount of chemical shim boron present by weight in the main coolant system
radial local	ratio of assembly maximum rod to assembly average x-y power
RCCA	<del>rod cluster control assembly; the type of control rod assembly used in McGuire and Catawba. (All RCCA are full length absorbers for both plants.)</del>
<b>RPS</b>	<b>reactor protection system</b>
$\rho$	reactivity

$\Delta\rho$	$\frac{K_1 - K_2}{K_1 \times K_2}$	where $K_1$ and $K_2$ are eigenvalues obtained from two calculations where only one parameter was varied
<b>SDM</b>	shutdown margin – the amount of negative reactivity ( $\rho$ ) by which a reactor core is maintained in a HZP subcritical condition after a control rod trip	
step	unit of control rod travel equal to 0.625 inch	
SWD	steps withdrawn	
TMOD	moderator temperature; defined as the temperature corresponding to the average water enthalpy of the core	
$T_{\text{res}}$	<del>resonance temperature of the fuel</del>	
UFSAR	updated final safety analysis report	
w/o	<del>weight percent</del>	

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

<u>Power density</u>	thermal power produced per unit volume of the core (KW/liter <b>or KW/cm<sup>3</sup></b> ) <b>or relative to the fuel volume</b>
<u>Linear Power Density</u>	thermal power produced per unit length of active fuel (KW/ft)
<u>Average Linear Power Density</u>	total thermal power produced in the core divided by the total active fuel length of all fuel rods in the core
<u>Local Heat Flux</u>	local heat flux on the cladding surface (BTU/ft <sup>2</sup> /hr)
<u>Rod Power or Integral Power</u>	is the length integrated linear power density in one rod (KW)

Various hot channel factors are:



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

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

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

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## **2. FUEL DESCRIPTION**

The ~~reactor cores for McGuire and Catawba~~ **reactors are based on Westinghouse's four-loop pressurized water reactor (PWR) design** containing 193 fuel assemblies. Each fuel assembly consists of 264 fuel rods, 24 guide thimble tubes and 1 instrumentation thimble tube assembled in a square 17x17 lattice. **The Harris and Robinson reactors are based on Westinghouse's three-loop PWR design containing 157 fuel assemblies. The fuel assembly design for the Harris reactor consists of the same 17x17 lattice design as used at McGuire and Catawba, while the fuel assembly design for the Robinson reactor consists of 204 fuel rods, 20 guide thimble tubes and 1 instrumentation thimble tube assembled in a 15x15 square lattice.** The assembly structure **for all fuel assembly designs** consists of top and bottom nozzles and grid assemblies positioned axially along the fuel assembly. Each fuel rod contains a column of stacked fuel pellets.

Detailed design data for the fuel pellets, fuel rods, fuel assembly, and reactivity control components are provided to Duke by the fuel vendor.

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### 3. NUCLEAR CODE SYSTEM

#### 3.1 Introduction

Nuclear design calculations performed for Westinghouse reactors are performed using NRC-approved models. **Either t**The CASMO-4/SIMULATE-3 **or CASMO-5/SIMULATE-3** code system is used to model the reactor core and to perform design calculations **as applicable**.

Presented in this section is a description of the sequence, cross section preparation and parameterization, and reload design modeling procedures. An overview of the CASMO-4/SIMULATE-3 **and CASMO-5/SIMULATE-3** code sequence is outlined in Figure 3-1. Additional details are contained in References 4 **and 14**.

#### 3.2 Sources of Input Data

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

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

Plant data, fuel and component specifications, supplemented by vendor reports and open literature, is the primary source of data for Items 1 to 3. The fuel temperature data used within SIMULATE-3 is developed from data derived from the fuel rod thermal model within SIMULATE-3K. These data are used as input to the cross section generators and core simulators.

#### 3.3 Cross Section Preparation

In order to model the neutronics of a reload core, it is necessary to generate cross sections and nuclear data (e.g. kinetics data, discontinuity factors, detector constants, etc.) for use in the core simulator.

**Either CASMO-4 or CASMO-5** is used for this purpose.

The CASMO-4 **and CASMO-5** codes ~~is~~ **contain** a multi-group, two-dimensional transport theory model for burnup calculations on fuel assemblies or fuel pin cells. The codes accommodate a geometry consisting of cylindrical rods of varying composition in a square pitch array. They ~~are CASMO-4 code~~ **are** used to model fuel pins, burnable absorbers (integral and discrete), control rods, guide tubes, incore instruments, water gaps, and reflectors.

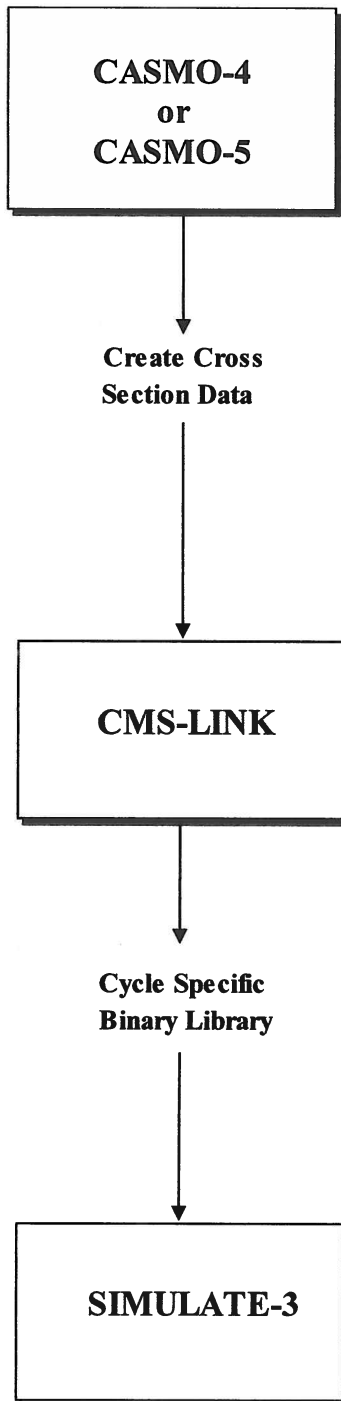
A series of ~~CASMO-4~~ **CASMO** cases ~~is~~ **are** executed for each unique fuel assembly lattice configuration. A typical case set characterizes the effect of fuel burnup, moderator temperature, fuel temperature, soluble boron concentration and control rod presence. For core reflector regions, the impact of changes in moderator temperature and soluble boron concentration are typically modeled. Refer to References 4 **and 14** for additional details.

The data from ~~CASMO-4~~ **CASMO** is next processed by the ~~CMSLINK~~ **CMS-LINK** code to produce a nuclear data library for input to the SIMULATE-3 core model. The data collected is processed and stored in multi-dimensional tables that characterize the effect of both instantaneous and integrated perturbations to local core conditions. The precise functionalization of the data varies depending on the type of data and the amount that a given data type changes as core conditions change. Refer to References 4 **and 14** for additional details.

### **3.4 SIMULATE-3 Model**

SIMULATE-3 is a three-dimensional diffusion theory reactor core simulator. The code calculates core wide power distributions and fuel depletion with macroscopic cross sections in two energy groups. The nodal solution is performed on a geometric mesh of either one or four nodes per assembly in the radial plane, and an appropriate axial mesh in the active fuel column. Explicit models of the top, bottom and radial reflector regions allow for an analytic solution for the flux and leakage at the core boundary. Pin power distributions are constructed by synthesizing results from the nodal mesh solution with heterogeneous lattice solutions extracted from ~~CASMO-4~~ **CASMO**. A more comprehensive description of the CASMO-4/SIMULATE-3 **and CASMO-5/SIMULATE-3** models, and the benchmark calculations performed to validate this software package can be found in References 4 **and 14, respectively**.

**Figure 3-1**  
**CASMO/SIMULATE Code Sequence**



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## **4. FUEL CYCLE DESIGN**

### **4.1 Preliminary Fuel Cycle Design – Initialization**

To commence the design of a reload, core operation requirements along with planned plant (primary or secondary) changes are assembled. A preliminary loading pattern is designed which meets operational requirements, while considering economic, operational flexibility and margins to Core Operating Limits report (COLR) and Updated Final Safety Analysis Report (UFSAR) Chapter 15 analysis limits. Physics and power distribution data from the preliminary design are compared against COLR values, UFSAR Chapter 15 licensing basis values and core operating requirements to determine the adequacy of the reload design and to verify conformance to existing limits.

#### **4.1.1 Review of Design Information**

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

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

#### **4.1.2 Determination of Cycle-Specific Operating Requirements**

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

#### **4.1.3 Preliminary Loading Pattern and Reload Region Determination**

The purpose of a preliminary loading pattern analysis is to determine the uranium and separative work requirements to meet a desired cycle lifetime, and to confirm that acceptable margins exist to COLR limits, and to key safety analysis assumptions and limits. The cycle lifetime is confirmed by modeling the depletion of a reload core. If the number of new fuel assemblies and the enrichment are known, this



analysis will yield an estimate of the cycle lifetime.

## **4.2 Final Fuel Cycle Design**

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

1.  $F_{\Delta H}$  must meet the limits specified in the Core Operating Limits Report (COLR).
2. Moderator Temperature Coefficient must meet the limits specified in the COLR.
3. Maximum fuel burnup must be less than the limits applicable for the type of fuel being used.
4. Shutdown Margin must meet limits specified in the COLR.
5. Maximum linear rod power must meet the limit specified in the COLR.
6. UFSAR Chapter 15 related physics parameters must be validated and the acceptability of thermal limits confirmed.

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

### **4.2.1 Fuel Shuffle Optimization and Cycle Depletion**

The preliminary fuel shuffle scheme is modified to minimize power peaking with consideration of economic, operational flexibility and margins to COLR and UFSAR Chapter 15 analysis limits. This is accomplished by a trial and error type search until an acceptable loading pattern is developed.

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

The cycle is then depleted to various times in the cycle to verify that power peaking versus burnup remains acceptable. The shuffling variations include rearranging the location of the burned or fresh fuel assemblies, BP placement, and rotation (**via cross quadrant shuffling**) of **burned or** the spent fuel assemblies **re-inserted from the spent fuel pool**. These calculations are typically performed assuming

quarter core symmetry.

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

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

#### **4.2.2 Rod Worth Calculations**

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

~~The McGuire, and Catawba,~~ **Harris and Robinson reactors** are typically operated in the **all rods out (ARO)** or feed-and-bleed mode. All rod cluster control assemblies (RCCAs) have full length absorber rods. RCCA bank locations ~~in McGuire and Catawba~~ usually are fixed and do not change from cycle to cycle. During full power operation, Control Bank D is typically inserted **a small amount** ~~about six inches (215 steps withdrawn)~~ in the active core **to provide immediate reactivity and axial offset control. Typical bite positions are around 215 steps withdrawn (SWD).** Control Bank D is used to control power during load follow maneuvers, and in conjunction with **other** Control Banks ~~B and C~~, to achieve criticality during startup. Control Banks can also be used to offset reactivity changes produced from fuel depletion and changes in boron concentration, xenon concentration, and moderator temperature.

Control and Shutdown Bank integral and differential rod worths are sensitive to local and global power distribution changes. Since the placement of fresh and depleted fuel assemblies produces unique power distributions, it is necessary to analyze Control and Shutdown Bank rod worths and individual rod worths (e.g. dropped and ejected rod worths) for each reload core. Rod worth related calculations that are evaluated for each reload core are:

- Shutdown margin

- Trip reactivity
- Control rod insertion limits
- Maximum differential and total rod worths at power
- Maximum differential and total rod worths from subcritical
- Dropped rod worth
- Ejected rod worth

Each of the above calculations is discussed in Chapter 5. The calculation of the above parameters is performed for each reload core to confirm that physics parameters assumed in the UFSAR Chapter 15 accident analysis are bounding. In addition, some of the above parameters are also calculated to support startup and operation of the reload core and to define core operating limits.

#### **4.2.3 Fuel Burnup Calculations**

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

#### **4.2.4 Reactivity Coefficients and Defects**

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

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

#### **4.2.5 Boron Related Parameters**

Critical boron concentrations for various core conditions during cycle lifetime are calculated using the core model. They are calculated as a function of reactor power, moderator temperature and control rod position to support confirmation of input assumptions made in the Chapter 15 accident analysis, and to support the startup and operation of the reload core. Differential boron worths are also calculated as function of various combinations of the above variables.

#### **4.2.6 Xenon Worth**

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

#### **4.2.7 Kinetics Parameters**

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

SIMULATE-3 is used to calculate the core averaged kinetics parameters of interest based on delayed neutron data from ~~CASMO-4~~ **CASMO**.

Calculations are performed at various times during core life. The sum of the six group  $\beta_i$  effective,  $\beta_{\text{effective}}$ , for the new reload cycle is compared to those values assumed in the safety analysis.

#### **4.2.8      Assessment of the Fuel Cycle Design**

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

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

**TABLE 4-1**

**Typical Nuclear Design Data**  
**For Reload Design**

1. Power operation mode: load follow or base load.
  - **Coastdown mode (power or Tave) and duration**
2. Vessel internal or core component modifications.
3. Expected minimum and maximum cycle burnups.
4. Feed enrichment (if already contracted for).
5. Number and design of feed assemblies.
6. RCS hydraulic conditions.
7. Analysis Limits. Includes:
  - Core Operating Limits Report Limits
  - Safety Analysis UFSAR Chapter 15 accident analysis assumptions and limits  
(Contained in the reload design safety analysis review checklist (REDSAR))
  - Fuel Mechanical limits
  - Thermal-Hydraulic limits

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## **5. NODAL ANALYSIS METHODOLOGY**

### **5.1 Purpose and Introduction**

Nodal analysis allows for modeling of the reactor core in three-dimensions and for the calculation of core physics parameters and power distributions. The calculation of these parameters is used to confirm the acceptability of UFSAR Chapter 15 accident analysis assumptions (Chapter 6), develop core operating limits (Chapter 7), and to calculate parameters to support the startup and operation of reload core (Chapters 9 and 10). The SIMULATE-3 code (Ref: [References 4 and 14](#)) is used to perform the required analyses.

This section addresses the role of a nodal code in performing cycle depletions, and calculating the following types of reload design information.

- Differential Rod Worths (Section 5.3.1)
- Integral Rod Worths (Section 5.3.2)
- Shutdown Margin (Section 5.4.1)
- Shutdown Boron Concentrations (Section 5.4.2)
- Rod Insertion Limits (Section 5.5)
- Minimum Trip Reactivity (Section 5.6.1)
- Trip Reactivity Shape (Section 5.6.2)
- Rod Ejection Analysis Parameters (Section 5.7)
- Dropped Rod Analysis Parameters (Section 5.8)
- Doppler Temperature Coefficient (Section 5.9)
- Moderator Temperature Coefficient (Section 5.10)
- Isothermal Temperature Coefficient (Section 5.11)
- Power Coefficient and Defect (Section 5.12)
- Miscellaneous Coefficients (Section 5.13)
- Critical Boron Concentrations and Boron Worth (Section 5.14)
- Xenon and Samarium Worths (Section 5.15)

The above calculations are performed in either quarter-core or full-core geometry.



## 5.2 Fuel Cycle Depletion – Nodal Code

A fuel cycle depletion is performed for each initial or reload cycle. This depletion is typically performed in the critical boron search mode, with nominal rod insertion (usually 215 SWD) and equilibrium xenon.

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

1. Two and three-dimensional power distributions at each burnup step.
2. A boron letdown curve, i.e., critical boron concentrations as a function of burnup.
3. Axially-dependent parameters such as offset or axial flux difference as a function of burnup.
4. Assembly exposures as a function of core-averaged burnup.

## 5.3 Rod Worth Analysis

A three-dimensional core simulator is used to calculate various rod worths which require three-dimensional capabilities. These calculations include differential rod worths, integral rod worths, total rod worths, stuck rod worths, ejected rod worths and dropped rod worths.

### 5.3.1 Differential Rod Worth Analysis

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

Differential rod worths for the control banks are typically calculated at HZP and HFP, at **beginning of cycle (BOC)** and **end of cycle (EOC)**, and at no xenon, equilibrium xenon, and peak xenon conditions. The rod banks are inserted both sequentially and in 50% overlap. Differential rod worths are also calculated with banks in normal overlap as function of time in life, power, and assuming adverse axial power distributions while adhering to the power dependent rod insertion limits. Calculations are also performed at HZP with combinations of two sequential Control Banks moving in 100% overlap.

### 5.3.2 Integral Rod Worth Analysis

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

Integral rod worth calculations for the control banks are typically performed at HZP and HFP, at BOC and EOC, and at no xenon, equilibrium xenon, and peak xenon conditions. The rod banks are inserted sequentially and with 50% overlap. Integral rod worths are also calculated with banks in normal overlap as function of time in life, power, and assuming adverse axial power distributions while adhering to the power dependent rod insertion limits. Calculations are also performed at HZP with combinations of two sequential Control Banks moving in 100% overlap. The total rod worth (**all rods inserted (ARI) worth**) is calculated at BOC, EOC, and any limiting burnup at HZP for use in the shutdown margin calculation.

## 5.4 Shutdown Margin Analysis

### 5.4.1 Shutdown Margin

The first step in performing the shutdown margin (SDM) calculation involves searching for the highest worth stuck rod at BOC, EOC, or any limiting burnup for HZP conditions using full core calculations.

The next steps of the SDM calculation consist of calculating:

1. The total rod worth (ARI) at HZP, BOC, and EOC. This worth is determined by running cases at ARO and ARI (with constant boron and xenon) and subtracting the reactivities.
2. The maximum stuck rod worth at HZP, BOC, and EOC. Utilizing full-core capabilities, the worth of the worst stuck rod is determined by subtracting the reactivities between two cases, one with ARI, the other with ARI and the stuck rod out.

3. The power defect from HFP to HZP, at BOC and EOC. This defect is determined by running cases at HFP and HZP (with constant boron and xenon) and subtracting the reactivities. This reactivity insertion accounts for Doppler and moderator defects, and for axial flux redistribution.
4. The maximum allowable inserted rod worth at HFP, BOC, and EOC. This worth is explicitly calculated or obtained by reading the integral rod worth curve at the rod insertion limits (See Section 5.3.2).

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

The required rod worth is calculated next in Steps 6-9. The power defect, Item 6 in Table 5-1, accounts for Doppler and moderator defects, and for axial redistribution. The maximum allowable inserted rod worth, Item 7, is obtained from the allowable rod insertion and the integral rod worth curve for that insertion. This accounts for the maximum allowed rod insertion at HFP. An axial flux redistribution occurs when the power level is reduced from HFP to HZP. This redistribution causes an increase in reactivity **and is accounted for when**. ~~If Item 6 is calculated using a 3-D model. Since SDM calculations are performed using a 3-D model,~~ no additional penalty is required. ~~If Item 6 was calculated using a 2-D model, where redistribution effects are not modeled, an additional reactivity penalty is assessed as Item 8.~~ The sum of these required worths (Item 9) is the total required worth.

Additional reactivity penalties are applied to both the power defect and the rod insertion allowance to account for xenon redistribution effects if the calculation was performed at equilibrium xenon conditions. **These penalties are included in Item 8 in the example provided in Table 5-1.**

The shutdown margin is shown as Item 10 and is defined as the total available worth minus the total required worth. Shutdown margin requirements are specified in the Core Operating Limits Report.

#### 5.4.2 Shutdown Boron Concentration

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

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

An alternate approach for calculating a shutdown boron concentration is to calculate the ARI boron concentration corresponding to the appropriate **Technical Specification required** shutdown margin (~~1.3%  $\Delta\rho$  or 1.0%  $\Delta\rho$~~ ) and then adjusting this boron concentration by an equivalent boron concentration using a boron worth to account for the highest worth stuck rod, and 10% of the ARI-stuck rod out worth. The resultant boron concentration is increased by at least an additional 100 ppm.

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

#### 5.5 Rod Insertion Limit Assessment

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

The control rod insertion limits are determined based on:

1. Maintaining the required minimum shutdown margin, as specified in the COLR, throughout the cycle life.
2. Maintaining the maximum calculated power peaking factors within the limit specified in the COLR.
3. The acceptability of the UFSAR Chapter 15 accident analyses.

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

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

The insertion limits also have to satisfy the peaking factor constraints. Power peaking values calculated by the three-dimensional nodal code are compared to the peaking limits specified in the COLR. If the peaking limits are not satisfied, the control rod insertion limits are adjusted until satisfactory power peaking values are obtained, or the core is redesigned.

In addition to satisfying reactivity and peaking factor constraints during normal operation, the control rod insertion limits may need to be modified based on the results from the evaluation of UFSAR Chapter 15 accidents. Evaluations are performed with the nodal code and are compared to the design criteria associated with each event. The acceptability of the control rod insertion limits is dependent on the criteria being satisfied.

## **5.6 Trip Reactivity Analysis**

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

### **5.6.1 Minimum Trip Reactivity**

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

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

### **5.6.2 Trip Reactivity Shape**

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

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

## **5.7 Rod Ejection Analysis**

The UFSAR Section 15.4.8 (References 2, ~~and 3~~, **15 and 16**) presents the limiting criteria for the ejected rod accident. The methodology used to evaluate the rod ejection accident (REA) **must be reviewed and approved by the NRC. For example, the REA methodology used for McGuire and Catawba** is described in References 10.

Cycle-specific analyses are performed to verify that the ejected rod parameters are within calculational limits. Ejected rod calculations are performed at BOC and EOC or at other limiting times in cycle life at both HFP and HZP.

The HZP ejected rod calculations are performed using full core geometry with Control Banks B and C at their insertion limits ~~in the core~~ and with Control Bank D fully inserted. No allowance for rod mis-positioning is made because the highest worth ejected rod is fully inserted. Single rods in Control Banks D, C, and B **(if inserted)** are removed in subsequent cases and the worth of the ejected rod is calculated by subtracting the reactivities of the cases before and after the rod was removed. The fuel and moderator temperature is held constant and equal to the HZP moderator temperature for these calculations. The highest worth calculated by the above procedure is the worst ejected rod at HZP. If the ejected rod worth exceeds the calculational limit, one of the following is performed: an evaluation, a revision to the rod insertion limits, a reanalysis of the Chapter 15 REA analysis, or a redesign of the core loading pattern.

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

A parallel analysis, addressing core peaking, is performed at the same time as the rod worth analyses. ~~Additional discussion of the rod ejection accident analysis methodology is contained in Reference 10.~~

## **5.8 Dropped Rod Analysis**

The UFSAR (References 2, ~~and 3~~, **15 and 16**) presents the limiting criteria for the dropped rod accident. The calculational limits are established using **an NRC-approved** ~~the~~ methodology. **For example, the NRC-approved dropped rod methodology used for McGuire and Catawba is** described in Reference 10.

A core model is used that evaluates pre-drop and post-drop physics parameters for possible dropped rod combinations. The physics parameters important to the dropped rod analysis are **described in the NRC-approved methodology report for this accident** ~~presented in Reference 10~~. Dropped rod worths are calculated by evaluating the reactivity difference produced from a control rod or rods dropped from the HFP all rods out or rod insertion limit condition with an allowance for rod position uncertainty.

## **5.9 Doppler Temperature Coefficient**

The Doppler Temperature Coefficient is the change in core reactivity produced by a small change in fuel temperature divided by the fuel temperature change. The fuel temperature change is produced by varying the fuel temperature uniformly at each core location or node, or by introducing a power level change. Doppler temperature coefficients calculated based on a power level change are known as power Doppler temperature coefficients because the reactivity change from the power level change can be equated against either a fuel temperature change or a power level change. If the reactivity change is equated against a power level change, the coefficient is referred to as a Doppler-only power coefficient.

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

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



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

Doppler Temperature Coefficients calculated by a power level change are calculated using the same equation except the perturbed condition is produced from a power level change with the core moderator temperature and fission products held at their reference values.

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

### 5.10 Moderator Temperature Coefficient

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

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

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

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

Alternately, the moderator temperature coefficient can be calculated by subtracting the Doppler temperature coefficient from the isothermal temperature coefficient.

### 5.11 Isothermal Temperature Coefficient

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

### 5.12 Power Coefficient and Power Defect

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

The power coefficient is defined by the following equation:

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

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

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

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

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

where:  $\alpha_{TMOD}$  is the moderator temperature coefficient and  $\alpha_D$  is the Doppler temperature coefficient.

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

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

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

If the second case is run with the core power level reduced while holding control rods, boron, xenon and moderator temperature constant, the difference in reactivity between the initial and perturbed case divided by the power level change produces a Doppler-only power coefficient.

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

Both HFP and the HZP cases should have equivalent xenon concentrations. The power defect is calculated from the following equation:

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

where:  $K_{eff}^1$  is K-effective at HZP and  
 $K_{eff}^2$  is K-effective at HFP

### **5.13 Miscellaneous Coefficients**

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

### **5.14 Critical Boron Concentrations and Boron Worths**

Critical boron concentrations and boron worths are calculated as a function of burnup, reactor power, moderator temperature, control rod position and xenon condition to support the startup and operation of the reload core and confirm the acceptability of UFSAR Chapter 15 accident analysis assumptions. Critical boron concentrations are calculated at different times in the fuel cycle with the core simulator by performing critical boron search calculations.

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

### **5.15 Xenon and Samarium Worths**

Xenon and samarium fission product worths are calculated as a function of cycle burnup. The nominal HFP depletion cases with equilibrium xenon are used as input to a second set of cases where the xenon concentration is set to zero (or the xenon cross sections are set to zero). The difference in reactivity between the equilibrium xenon and no xenon cases equals the equilibrium xenon worth at HFP. A similar procedure is used to calculate samarium worth.

Xenon worths are also calculated as a function of power level. Worths calculated in this manner are usually referred to as the equilibrium xenon reactivity defect and are quoted in either pcm or  $\% \Delta \rho$ .

### **5.16 Assessment of Nodal Analyses**

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

**TABLE 5-1**  
**Example Shutdown Margin Calculation**

<i>Available Rod Worth</i>		<u>BOC, % <math>\Delta\rho</math></u>
1.	Total rod worth, HZP	<del>6.46</del> <b>6.14</b>
2.	Maximum stuck rod, HZP	<del>-1.39</del> <b>-0.80</b>
3.	Net Worth	<del>5.07</del> <b>5.34</b>
4.	Less 10% uncertainty	<del>.51</del> <b>.54<sup>+</sup></b>
5.	Total available worth	<del>4.56</del> <b>4.80</b>
<i>Required Rod Worth</i>		
6.	Power defect, HFP to HZP	<del>.88</del> <b>1.53</b>
7.	Max allowable inserted rod worth	<del>1.36</del> <b>.17</b>
8.	Flux/Xenon redistribution	<del>.63</del> <b>.27</b>
9.	Total required worth	<del>1.87</del> <b>1.97</b>
10.	Shutdown Margin (total avail. worth minus total required worth)	<del>2.69</del> <b>2.83</b>

**+ rounded up**

NOTE: Required shutdown margin is specified in the COLR.

**TABLE 5-2**  
**Example BOC Trip Reactivity Calculation**

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

## **6. CALCULATION OF SAFETY RELATED PHYSICS PARAMETERS**

### **6.1 Safety Analysis Physics Parameters**

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

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

Each of the above has its own subset of specific parameters. These core physics parameters are considered in the UFSAR Chapter 15 safety analyses. The core physics parameters whose values have an important influence on the course or the consequences of the accidents analyzed in the UFSAR are designated as safety analysis physics parameters. ~~Reference 10 identifies these~~ **Key safety analysis** parameters, ~~describes~~ the approach for calculating the values of these parameters, and ~~discusses~~ the manner in which the reload values are evaluated for acceptability with respect to the accident analysis assumptions **are described in a NRC-approved methodology report. For example, Reference 10 describes the NRC-approved safety analysis physics parameter methodology for McGuire and Catawba.**

### **6.2 Core Power Distributions**

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

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

The methodology for performing these analyses **are described in NRC-approved methodology reports (e.g. References 9 and 10)** ~~is presented in Reference 9.~~



**Initial condition power distributions are those power distributions permitted by Technical Specification axial flux difference (AFD) and rod insertion limits. These core power distributions are a valid initial condition for the initiation of UFSAR Chapter 15 transients and accidents.**

**Core power distributions generated for transients that produce high peaking factors and/or high power levels are compared against fuel thermal limits to confirm acceptability of accident analysis acceptance criteria and reactor protection limits. For the accidents where thermal limits are allowed to be exceeded, the dose consequences are confirmed to be within acceptable regulatory limits.**

**Acceptance criteria for each accident are identified in plant specific UFSARs. Accidents for which either the thermal limit acceptance criteria or Technical Specification limits are exceeded are evaluated or reanalyzed to ensure acceptable accident consequences, or the core is redesigned to obtain acceptable peaking factors and accident consequences.**

### **6.3 Power Peaking Factors and Reliability Factors**

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

The general formulation for the peaking reliability factor is:

$$\text{Peaking Reliability Factor} = 1 + \text{Bias} + \sqrt{UC^2 + U_{x1}^2 + U_{x2}^2 + \dots} \quad (6-1)$$

Where:

UC – Calculation Uncertainty

For the Pin Total Peak ( $F_Q$ ):  $UC^2 = U_{CT}^2 + U_{RL}^2$

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

For the Assembly Axial Peak ( $F_Z$ ):  $UC^2 = U_{CA}^2$

UCT – Assembly Total Peaking Uncertainty

URL – Pin Power Peaking Uncertainty

UCR – Assembly Radial Power Peaking Uncertainty

UCA– Assembly Axial Power Peaking Uncertainty

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

BIAS - Calculation Bias

Reference 4 contains the CASMO-4/SIMULATE-3 assembly and pin power uncertainty factors.

**Reference 14 contains the CASMO-5/SIMULATE-3 assembly and pin power uncertainty factors.**

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## **7. THREE-DIMENSIONAL POWER PEAKING ANALYSIS**

As part of the reload design, detailed analysis of the core power distribution for normal operation and anticipated transient conditions are made. These analyses are performed (1) to confirm that the initial condition power peaking factors for certain accidents remain valid, (2) to verify that certain transient induced power peaks will satisfy the fuel design limits, and (3) to facilitate the selection of operating limits and RPS setpoints. The methods for performing these analyses are outlined in Reference 9.

**Reference 9 contains a description of the Maneuvering Analysis methodology used to generate operating axial flux difference limits and rod insertion limits that limit the magnitude of peaking factors allowed during normal operation, and at the initiation of UFSAR Chapter 15 transient and accident conditions. Axial flux difference and rod insertion limits are also established to ensure the acceptability of peaking assumptions made in the loss of coolant accident (LOCA) and the limiting Condition II transient where peaking does not change as the result of the event (typically the loss of flow accident (LOFA)). Operating the reactor within these limits preserves the power peaking assumptions made in the LOCA and LOFA analyses, and provides assurance that fuel thermal limits and reactor protection setpoints are acceptable. The maneuvering analysis also confirms the acceptability of the  $f(\Delta I)$  reset functions of the over-power delta temperature (OPDT) and over-temperature delta temperature (OTDT) reactor protection system (RPS) trip functions. These trip functions, in combination with other trip functions, provide protection against fuel failure due to fuel melting (CFM) or departure from nucleate boiling (DNB) during anticipated transients. The relevant limits are determined such that the RPS will trip the reactor before fuel damage occurs.**

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## 8. RADIAL LOCAL ANALYSIS

### 8.1 Background

The SIMULATE-3 methodology calculates integrated and local pin powers directly. As a result, radial local factors are not required in this methodology. **Different pin power uncertainty factors have been calculated for CASMO-4 and CASMO-5 based SIMULATE-3 models, and for low enriched uranium fuel (LEU) with and without gadolinium.** These ~~pin power uncertainty factors associated with the SIMULATE-3 methodology was~~ **were** calculated by comparing predicted pin power distributions against pin power measurements from critical experiments **and against those predicted by CASMO.** Refer to References 4 **and 14** for additional details **and specific values for the CASMO-4 and CASMO-5 based SIMULATE-3 pin power uncertainties.**

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## **9. DEVELOPMENT OF CORE PHYSICS PARAMETERS**

Upon completion of the Final Fuel Cycle Design, physics parameters such as boron concentrations and worths, power distributions, etc. are calculated primarily for HFP and some HZP conditions. The purpose of this stage of developing core physics parameters is to provide additional calculations to supplement those already performed. The results of these calculations are used for startup test predictions and core physics parameters throughout the cycle. The information developed is compiled into a Startup and Operation Report. The purpose of this report is to document the predicted behavior of the reactor core as a function of burnup and power level. The report also includes sufficient information to calculate reactivity balance throughout the cycle as required to support reactor startups and confirm shutdown margin. The Startup and Operation Report is intended to be used for operator guidance and to aid the site engineer.

### **9.1 Startup Test Predictions**

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

- Critical boron concentrations and boron worths
- Control rod worths
- Isothermal, Doppler and moderator temperature coefficients
- Core power distributions
- Kinetics data

The method used to calculate each of the above parameters is discussed in Section 5.



## **9.2     Operational Data**

Data is generated for each reload core to support cycle operation. The cycle specific information generated includes predicted boron concentrations and worths, control rod worths, reactivity coefficients and fission product worths. The calculation methodology associated with the calculation of each of these types of parameters is discussed in Section 5.

### **9.2.1   Critical Boron Concentrations and Boron Worths**

Critical boron concentrations and boron worths are calculated as a function of burnup, reactor power, moderator temperature, xenon concentration and control rod position to support the startup and operation of the reload core. Typical calculations include HFP and HZP all rods out (ARO) critical boron concentrations as a function of burnup, Technical Specification shutdown boron concentrations, and boron concentrations to support refueling operations, and mode changes.

### **9.2.2   Xenon and Samarium Worth and Defect**

Xenon and samarium fission product worths are calculated as a function of cycle burnup at HFP conditions to produce HFP equilibrium xenon and samarium worths.

Xenon worth can also be presented as a function of power level. Worths presented in this manner are usually referred to as the equilibrium xenon reactivity defect and are quoted in either pcm or  $\% \Delta \rho$ .

### **9.2.3   Rod Worths**

Rod worths are calculated as a function of burnup, reactor power, moderator temperature, and bank position to support the startup and operation of a reload core. The types of rod worth calculations performed are:

- Individual worths of shutdown and control banks at BOC HZP
- Integral rod worths with control banks in 50% overlap at HFP and HZP conditions as a function of burnup and xenon
- Stuck rod worths at no-load conditions as a function of burnup and moderator temperature

- Total rod worths with the highest worth rod fully withdrawn at no load conditions as a function of burnup and moderator temperature

The above information is used by site engineers in reactivity balance procedures to ensure shutdown margin, to support power maneuvers and reactor startups.

#### **9.2.4 Reactivity Coefficients and Defects**

Reactivity coefficients are calculated as a function of burnup, reactor power, boron concentration, moderator temperature, xenon concentration and control rod position to support the startup and operation of the reload core. The types of reactivity coefficients calculated include isothermal, moderator and Doppler temperature coefficients as a function of power level, burnup, control rod position, and moderator temperature. Power defects are calculated as a function of power level and burnup. Typical calculations performed to support the startup and operation of a reload cycle include:

- HFP and HZP all rods out (ARO) isothermal and moderator temperature coefficients as a function of burnup
- ARO isothermal and moderator temperature coefficients as a function of boron concentration and moderator temperature
- EOC moderator and Doppler temperature coefficients to support the BOC and EOC MTC measurement
- Power defects as a function of power level and burnup

The above information is used by site engineers in reactivity balance procedures to ensure shutdown margin, to support power maneuvers, surveillance testing and reactor startups. The calculation method used to calculate reactivity coefficients and defects is discussed in Section 5.

#### **9.2.5 Power Distribution**

Power distributions, both assembly radial and total peaking factors, are measured at various power levels as identified in the **startup** test procedures for McGuire/Catawba reload startups. Calculations are performed at these power levels, and **at** nominal conditions to provide predicted power distributions for comparison **to verify the acceptability of thermal limits and to demonstrate the core is operating as designed.**

### 9.2.6 Kinetics Parameters

Kinetics parameters are calculated using the methodology as discussed in Section 4.2.7. These parameters include the six group  $\beta_i$  effective and  $\lambda_i$ , total  $\beta$  effective and  $\ell^*$ . Kinetics data are used in the reactivity computer to support startup physics testing at the beginning of each fuel cycle. These kinetics parameters are generated at BOC HZP conditions.

## 10. PHYSICS TEST COMPARISONS

Startup physics testing is described in UFSAR Chapter 14. The startup testing information contained in this section is presented to provide the perspective on the types of comparisons of measured to predicted results that are performed to demonstrate the validity and accuracy of a core simulator. References 4 and 14 presents the benchmarking results for the CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 models, respectively.

### 10.1 Introduction

This section presents measurement and calculational techniques and the key core physics parameters that are compared to validate a core simulator. The physics parameters include hot zero power (HZP) and hot full power (HFP) critical boron concentrations, HZP control rod worths, and HZP isothermal temperature coefficients.

The measured data is obtained from previously operated fuel cycles. The measurement techniques discussed are those used at the station. The HZP measurements are taken at beginning-of-cycle (BOC) during the Zero Power Physics Testing. The HFP boron concentration measurements are taken at various time steps throughout the cycles. Calculations are performed with a core simulator.

Comparisons between calculated and measured results are performed to develop the means of the differences between the measured and calculated data, and the corresponding standard deviations. The mean and standard deviation are defined as follows:

$$Mean = \bar{x} = \frac{\sum x_i}{n}$$

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

where:  $x_i$  = value for the  $i^{th}$  observation

$n$  = number of observations.

## 10.2 Critical Boron Concentrations

### 10.2.1 Measurement Technique

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

The measurement uncertainty for critical boron concentrations is due to (1) error in the titration method, and (2) error due to differences between the sample concentration and the core average concentration, and (3) B-10 depletion effects.

### 10.2.2 Calculational Technique

Critical boron concentrations are calculated at HZP and HFP using the core simulator in the boron search mode. Comparisons between calculated and measured boron concentrations are performed to determine the mean and standard deviations of the difference between measured and calculated boron concentrations.

## 10.3 Control Rod Worth

### 10.3.1 Measurement Techniques

Individual control rod bank worths are measured by the boron swap (**boron dilution**), rod swap or dynamic rod worth techniques at BOC HZP conditions. The boron swap technique involves a continuous decrease in boron concentration together with an insertion of the control rods in small, discrete steps. The change in reactivity due to each insertion is determined from reactivity computer readings before and after the insertion. The worth of each rod bank is the sum of all the reactivity changes for that bank. Measured bank worths in ppmB can be determined independent of the reactivity computer by using the measured boron endpoints.

The rod swap technique measures individual control **and shutdown** rod bank worths. The control **or shutdown** rod bank with the highest predicted worth is measured with a boron dilution technique. This technique compensates for a continuous decrease in boron concentration by inserting the control rod bank in

small, discrete steps. The change in reactivity due to each insertion is determined from reactivity computer readings before and after the insertion. ~~These differential rod worths were integrated to define a reference bank worth versus bank insertion. Other individual control rod banks are next inserted without changes in boron concentration by offsetting their worth by removal of the reference rod bank. The amount of reference bank withdrawal is used to infer the worth of other individual control rod banks.~~ **Reference bank differential rod worths determined from the reactivity computer before and after each bank insertion are integrated to define a Reference Bank Worth curve as a function of bank position. For the initial test bank measurement, the test bank is inserted and its worth is offset by withdrawing the Reference Bank. For the remaining test bank measurements, the next test bank is exchanged with the previously inserted test bank, and then the Reference Bank. A critical configuration is established with the test bank fully inserted after each measurement. The worth of individual test banks is inferred from the amount the Reference Bank is withdrawn compensating for any changes in reactor conditions during the test.** A more detailed discussion of the rod swap measurement technique is contained in Reference 11.

The dynamic rod worth measurement (DRWM) technique is a relatively fast method to measure individual control rod bank worths by inserting and withdrawing the bank at the maximum stepping speed without changing boron concentration. Excore detector signals are processed by a reactivity computer with appropriate analytical compensation for significant space-time (dynamic) effects that occur during control rod insertion. A more detailed discussion of the DRWM technique is provided in References 12 and 13.

### **10.3.2 Calculational Techniques**

Individual and total control rod worth calculations are performed with the simulator. For the boron swap method control rod worths are calculated in 100% overlap. For the rod swap and DRWM techniques, individual control rod worths are calculated. Comparisons between the calculated and measured rod worths are performed to determine the mean, and standard deviations of the percent difference between measured and calculated individual control rod worths, and total rod worths.

## **10.4 Isothermal Temperature Coefficients**

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

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

#### 10.4.1 Measurement Techniques

The isothermal temperature coefficient (ITC) is measured starting with an equilibrium boron concentration by changing the average reactor coolant moderator temperature and measuring the change in reactivity with a reactivity computer. The ITC is determined from the slope of the reactivity as a function of temperature **or by dividing the reactivity change calculated by the reactivity computer by the corresponding change in reactor coolant moderator temperature.** The measurement is repeated with a reactor coolant temperature change in the opposite direction and the resultant ITC's averaged. UFSAR Chapter 14 has additional details about ~~the~~ **each** measurement technique.

#### 10.4.2 Calculational Technique

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

A comparison of calculated and measured isothermal temperature coefficients is performed to determine the mean and standard deviation of the calculated to measured difference.

## **11. POWER DISTRIBUTION COMPARISONS**

This section summarizes the power distribution comparisons and statistical analyses that are performed to determine the predicted-to-measured performance of a core simulator, and to develop power distribution uncertainty factors.

### **11.1 Introduction**

Core power distributions are measured at regular intervals during the operation of each fuel cycle. A three-dimensional core simulator is used to develop predicted power distributions at measured conditions. Comparisons between predicted and measured powers are used to define the error in the predicted value for each power distribution measurement. The results are statistically combined to derive 95/95 Observed Nuclear Reliability Factors (ONRF's). ONRF's are calculated for assembly radial powers, assembly peak axial powers, and assembly normalized axial powers. The assembly radial power ( $F_{\Delta H}$ ) is defined as the ratio of assembly average power to core average power. The assembly peak axial power ( $F_Q$ ) is defined as the maximum assembly x-y planar average power along the fuel assembly length relative to the core average power. The assembly axial power ( $F_Z$ ) is defined as the ratio of the assembly peak axial power and the assembly radial power ( $F_Q/F_{\Delta H}$ ). The process used to develop measured and predicted power distribution data is described in subsequent sections.

### **11.2 Measured Power Distribution Data**

#### **11.2.1 Measured Assembly Power Data**

Three dimensional measured assembly power distributions are determined from processing of signals from the incore detector system.

#### **11.2.2 Measurement System Description**

The incore detector systems at McGuire and Catawba consist of six movable miniature fission chamber neutron detectors. **The incore system at Harris and Robinson consists of five 5 movable fission chambers.** The detectors are inserted into the bottom of the reactor vessel and driven up through the core



to the top. They are then slowly withdrawn through the core. Incore flux maps are obtained by taking voltage signal readings from the detectors as they are withdrawn through the core. These data are then stored on the plant computer.

The detectors travel inside thimbles that are located in the Instrument Guide Tube of the fuel assemblies.

**McGuire and Catawba have 58 instrumented assemblies out of a total of 193 fuel assemblies, while Robinson and Harris have 49 and 50 instrumented core locations distributed among 157 fuel assemblies, respectively. Robinson has one less instrumented core location because the original instrument at core location L-05 was converted to a reactor vessel level instrument.** There are more than 600 individual voltage signals recorded available axially along each of instrumented fuel assemblies. The instrumented **McGuire/Catawba, Harris and Robinson fuel assemblies instrumented core locations** are shown in Figures 11-1, 11-2 and 11-3.

Raw measured signals are processed to remove clearly spurious information and any data taken above or below the active fuel column. The remaining information is normalized to account for differences in individual fission chamber performance and changes in reactor power level that may have occurred while the data was taken. The normalized signals are converted to normalized relative power by applying signal to power conversion factors that are derived from cycle specific core models. These conversion factors are dependent upon core location, burnup and control rod presence.

The final product is a full core, assembly mesh, three-dimensional measured relative power distribution. These data are used to calculate three types of power peaking factors which characterize important radial and axial properties of the measured power distribution. Assembly  $F_{\Delta H}$  or assembly radial power is simply the average relative power in each fuel assembly. Assembly  $F_Q$  or assembly maximum power is the largest relative power in each assembly. Assembly  $F_z$  or assembly axial power is the assembly  $F_Q$  normalized to the assembly average power ( $F_z = F_Q/F_{\Delta H}$ ) for each assembly. Measured assembly  $F_{\Delta H}$ ,  $F_Q$  and  $F_z$  may be compared directly to equivalent edits generated by the core simulator. All power measurements are taken at approximately equilibrium xenon conditions.

### 11.3 Predicted Power Distribution Data

#### 11.3.1 Fuel Cycle Simulations

Core predicted power distributions are developed using a three-dimensional core model. Fuel cycle depletions are performed to simulate the operation of the cycle being evaluated, and to develop power distributions at the reactor conditions of the power distribution measurement. The predicted power distribution data is compared against equivalent  $F_{\Delta H}$ ,  $F_Q$  and  $F_z$  measured power distribution data to determine calculated minus measured power differences or relative differences. This information is used to develop one-sided upper tolerance limit uncertainties ~~with a 95% confidence level that 95% of the predicted powers are equal to or larger than the measured value.~~ **The relative difference between predicted and measured power is:**

$$\text{Relative Difference} = \frac{(P - M)}{M} * 100$$

### 11.4 Statistical Analysis

**Predicted and measured power distribution data is used to develop** ~~One-sided upper tolerance limit uncertainties are developed~~ with a 95% confidence level that 95% of the predicted powers are equal to or larger than the measured value. This statistical method requires that the predicted to measured data set pass a test for normality which is performed at a 1% level of significance. If a given data set fails this normality test a conservatively large uncertainty is determined by a non-parametric evaluation of the data. These statistical methods are described in References 5 through 8.

Observed nuclear reliability factors (ONRFs), or assembly uncertainty factors, for  $F_{\Delta H}$ ,  $F_Q$  and  $F_z$  are calculated using the following expression:

$$ONRF = 1 - bias + K_a \sigma_a$$

The bias term in this equation is defined as the mean relative error in the calculated peaking factor for all power distribution measurements excluding core locations with a normalized measured power less than or equal to 1.0. The bias term can be expressed algebraically as:

$$Bias = \frac{1}{n} \sum_i^n \frac{(P_i - M_i)}{M_i}$$

where,  $P_i$  = the  $i^{th}$  predicted or calculated value  
 $M_i$  = the  $i^{th}$  measured value  
 $n$  = the data set size

The term  $K_a\sigma_a$  is equivalent to the 95/95 one-sided upper tolerance uncertainty relative to the bias. The subscript “a” denotes that these factors are assembly parameters. If the data set is normal, then the following definitions apply:

$K_a$  = 95/95 one-sided tolerance factor (Ref. 7)

$\sigma_a$  = the standard deviation of the relative error between  
predicted and measured powers

If the data set is not normal, then the  $K_a\sigma_a$  term is determined from a non-parametric analysis. In this instance, the total uncertainty, or ONRF, is equal to:

$$ONRF = 1 - E_{mth} \quad (\text{non-parametric})$$

$E_{mth}$  is the  $m^{th}$  smallest relative error (negative errors indicate that the measured power is greater than the predicted power) for a sample size of  $n$  such that there is a 95% confidence level that at least 95% of the population has a relative error greater than this value. The  $K_a\sigma_a$  term in this instance is defined as:

$$K_a\sigma_a = Bias - E_{mth}$$

where,  $Bias$  = the mean predicted minus measured relative error  
 $E_{mth}$  = the relative error corresponding to the  $m^{th}$  smallest  
relative error that defines a 95/95 one-sided tolerance  
uncertainty based on non-parametric statistics. (Ref. 5)

ONRF's can also be developed based on an alternative methodology that was used to develop the legacy ERPI-NODE-P uncertainty factors. This methodology is described below.

In deriving the calculational uncertainty, an alternate approach is to use the algebraic difference between a calculated and a measured value forms a normally distributed random variable.

The difference variable is defined:

$$D_i = C_i - M_i \quad (11-1)$$

where:  $D$  is the  $i^{th}$  difference;  $1 \leq i \leq N$   
 $C$  is the  $i^{th}$  calculated value (radial or assembly peak axial power)  
 $M$  is the  $i^{th}$  measured value (radial or assembly peak axial power)

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

$$\bar{D} = \bar{C} - \bar{M} \quad (11-2)$$

$n$  = number of observations in sample

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

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

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

$n$  = number of observations in sample

A one-sided upper bound factor is derived by employing one-sided upper tolerance Limit (OSUTL) methodology. For a normal random variable  $X$  with a sample mean  $\bar{x}$  and standard deviation  $S$ , the *OSUTL* of  $X$  is defined by:

$$OSUTL(X) = \bar{x} + K_{\alpha} S \quad (11-3)$$

$$\text{where: } \bar{x} = \frac{1}{n} \sum_{i=1}^n X_i \quad (11-4)$$

$$S = \left[ \left( \sum_{i=1}^n (X_i - \bar{x})^2 \right) \div (n - 1) \right]^{1/2} \quad (11-5)$$

In equation 11-3,  $K$  is the one-sided tolerance factor. Equation 11-3 is formulated such that a predetermined proportion of the population ( $P$ ) is below the OSUTL with a confidence factor  $\alpha$  (Ref. 7).  $K$  is explicitly dependent on  $n$ ,  $P$ , and  $\alpha$ . Following industry practice,  $P = 95\%$  and  $\alpha = 95\%$ .

The *OSUTL* is given for  $D$  by:

$$OSUTL(D) = \bar{D} + KxS(D) \quad (11-6)$$

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

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

$$UL(C) = \bar{M} + \bar{D} + KxS(D) \quad (11-7a)$$

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

$$UL(C) = \bar{M} + \bar{C} - \bar{M} + KxS(D) \quad (11-8)$$

$$\text{or } UL(C) = \bar{C} + KxS(D) \quad (11-8a)$$

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

$$UL(C) = \bar{M} + \bar{C} + KxS(D) \quad (11-9)$$

Finally, the observed nuclear reliability factor (ONRF) is defined as the quotient of  $UL(C)$  from equation 11-9 and the mean of the measurements:

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

$$\text{or } ONRF = \frac{\bar{M} - \bar{D} + KxS(D)}{\bar{M}} \quad (11-11)$$

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

$$ONRF \times C \geq M \quad (11-12)$$

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

### **11.5 Statistically Combined Power Distribution Uncertainty Factors**

Power distribution uncertainty factors are applied in both the design of reload cores and the surveillance of an operating fuel cycle. In each case an uncertainty factor is applied to power distribution peaking factors to ensure a conservative comparison against thermal design limits. The uncertainty factor applied accounts for the calculation uncertainty (assembly and pin) and other applicable factors that may affect power peaking in the core. The  $F_{\Delta H}$  and  $F_Q$  peaking uncertainty factors are determined by statistically combining the assembly uncertainty and pin uncertainty. Additional uncertainties may be statistically combined as shown in Section 6.3 to obtain a combined uncertainty factor provided they independent.

FIGURE 11-1

Instrumented Fuel Assemblies  
McGuire and Catawba

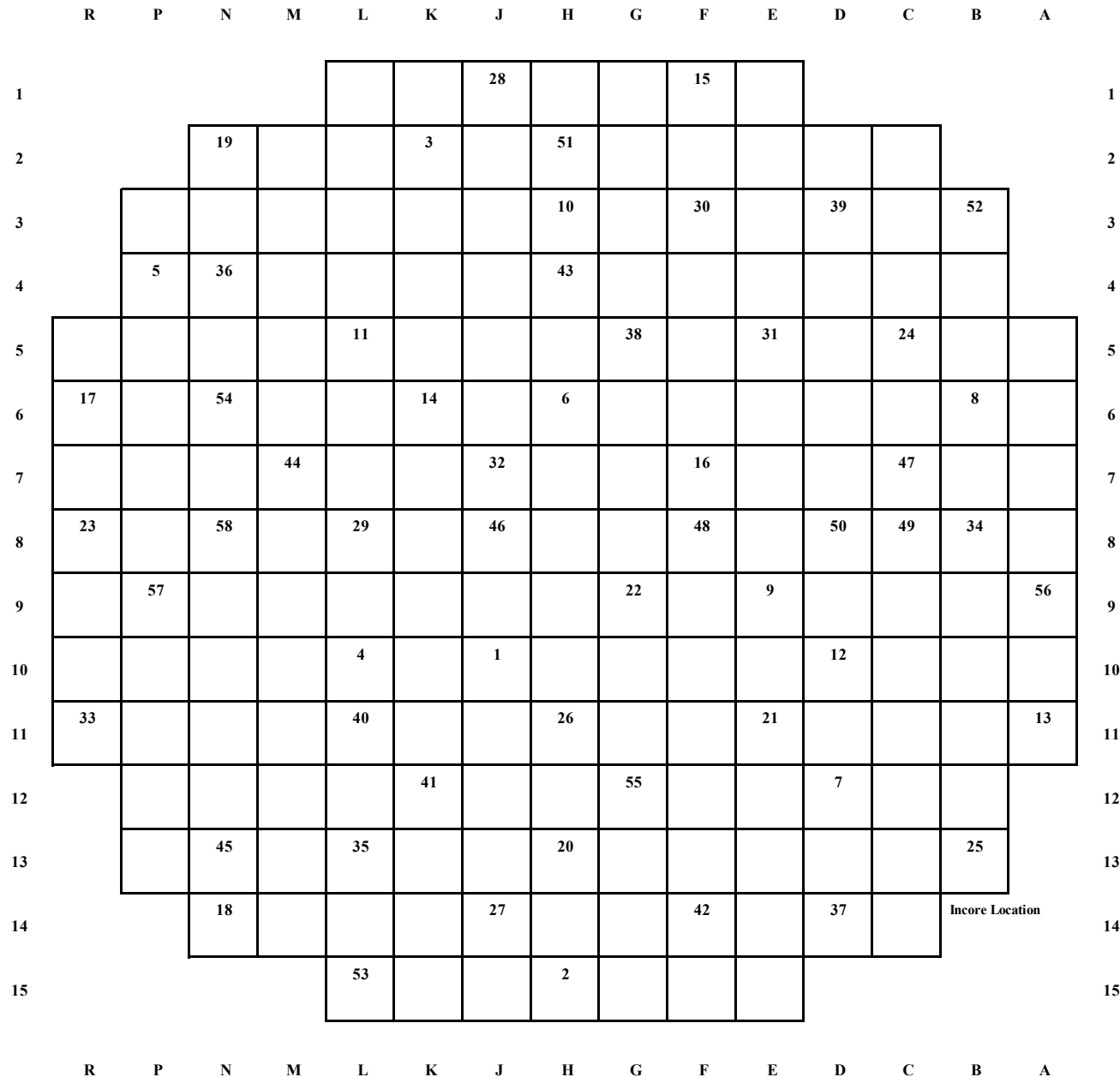


FIGURE 11-2

Instrumented Fuel Assemblies  
Harris

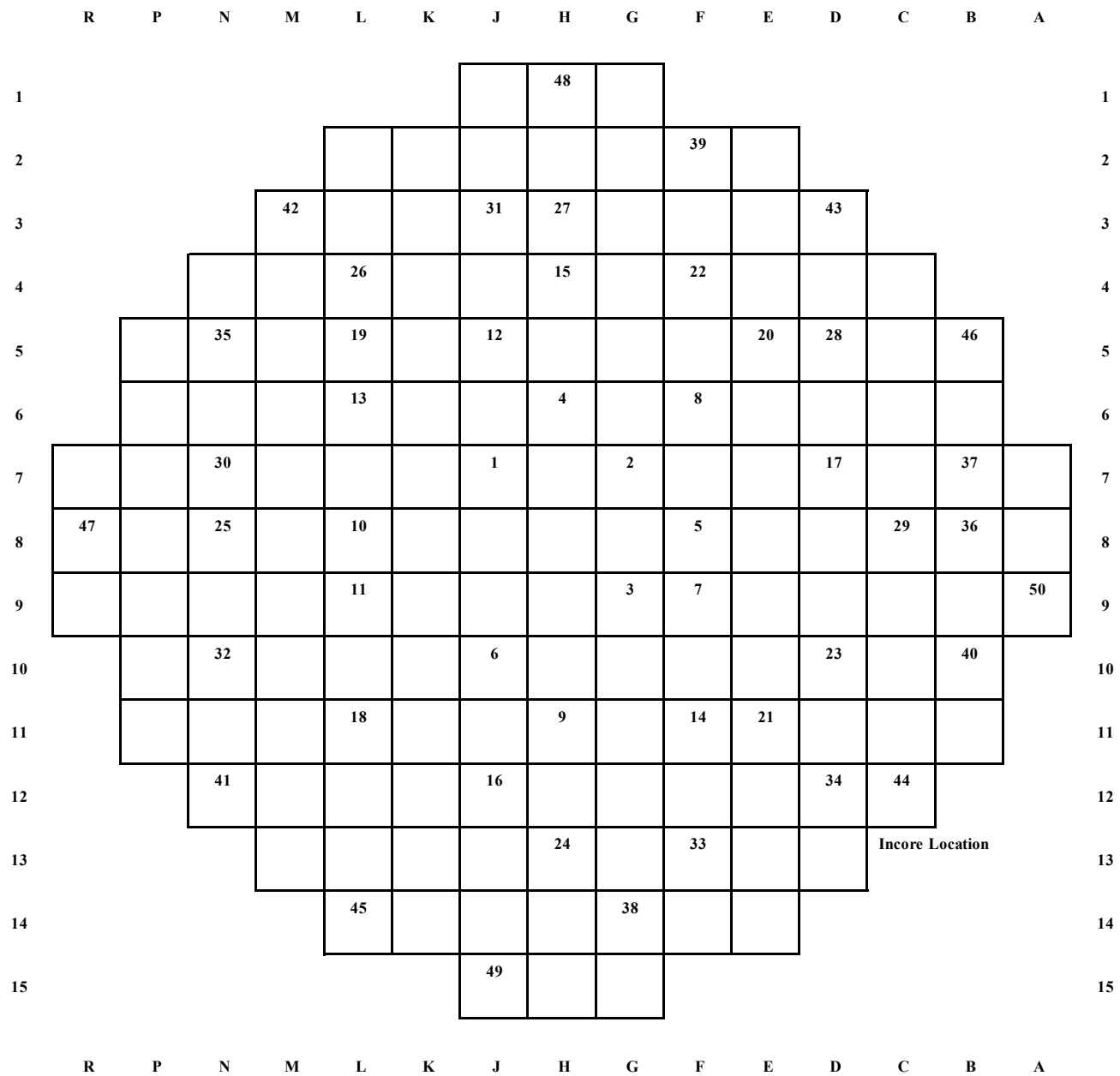
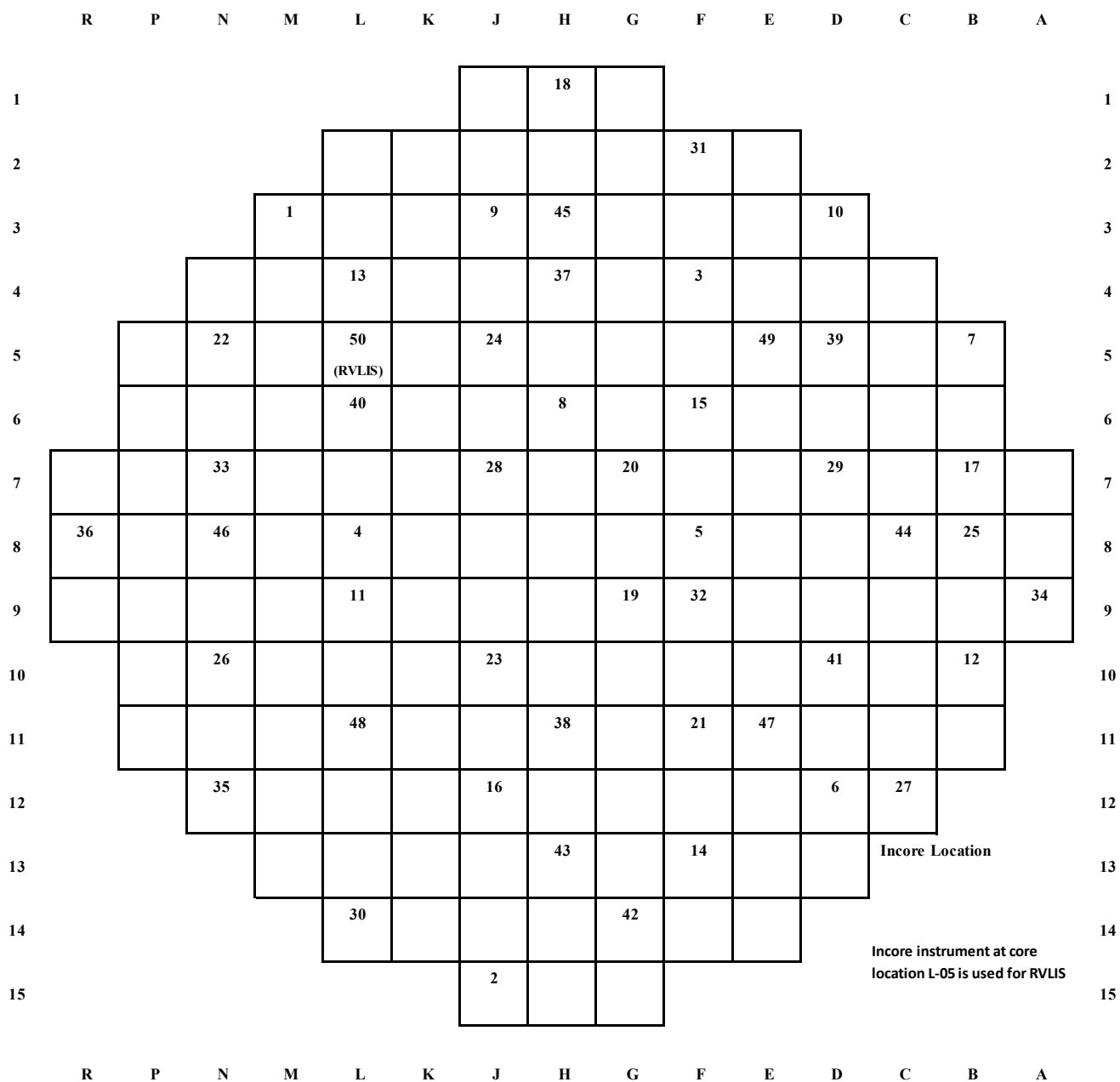




FIGURE 11-3

Instrumented Fuel Assemblies

Robinson



## 12. REFERENCES

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13. WCAP-13360-P-A, “Westinghouse Dynamic Rod Worth Measurement Technique”, Rev. 1, October 1998.
14. **Duke Energy Carolinas, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors”, Rev. 0, DPC-NE-1008-P, August 2015**
15. **“H. B. Robinson Steam Electric Plant Unit 1 (HBR2) Updated Final Safety Analysis Report”, Docket No.: 50-261**
16. **“Shearon Harris Nuclear Power Plant Final Safety Analysis Report”, Docket No.: 50-400**

### **Technical Justification of Changes for Revision 3**

### **DPC-NF-2010 Changes Constituting Revision 3**

DPC-NF-2010-A describes Duke's reload design process for McGuire and Catawba and is only approved for use per the NRC safety evaluation for McGuire and Catawba. The purpose of this calculation is to document the changes made to DPC-NF-2010-A to extend its applicability to Harris and Robinson.

The remainder of this section provides a description of each change along with a justification for each change as appropriate.

## **Global Changes**

The format of the report was updated to the current version of WORD 2010 including a font change from “Courier” to “Times New Roman”. These changes resulted in pagination changes and consequently changes to page numbers included in the Table of Contents. *Identification of the report location of each change refers back to the Revision 2a page numbering.*

## **Report Cover Sheet, Revision History, Abstract and Table of Contents Changes**

### **Intr-1 Cover Page**

Description of Change: References to the McGuire and Catawba Nuclear Station are removed, the report revision number is changed from 2a to 3, the report date is updated and the name of the Nuclear Division is changed to “Nuclear Fuels Engineering”. Duke Energy Progress is added to the applicable organizations that this report supports.

Technical Justification: Editorial.

### **Intr-2 Statement of Disclaimer, p. i**

Description of Change: Duke Energy Carolinas, LLC is replaced with Duke Energy and its subsidiaries to remove the specificity that the disclaimer is only applicable to Duke Energy Carolinas, LLC.

Technical Justification: Editorial.

### **Intr-3 Revision History, p. iii**

Description of Change: Content that is no longer needed in the Revision 1 and 2 descriptions is removed.

Technical Justification: Editorial.

### **Intr-4 Revision History, p. iii**

Description of Change: The reasons for revision 3 are added.

Technical Justification: Editorial.

### **Intr-5 Abstract, p. v**

Description of Change: Duke Energy Carolinas, LLC is replaced with Duke Energy, and the Harris and Robinson nuclear units are added to the list of reactors for which the DPC-NF-2010 design methodology applies.

Technical Justification: Editorial.

Intr-6 Table of Contents, pp. vi – viii, List of Tables, p. ix, and List of Figures, p. x

Description of Change: The page numbers are modified to reflect changes produced from the addition and removal of detail, and from the reformatting of the report.

Technical Justification: Editorial.

Intr-7 List of Figures, p. x

Description of Change: Figures 11-2 and 11-3 were added which show the instrumented core locations for Harris and Robinson, respectively.

Technical Justification: Editorial.

## Section 1 Changes – Introduction

### 1-1 Section 1.1 General Nuclear Design Description, second para., p. 1-1,

Description of Change: The Harris and Robinson nuclear units are added to the list of reactors where the methodology is applied.

Technical Justification: Editorial

### 1-2 Section 1.2 Definition of Terms, pp. 1-3 to 1-5

Description of Change: Definitions for the following terms are added:

AFD	flux difference between the top and bottom halves of the core
BTU	british thermal unit
DRWM	dynamic rod worth measurement technique
FFCD	final fuel cycle design
ITC	isothermal temperature coefficient
KW	kilowatt
LOCA	loss of coolant accident
LOFA	loss of flow accident
MTC	moderator temperature coefficient
OPDT	over-power delta-T trip function
OTDT	over-temperature delta-T trip function
RPS	reactor protection system
SDM	shutdown margin – the amount of negative reactivity ( $\rho$ ) by which a reactor core is maintained in a HZP subcritical condition after a control rod trip

Technical Justification: The above definitions were added to be consistent with the material contained in the report.



## 1-3 Section 1.2 Definition of Terms, pp. 1-3 to 1-5

The following definitions are modified.

**BO~~L~~C** beginning of ~~life~~ **cycle**

**MO~~L~~C** middle of ~~life~~ **cycle**

**EO~~L~~C** end of ~~life~~ **cycle**

**IFBA** **zirconium diboride** integral fuel burnable absorber

**RCCA** rod cluster control assembly; ~~the type of control rod assembly used in McGuire and Catawba. (All RCCA are full length absorbers for both plants.)~~

**Power density** thermal power produced per unit volume of the core (KW/liter **or KW/cm<sup>3</sup>**)  
**or relative to the fuel volume**

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

Technical Justification: The term BOL, MOL and EOL were changed to BOC, MOC and EOC to be consistent with the acronyms used in the report.

The type of integral burnable absorber for this acronym was identified to distinguish between gadolinium and zirconium diboride integral burnable absorbers.

Full length rod cluster control assemblies are used at McGuire, Catawba, Harris and Robinson. The specificity that RCCAs are only used at McGuire and Catawba is removed.

The definition of power density was modified to reflect there are several definitions of this term. Power density can be defined relative to core volume or fuel volume. It is typically defined relative to fuel volume.

The engineering hot channel factor is a vendor supplied value and for example may include other components such as surface area of the fuel rods and absorber variations. The engineering hot channel factor is vendor specific and its value is tied to manufacturing tolerances assumed in the manufacturing process.

## 1-4 Section 1.2 Definition of Terms, pp. 1-3 to 1-5

The following definitions are deleted.

a/o	atom percent
EQXE	equilibrium xenon condition
GWD/MTU	Gigawatt days per metric ton of initial uranium metal, 1 GWD/MTU is 1000 MWD/MTU
I	delayed neutron importance factor
$\Delta I$	flux difference between the top and bottom halves of the core; in this report, $\Delta I$ is a calculated value, rather than the difference between measured signals from the excore detectors
K(z)	The $F_Q$ limit assumed in the LOCA analysis normalized to the maximum value allowed at any core height
MWD/MTU	measure of energy extracted per unit weight of initial uranium metal fuel; is equal to 1 megawatt times 1 day, divided by 1 metric ton of uranium
$T_{\text{res}}$	resonance temperature of the fuel
w/o	weight percent

Technical Justification: These terms were deleted because they are no longer referenced in the report.

## **Section 2.0 – Fuel Description**

### 2-1 Section 2. Fuel Description, p. 2-1

Description of Change: A description of the Harris and Robinson reactors and fuel designs employed in each reactor is added.

Technical Justification: Editorial.

## **Section 3.0 – Nuclear Code System**

### **3-1 Section 3. Entire Section, pp. 3-1 – 3-3**

Description of Change: The CASMO-5/SIMULATE-3 methodology (new Reference 14) is added as an alternate nuclear analysis methodology for performing nuclear physics and power distribution calculations.

Technical Justification: Nuclear analyses at McGuire and Catawba are currently performed using the NRC-approved CASMO-4/SIMULATE-3 methodology described in DPC-NE-1005-P (Ref. 4). The nuclear analysis methodology that will be used for Harris and Robinson, and in the future for McGuire and Catawba, is based on the CASMO-5/SIMULATE-3 code system. This methodology is currently under NRC review and is described in DPC-NE-1008-P, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors”, new Reference 14. Use of the DPC-NF-2010 methodology for Harris and Robinson requires approval of the DPC-NE-1008-P methodology report prior to its use.

### **3-2 Section 3. Figure 3-1, p. 3-4**

Description of Change: Figure 3-1 is updated to include the CASMO-5/SIMULATE-3 nuclear analysis method.

Technical Justification: Refer to the technical justification for item 3-1.

## **Section 4.0 – Fuel Cycle Design**

### **4-1 Section 4.2.1 Fuel Shuffle Optimization and Cycle Depletion, third para., p. 4-3**

Description of Change: A qualifier is added to the term rotation and the source of the spent fuel assemblies is identified.

Technical Justification: Fuel assemblies are not physically rotated. Fuel rotations during the core shuffle process are effectively accomplished by the cross quadrant shuffling of fuel assemblies across lines of symmetry, not by physically changing the fuels' orientation. Fuel assemblies rotated are either burned or fuel assemblies re-inserted from the spent fuel pool. The source of spent fuel was also added for completeness.

### **4-2 Section 4.2.2 Rod Worth Calculations, first para., p. 4-3**

Description of Change: Specific reference to the McGuire and Catawba reactors is removed.

Technical Justification: Editorial.

### **4-3 Section 4.2.2 Rod Worth Calculations, first para., p. 4-4**

Description of Change: The Harris and Robinson reactors are added to the list of reactors that operate in an all rods out or feed-and-bleed mode.

Technical Justification: All Duke units are operated as base load units with Control Banks near their ARO position.

### **4-4 Section 4.2.2 Rod Worth Calculations, first para., p. 4-4**

Description of Change: The reason control rods are inserted a small amount is provided along with the typical bite position.

Technical Justification: This clarification states Control Bank D is inserted a small amount to provide immediate reactivity and axial offset control. This is typical for reactors using axial blanket fuel.

### **4-5 Section 4.2.2 Rod Worth Calculations, first para., p. 4-4**

Description of Change: A clarification is made that Control Bank B may or may not be inserted by removing specific references to the control banks inserted during startup.

Technical Justification: Rod insertion limits for Harris are such that Control Bank B is not inserted during mode 1 or 2 operation. Control Bank B may be inserted during mode 1 or 2 operation at McGuire, Catawba, and Robinson.

### **4-6 Section 4.2.4 Reactivity Coefficients and Defects, first para., p. 4-5**

Description of Change: Acronyms HFP and HZP are defined.

Technical Justification: Editorial.

4-7 Section 4.2.7 Kinetics Parameters, second para., p. 4-6

Description of Change: The specificity that core averaged kinetics parameters are calculated using delayed neutron data from CASMO-4 is changed to just refer to the CASMO code.

Technical Justification: Kinetics data is calculated consistent with the code system being used to perform cycle-specific reload calculations, either CASMO-4/SIMULATE-3 or CASMO-5/SIMULATE-3.

4-8 Table 4-1, p. 4-8

Description of Change: Coastdown modes are added.

Technical Justification: Power and T-ave coastdowns were added as a mode of operation. These type of coastdowns are typically performed at end-of-cycle conditions to extend operation of the nuclear unit. This type of operation has been found to be a more efficient means of utilizing nuclear fuel.

## **Section 5.0 Changes – Nodal Analysis Methodology**

### **5-1 Section 5.1 Purpose and Introduction, first para., p. 5-1**

Description of Change: The CASMO-5/SIMULATE-3 methodology (new Reference 14) was added to the list of analytical models that may be used to perform nuclear calculations.

Technical Justification: Refer to justification 3-1.

### **5-2 Section 5.3.1 Differential Rod Worth Analysis, first para., p. 5-3**

Description of Change: BOC and EOC acronyms are defined.

Technical Justification: Editorial.

### **5-3 Section 5.3.2 Integral Rod worth Analysis, first para., p. 5-3**

Description of Change: The reference to the use of a 2-dimensional code for calculating integral rod worths is removed. Minor editorial changes are also made.

Technical Justification: The reference that rod worth calculations may be performed in 2-dimensions is a holdover from when computer computations times were significant and the number of three-dimensional calculations were reduced where possible to manage computer resources. With today's computational resources, all calculations are performed in three dimensions. In addition, the use of a three-dimensional calculation is more precise since it appropriately models the spatial change in the flux distribution between the ARO and control rod inserted configuration relative to a two-dimensional model.

### **5-4 Section 5.3.2 Integral Rod worth Analysis, second para., p. 5-3**

Description of Change: The ARI acronym is defined.

Technical Justification: Editorial.

### **5-5 Section 5.4.1 Shutdown Margin, last para., p. 5-4**

Description of Change: A clarification is made in the last paragraph to the second sentence where the phrase "the highest stuck rod" is changed to the "the highest worth stuck rod".

Technical Justification: Editorial.

### **5-6 Section 5.4.1 Shutdown Margin, first para., p. 5-5**

Description of Change: The reference to the use of a 2-D model is the calculation of the power defect is removed.

Technical Justification: All nuclear analysis calculations are performed in three-dimensions. The two-dimensional analysis method is a holdover from the original inception of the report when computational times were a concern when using codes such as PDQ.

5-7 Section 5.4.1 Shutdown Margin, second para., p. 5-5, Table 5-1(Item 8)

Description of Change: A clarification is made that xenon redistribution effects are included in Item 8 in Table 5-1. The values in Table 5-1 are updated to values representative of contemporary core designs.

Technical Justification: In the shutdown margin example provided, a clarification is made that xenon redistribution effects associated with a transient xenon condition are included in Item 8 in Table 5-1. BOC shutdown margin values in Table 5-1 are updated to reflect a typical values for a contemporary core designs.

5-8 Section 5.4.2 Shutdown Boron Concentration, second and third paragraphs, pp. 5-5 – 5-6

Description of Change: Specific reference to the shutdown margin values of 1.3 and 1.0% are removed and a reference to the plants Technical Specifications for these values is made.

Technical Justification: Values of the required shutdown margins for McGuire and Catawba were removed because they are different than shutdown margins for Harris and Robinson. The reference to Technical Specifications provides the source of plant specific required shutdown margin values.

5-9 Section 5.4.2 Shutdown Boron Concentration, last paragraph in section, p. 5-6

Description of Change: The specific reference to HZP moderator temperature is removed.

Technical Justification: The HZP moderator temperatures for Harris, McGuire and Catawba is 557 °F, while the HZP moderator temperature for Robinson is 547 °F. The specific value of the HZP power condition is not required.

5-10 Section 5.7 Rod Ejection Analysis, first para., p. 5-9

Description of Change: References 15 and 16 for Harris and Robinson are added to the Updated Final Safety Analysis Report (UFSAR) reference list.

Technical Justification: References 15 and 16 refer to the Harris and Robinson UFSARs. Each units' UFSAR contains the limiting criteria for the rod ejection accident.



5-11 Section 5.7 Rod Ejection Analysis, first and fifth paragraphs in section, pp. 5-9 – 5-10

Description of Change: A clarification is made that a NRC-approved rod ejection accident methodology is required for evaluation of this event. The specific reference to the McGuire and Catawba rod ejection methodology (Ref. 10) is removed in the fifth paragraph.

Technical Justification: The clarification is made to avoid updating the report if a new or updated methodology is introduced. Reference 10 is retained as an example NRC-approved rod ejection methodology. The important aspect is a NRC-approved evaluation methodology must be used to analyze the REA event. Reference to the rod ejection methodology in the fifth paragraph is removed because it is redundant information.

5-12 Section 5.7 Rod Ejection Analysis, third para., p. 5-9

Description of Change: A clarification is made that Control Bank B may or may not be inserted.

Technical Justification: Rod insertion limits for Harris are such that Control Bank B is not inserted during mode 1 and 2 operation at Harris. Control Bank B may be inserted during mode 1 or 2 operation at McGuire, Catawba, and Robinson.

5-13 Section 5.8 Dropped Rod Analysis, first paragraph, p. 5-10

Description of Change: References 15 and 16 for Harris and Robinson are added to the Updated Final Safety Analysis Report (UFSAR) reference list.

Technical Justification: References 15 and 16 refer to the Harris and Robinson UFSARs. They contain the limiting criteria for the dropped rod accident.

5-14 Section 5.8 Dropped Rod Analysis, first and second paragraphs, p. 5-10

Description of Change: A clarification is made that a NRC-approved dropped rod accident methodology is required for evaluation of this event. The specific reference to the McGuire and Catawba dropped rod methodology (Ref. 10) is removed in second paragraph.

Technical Justification: The clarification is made to avoid updating the report if a new or updated methodology is introduced. Reference 10 is retained as an example NRC-approved dropped rod methodology. The important aspect is a NRC-approved evaluation methodology must be used to analyze the dropped rod accident.

5-15 Section 5.10 Moderator Temperature Coefficient, first paragraph, p. 5-11

Description of Change: The specific reference to McGuire and Catawba is removed.

Technical Justification: Editorial.

5-16 Section 5.10 Moderator Temperature Coefficient, first paragraph, p. 5-11

Description of Change: The acronym cold hot zero power (CHZP) is removed .

Technical Justification: The term no longer has any specific meaning and is replaced by HZP. HZP is implied to mean the nominal no load temperature (typically 547 °F or 557 °F.)

5-17 Table 5-1, p. 5-11

Description of Change: “Example” was added to the Table title.

Technical Justification: Editorial.

## **Section 6.0 Changes – Calculation of Safety Analysis Physics Parameters**

### **6-1 Section 6.1 Safety Analysis Physics Parameters, second para., p. 6-1, Section 6.2 Core Power Distributions, second para., p. 6-1**

Description of Change: The specific references to the McGuire and Catawba safety analysis physics parameters methodology (Reference 10) and the core operating limits methodology (Ref. 9) are removed and cited as examples of NRC-approved methodologies.

Technical Justification: Reference 10 describes the safety analysis physics parameters methodology for McGuire and Catawba. Reference 9 describes the core operating limits methodology. The clarification is made to avoid updating the report if a new or updated methodologies are introduced. References 9 and 10 are retained as an examples NRC-approved core operating limits and safety analysis physics parameters methodologies. The important aspect is a NRC-approved evaluation methodology must be used in the evaluation of UFSAR Chapter 15 accidents.

### **6-2 Section 6.2 Core Power Distributions, second para., p. 6-1**

Description of Change: Additional detail is added to describe the difference between initial conditions and transient or accident analysis induced power peaking.

Technical Justification: This clarification describes the initial condition peaking condition as peaking factors that are allowed by Technical Specification axial flux difference and rod insertion limits. Accident or transient analysis peaking factors are those that occur at the conditions defining the limiting departure from nucleate boiling (DNB) or fuel melt state point for Condition II, III or IV transients. Accidents for which either the thermal limit acceptance criteria or Technical Specification limits are exceeded are evaluated or reanalyzed to ensure acceptable accident consequences, or the core is redesigned to obtain acceptable peaking factors and accident consequences.

### **6-3 Section 6.3 Power Peaking Factors and Reliability Factors, last para., p. 6-2**

Description of Change: The CASMO-5/SIMULATE-3 methodology (new Reference 14) is added as a source of assembly and pin power uncertainty factors.

Technical Justification: Assembly and pin power uncertainty factors for McGuire and Catawba are based on the NRC-approved CASMO-4/SIMULATE-3 methodology described in DPC-NE-1005-P. The assembly and pin power uncertainty factors that will be used for Harris and Robinson, and in the future for McGuire and Catawba, are based on the CASMO-5/SIMULATE-3 methodology. This methodology is currently under NRC review and is described in DPC-NE-1008-P, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors”. DPC-NE-1008-P is new Reference 14. The DPC-NE-2010-A methodology for Harris and Robinson requires approval of the DPC-NE-1008-P methodology report prior to its use.

## **Section 7.0 Changes – Three-Dimensional Power Peaking Analysis**

### **7-1 Section 7.0 Three-Dimensional Power Peaking Analysis, p. 7-1**

Description of Change: A high level description of the Maneuvering Analysis methodology is added.

Technical Justification: A description of the Maneuvering Analysis methodology is added for completeness. The information provided is from Reference 9.

## **Section 8.0 Changes – Radial Local Analysis**

### **8-1 Section 8.0 Radial Local Analysis, p. 8-1**

Description of Change: A clarification is made to identify that different pin uncertainty factors exist depending upon the nuclear analysis methodology employed and fuel rod type.

Technical Justification: The information added conveys that different pin power uncertainties exist for standard UO<sub>2</sub> fuel rods and fuel rods with the gadolinia integral burnable absorber. Different factors also exist based on the nuclear analysis methodology used – either CASMO-4/SIMULATE-3 or CASMO-5/SIMULATE-3. Pin uncertainties for the CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 based methodologies are described in References 4 and 14. The CASMO-5/SIMULATE-3 methodology is under NRC review.

## **Section 9.0 Changes – Development of Core Physics Parameters**

### **9-1 Section 9.2.5 Power Distribution, first para., p. 9-4**

Description of Change: The specific reference to the McGuire and Catawba reactors is removed and a clarification is made stating the reasons for performing the power distribution comparisons.

Technical Justification: Editorial.

## **Section 10.0 Changes – Physics Test Comparisons**

### **10-1 Section 10.0 Physics Test Comparisons, p. 10-1**

Description of Change: New Reference 14 is added which refers to the CASMO-5/SIMULATE-3 methodology.

Technical Justification: The CASMO-4/SIMULATE-3 methodology described in DPC-NE-1005-P contains the benchmarks for this codes system. This methodology is approved by the NRC and is used for McGuire and Catawba reload design analysis. The CASMO-5/SIMULATE-3 benchmark calculations are described in DPC-NE-1008-P, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors”, new Reference 14. This is the methodology that is applicable to Harris and Robinson. The DPC-NE-1008-P methodology is currently under NRC review.

### **10-2 Section 10.2.1 Measurement Technique, second para., p. 10-2**

Description of Change: B-10 depletion is identified as another source of uncertainty in measured critical soluble boron concentrations.

Technical Justification: The uncertainty in the reactor core B-10 concentration is an additional component of the total uncertainty of measured critical soluble boron concentrations.

### **10-3 Section 10.3.1 Measurement Technique p. 10-2**

Description of Change: Boron dilution is added as another name for boron swap.

Technical Justification: Editorial. Boron swap or boron dilution is an industry accepted technique for performing rod worth measurements.

### **10-4 Section 10.3.1 Measurement Technique, second para., p. 10-3**

Description of Change: Measurement of shutdown banks is added to the list of rod banks when using the rod swap technique.

Technical Justification: This clarification specifies that all control and shutdown banks are measured when the rod swap measurement technique is used.

### **10-5 Section 10.3.1 Measurement Technique, second para., p. 10-3**

Description of Change: The rod swap measurement technique is modified to be consistent with Revision 1 of DPC-NE-1003-A, Reference 11.

Technical Justification: The clarification in the rod swap technique is based on the NRC-approved method described in Rev. 1 to DPC-NE-1003-A. Revision 1 allows for the swap of the two test banks prior to movement of the Reference bank following the initial test bank measurement.

10-6 Section 10.4.1 Measurement Technique, p. 10-4

Description of Change: An alternate method for measuring the isothermal temperature coefficient is added.

Technical Justification: The isothermal temperature coefficient measurement technique used at Harris and Robinson is different from that employed at McGuire and Catawba. At Harris and Robinson, the temperature coefficient is calculated by dividing the reactivity change produced by a change in reactor coolant moderator temperature by the change in moderator temperature. This contrasts with the slope measurement technique used at McGuire and Catawba.

## **Section 11.0 Changes – Power Distribution Comparisons**

### **11-1 Section 11.2.2 Measurement System Description, first and second paragraphs, pp. 11-1 – 11.2**

Description of Change: A description of the Harris and Robinson incore system is added.

Technical Justification: The incore systems at McGuire/Catawba and Harris/Robinson are functionally equivalent except for the number of incore detectors and the number of instrumented locations. Differences in the number of fission detectors and instrumented core locations used to monitor the 3-loop Harris and Robinson, and the 4-loop McGuire and Catawba reactor cores is due to differences in core size. The number of instrumented locations was set by NSSS vendor to maintain similar measured core fractions. The incore system at Robinson and Harris each have 5 fission detectors. Harris has 50 instrumented core locations while Robinson has 49 instrumented core locations. This compares to 58 instrumented core locations for McGuire and Catawba. Robinson has one less instrumented core locations because the instrument at core location L-05 was converted to a reactor vessel water level instrument. The wording that more than 600 signals are “recorded” was changed to “available” to remove the implication that while there are more than 600 signals available axially along each of the instrumented fuel assemblies, they may not be recorded. The number of signals recorded is dependent upon each plants computer system.

### **11-2 Section 11.3.1 Fuel Cycle Simulations, last sentence, p. 11-3**

Description of Change: The phrase “with a 95% confidence level that 95% of the predicted powers are equal to or larger than the measured value” is removed.

Technical Justification: This clarification is made because the subject information is redundant with that already stated in section 11.4.

### **11-3 Section 11.3.1 Fuel Cycle Simulations, p. 11-3**

Description of Change: The definition for relative difference is inserted after the first paragraph.

Technical Justification: This clarification is made to avoid application of an incorrect definition of the relative difference used to calculate power distribution uncertainty factors. The definition is consistent with the definition used in References 4 and 14.

### **11-4 Section 11.4 Statistical Analysis, first para., p. 11-3**

Description of Change: The phrase “Predicted and measured power distribution data is used to develop” is added.

Technical Justification: The phrase is added to improve clarity.

11-5 Section 11.4 Statistical Analysis,  $K_a$  definition, p. 11-4

Description of Change: A reference for the 95/95 one-sided tolerance factor, “K”, is added.

Technical Justification: Editorial.

11-6 Section 11.4 Statistical Analysis,  $E_{mth}$  definition, p. 11-5

Description of Change: A reference determine the “m” factor for non-parametric 95/95 one-sided tolerance uncertainty is added.

Technical Justification: Editorial.

11-7 Figure 11-1, p. 11-9

Description of Change: Figure 11-1 is replaced to show incore instrument identifiers.

Technical Justification: Editorial.

11-8 Figures 11-2 and 11-3, New

Description of Change: Instrumented core locations for the Harris and Robinson reactors are added as new Figures 11-2 and 11-3.

Technical Justification: Editorial.



## **Section 12.0 Changes - References**

### 12-1 Section 12 References, p. 12-1 to 12-2

Description of Change: References 6, 10 and 12 are updated to the current revision of these reports. Reference 9 is updated to the future version of this report. References 14 through 16 are added.

Technical Justification: Editorial. References 10 and 12 are updated to the current NRC-approved versions of these reports. Reference 6 is updated to the current version of Reg. Guide 1.126. Reference 14 describes the CASMO-5/SIMULATE-3 nuclear analysis method for Harris and Robinson. The current approved nuclear analysis methodology (applicable to McGuire and Catawba reload design analyses) is based on the CASMO-4/SIMULATE-3 code system (Ref. 4). References 15 and 16 refer to the Harris and Robinson final safety analysis reports.

It is noteworthy that Reference 14 has been submitted to the NRC for review and approval. The request for review of Reference 9 is coincident with the review of DPC-NF-2010. The approval date will be specified in the published version of the DPC-NF-2010 report following NRC approval.

**Attachment 8**

**DPC-NE-2011, Revision 2, “Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors” and Technical Justification of Changes (Redacted)**

**Nuclear Design Methodology Report  
For Core Operating Limits of  
Westinghouse Reactors**

**DPC-NE-2011-A  
Revision 1a 2**

**~~June 2009~~ January 2016**

Nuclear **Fuels** Engineering ~~Division~~  
Nuclear Generation Department  
Duke Energy Carolinas, LLC  
**and**  
**Duke Energy Progress, INC**

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### Revision History

Revision	Description
DPC-NE-2011P, Original Issue	Originally submitted to the NRC for approval in April 1988. An additional submittal was made to the NRC supplying responses to a request for additional information.
DPC-NE-2011P-A, Original Issue	NRC approved version issued in January 1990.
DPC-NE-2011P, Revision 1	<p>Submitted to the NRC for approval in August 2001.</p> <p>This revision updates the report for completeness (1) to indicate the use of methods approved by the NRC subsequent to the implementation of the original issue and (2) to expand the description of <math>F_Q</math> monitoring to include the centerline fuel melt criterion.</p> <p>This revision also reflects changes in the descriptions of the xenon transient conditions.</p> <p>Finally, various editorial changes are made, including updating the Table of Contents and revising the page numbering format.</p> <p><del>Changes associated with this revision are denoted by revision bars, except for format changes.</del></p>
DPC-NE-2011P-A Revision 1	NRC Approved. SER issued October 1, 2002.
DPC-NE-2011P-A Revision 1a	<p>This revision to DPC-NE-2011P updates the methodology with methodology that has been approved by the NRC in the following methodology reports:</p> <ul style="list-style-type: none"> <li>• DPC-NE-2009P-A, “Duke Power Company Westinghouse Fuel Transition Report”</li> <li>• DPC-NE-1005P-A, “Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX”</li> </ul>

### Revision History Continued

Revision	Description
DPC-NE-2011P-A Revision 1a	<p>Additional revisions are performed that consist of clarifications, editorial changes, and error corrections. The major elements consisting of revision <del>2</del> <b>1a</b> are:</p> <ul style="list-style-type: none"> <li>• Removal of the EPRI-ARMP PDQ and EPRI-NODE-P methodology</li> <li>• The addition of the 31 EFPD surveillance methodology (NRC approved methodology from DPC-NE-2009P-A)</li> <li>• Replacement of the CASMO-3 methodology with the CASMO-4 methodology for the generation of nuclear power distributions (NRC approved content from DPC-NE-1005-P-A)</li> <li>• Addition of OTDT and OPDT calculation methodology</li> <li>• Revisions based on accumulated experience with the methodology</li> <li>• Editorial changes, error corrections, and clarifications</li> </ul>
DPC-NE-2011-P Revision 2	<p><b>Revision 2 extends the applicability of the Core Operating Limits analysis methodology to the Harris and Robinson nuclear units. The major elements that comprise Revision 2 are:</b></p> <ul style="list-style-type: none"> <li>• <b>Modify the write-up so the method applies to the McGuire, Catawba, Harris and Robinson Westinghouse nuclear units</b></li> <li>• <b>Revise the method for generating xenon distributions based on accumulated experience with the methodology</b></li> <li>• <b>Add the option to adjust the OTDT <math>f_1(\Delta I)</math> trip reset function breakpoints if the <math>F_Q</math> centerline fuel melt surveillance limit is exceeded</b></li> <li>• <b>Remove the base load option</b></li> <li>• <b>Remove the quadrant power tilt pently from power distribution surveillances</b></li> <li>• <b>Perform editorial changes and clarifications</b></li> </ul>

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## 1. INTRODUCTION

### 1.1. Purpose

This report describes the methodology for performing a maneuvering analysis for ~~four-loop, 193 fuel assembly~~ **Duke Energy's** Westinghouse reactors, **encompassing the** ~~such as McGuire, and Catawba, Harris and Robinson Nuclear Stations~~ **units**. Duke Energy ~~Carolina~~ has developed this methodology as an alternative to the existing Relaxed Axial Offset Control (RAOC) Methodology (1). This maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of Duke Energy's ~~Carolina~~ nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the over-power  $\Delta T$  (OP $\Delta T$ ) and over-temperature  $\Delta T$  (OT $\Delta T$ ) reactor protection system (RPS) trip functions.

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

These limits are monitored via Technical Specifications.

### 1.2. Summary of the Methods

The operating limits define the AFD - power level space and rod insertion limits which provide assurance that the peak local power in the core is not greater than that assumed in the analysis of design basis accidents or transients (loss of coolant accident (LOCA), ~~or~~ loss of flow accident (LOFA)<sup>1</sup> **or other limiting Condition II transient where the initial condition peaking does not change as the result of the event**). The loss of flow accident is **typically** the limiting departure from nucleate boiling Condition II transient (transient of moderate frequency). Operating the reactor within the allowed AFD - power level window and rod insertion limits satisfies the power peaking assumptions of the LOCA and LOFA analyses.

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<sup>1</sup> Referral to the LOFA transient or LOFA limits in the remainder of the report is implied to mean the limiting Condition II transient where the initial condition peaking does not change as the result of the event.

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

The maneuvering analysis uses a three dimensional nodal reactor model to calculate a set of power distributions at several points in core life. These power distributions are based on a set of abnormal xenon distributions to insure predicted power distributions are conservative with respect to those expected to occur. Three dimensional local peak pin power distributions are explicitly calculated with SIMULATE-3. Appropriate uncertainty factors **as described in section 4 of this report** are applied to the calculated power distributions which are then evaluated against the various thermal limits. The operating limits and the  $f(\Delta I)$  function of either the OPAT or the OTAT RPS trip functions are then set to exclude the power distributions that exceed the respective thermal limits. Figure 1 shows a representative flow chart of the data as it goes through a SIMULATE-3 based maneuvering analysis.

### **1.3. Applicability of the Method**

The maneuvering analysis presented in this report applies to Westinghouse four loop **and three loop**, ~~assembly~~ <sup>193</sup> reactors **at the McGuire, Catawba, Harris and Robinson nuclear sites**. This method is ~~intended to be~~ used to set or validate the AFD - power level operating limits, the control rod insertion limits, and the RPS trip limits.

**Implementation of the method requires the use of a NRC-approved three-dimensional core simulator capable of calculating assembly average and pin power distributions. The CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 code is used for this purpose. A system of computer programs is used to implement this method. A description of the computer programs currently in use is contained in Appendix A. This list includes both the major design ~~These~~ codes **were** approved by the NRC (15) in References 15 and 16 and minor codes that are used for post processing data. A description of the SIMULATE-3 code is contained in Appendix A, and in References 15 and 16. Post processing codes are used to process data generated by the core simulator to calculate margins to thermal limits.**



#### 1.4. Definition of Terms

##### AFD

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

##### Xenon Offset

Xenon offset is defined as the difference in the xenon concentration in the top half of the core minus the concentration in the bottom half of the core divided by the total xenon concentration multiplied by 100.

##### $F_Q$

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

##### $F_{\Delta H}$

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

##### QPTR

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

##### MATP

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

##### MARP

Maximum Allowed Radial Peak values derived from MATP values by dividing the MATP by the axial peak. **Axial peak is defined as the ratio of  $F_Q$  divided by  $F_{\Delta H}$ .**

##### OTΔT

The thermal over-temperature delta-T trip function to provide protection against postulated UFSAR transients.

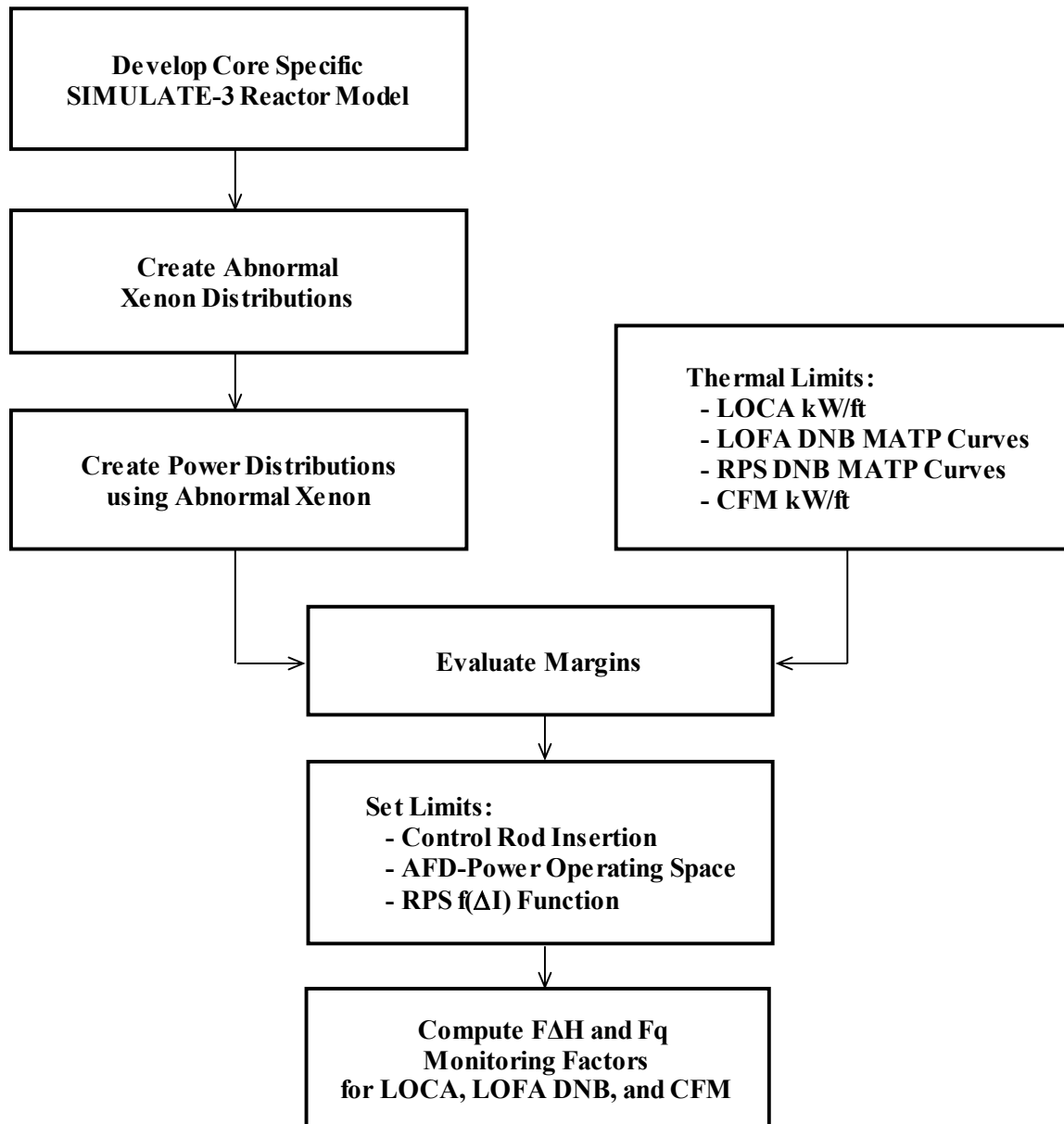
#### OP $\Delta$ T Over-power delta-T trip function

The thermal over-power delta-T trip function used to provide protection against postulated UFSAR transients.

#### $f(\Delta I)$

The delta-imbalance trip reset function used by both the OT $\Delta$ T and OP $\Delta$ T trip functions. The  $f(\Delta I)$  function defines the indicated difference between the top and bottom power-range excore detectors.

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



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## 2. GENERATION OF POWER DISTRIBUTIONS

### 2.1. Description of the Models Used

The three dimensional xenon and local peak pin power distributions are generated with SIMULATE-3. The SIMULATE-3 model was approved for use in reload core design analyses in References 15 and 16. A description of the SIMULATE-3 model is presented in these References 15.

### 2.2. Times in Core Life

The maneuvering analysis is typically performed at [

] <sup>a,c</sup>

### 2.3. Generation of Abnormal Xenon Distributions

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

] <sup>a,c</sup> Table 1 shows typical initial and transient conditions of the reactor for each of the transients. [

] <sup>a,c</sup>

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

] <sup>a,c</sup> Because of these factors, the xenon transients in the reactor model will be more severe than could be reasonably expected to occur.

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

] <sup>a,c</sup>

The control rod positions for the xenon transients were chosen to be at or near the expected rod insertion limits. **Control rods are inserted to the control rod insertion limit** [

] <sup>a,c</sup> ~~The final control rod insertion limits may be different from the positions used in the xenon transients and the analysis will still be valid. This is because the xenon transients are so severe that the maneuvering analysis results are not sensitive to the control rod motions that drive the xenon transients.~~

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

## 2.4. Generation of Power Distributions

Using the abnormal xenon distributions from the xenon transients, three dimensional power distributions are generated so that the operating and the RPS limits can be determined. As shown on Table 2, power distributions are generated with [ <sup>a,c</sup> The operating limits are pre-conditions that would prevent exceeding the peak local power in the core assumed in the loss of coolant accident (LOCA) analysis or the loss of flow accident (LOFA, or a primary coolant pump trip **another limiting Condition II transient where the initial condition peaking does not change as the result of the event**) analysis. Because this is the normal operating mode of the

reactor, control rod motion will be constrained by the power dependent rod insertion limits. [

] <sup>a,c</sup> Power distributions for the operating limits are generated with these abnormal xenon distributions with the reactor at nominal conditions.

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

] <sup>a,c</sup>

The limit of the control rod motion for [

] <sup>a,c</sup>

During an [

] <sup>a,c</sup>

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

] <sup>a,c</sup> Table 2 shows the reactor conditions and range of control rod positions. Criticality in the reactor model is maintained by instantaneous changes in soluble boron concentrations.

**Table 1**  
**Typical Reactor Conditions During Xenon Transients**

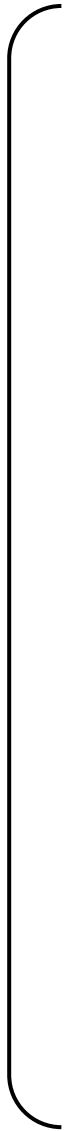
<u>Transient Name</u>	Initial Conditions		Transient Conditions		
	<u>% Power</u>	<u>Control Rods</u>	<u>% Power</u>	<u>Control Rods</u>	
					a, c



### Typical Power Levels and Control Rod Bank Positions for Generating Power Distributions

a, c

**Figure 2**  
**Sample Xenon Transient at Beginning of Life**  
**AFD Versus Transient Time**



a, c

**Figure 3**  
**Sample Xenon Transient at Beginning of Life**  
**Xenon Concentration Versus Transient Time**

a, c

**Figure 4**  
**Sample Xenon Transient at Beginning of Life**  
**Xenon Offset Versus Transient Time**

a, c

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

a, c

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### 3. UNCERTAINTY FACTORS

#### 3.1. Power Distribution

The power peaks calculated in the Maneuvering Analysis are adjusted to account for calculation uncertainty and other applicable factors that may affect the power peaking in the core. Additional potential power peaking uncertainties may be included for effects such as engineering hot channel factor, rod bow, and fuel assembly bow.

References 15 and 16 presents calculation~~ed~~ peaking uncertainties based on the benchmarking analysis of measured to predicted power distribution **for the CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 methodologies**. The peaking uncertainty factor is calculated as described below.

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

Where:

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

UC – Calculation Uncertainty

$$\text{For the Pin Total Peak (F}_Q\text{):} \quad \text{UC}^2 = \text{UCT}^2 + \text{URL}^2$$

$$\text{For the Pin Radial Peak (F}_{\Delta H}\text{):} \quad \text{UC}^2 = \text{UCR}^2 + \text{URL}^2$$

$$\text{For the Assembly Axial Peak (F}_Z\text{):} \quad \text{UC}^2 = \text{UCA}^2$$

UCT – Assembly Total Peaking Uncertainty

URL – Pin Power Peaking Uncertainty

UCR – Assembly Radial Power Peaking Uncertainty

UCA – Assembly Axial Power Peaking Uncertainty

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

BIAS – Calculation Bias

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

### 3.2. Quadrant Tilt

The excore detector system is used to monitor gross changes in the core power distribution. The primary purpose of the excore detectors with respect to quadrant power tilts is to detect changes in tilt from the previous calibration. Since the Technical Specifications (**References 2, 3, 6 and 7**) allow reactor operations with excore quadrant power tilts up to 2%, the relationship between excore quadrant power tilt and a penalty to apply to the thermal limits calculations had to be determined.

This relationship was determined by evaluating various tilt causing mechanisms for several reactor cores using a full core 3-dimensional core model. The ~~results showed that a~~ <sup>a,c</sup> power peaking **allowance** ~~penalty is required to account for~~ **corresponding to** the allowed 2% excore quadrant power tilt **is contained in the COLR**. This penalty will be applied as TILT to the LOCA, DNB and centerline fuel melt margin calculations in Section 4.



#### 4. LCO AND RPS LIMITS

##### 4.1. General Methodology

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

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

$$\text{MARGIN \%} = (\text{ALLOWED PEAK} - \text{CALCULATED PEAK}) * 100 / \text{ALLOWED PEAK}$$

The calculated peak is obtained directly from SIMULATE-3. Depending on the limit type, this equation may be in terms of a peaking factor or a linear heat rate. Either the calculated peak or the allowed peak would contain sufficient factors to account for the various uncertainties and tolerances. AFD and control rod insertion limits for each limit type are set to exclude all power distributions with negative margins of the same limit type.

##### 4.2. LOCA Margin Calculations

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

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

Where:

LOCAMX(z)	= Maximum allowable linear heat rate in kw/ft. (May be axial and/or burnup dependent)
LHR(x,y,z)	= $NP(x,y,z) * FP * AVGLHR * UCT * TILT * RPF * AMF$
NP(x,y,z)	= Normalized peak local power for each axial elevation, z, from the power distribution calculation.
FP	= Fraction of core power level, including power level uncertainty.
AVGLHR	= Total core power divided by the total length of fuel rods in the core, kw/ft, accounting for fuel densification and thermal expansion.
UCT	= Uncertainty factor on the pin total peak, including engineering hot channel factor, assembly bow and rod bow if not included in the LOCA analysis (see Section 3.1).
TILT	= Factor to account for a peaking increase due to an allowed quadrant tilt (see Section 3.2).
RPF	= Factor to account for the power deposited in the fuel rod.
AMF	= Additional Margin Factor, optionally used to incorporate additional design margin.

The values for LOCAMX(z) are derived from the Core Operating Limits Report (COLR) limits on  $F_Q$ . Typical limiting values are shown in Figure 6.

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

#### **4.3. LOFA DNB Margin Calculations**

The LOFA DNB limits are also used to define the operating limits, so the operating limits power distributions, as described in Section 2.4, are used in this calculation. The DNB margin calculation is based on a set of Maximum Allowed Total Peak (MATP) curves that are calculated with a NRC approved thermal-hydraulic method (References 4 and 14). The MATP curves are determined for several power levels (e.g., 100, 75 and 50% power). The input power distributions are selected to match the power level of each set of MATP curves. Sample MATP curves for LOFA DNB are shown in Figure 10. The DNB margin is computed for each assembly in the core, but only the minimum margin for each power distribution is used in the determination of the AFD - power limits. DNB margin, DNBM, is calculated as:

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

Where:

MARP(x,y)	= MATP(z,AP(x,y))/(AP(x,y)*UCA)
AP(x,y)	= Axial peak in an assembly, on an assembly normalized basis.
UCA	= Assembly axial peak uncertainty factor (see Section 3.1).
MATP(z)	= Maximum allowed total peak, at the axial plane of the axial peak.
RPP(x,y)	= RNP(x,y) * AMF * TILT * UCR
RNP(x,y)	= Normalized radial peak integrated pin power from the power distribution calculation.
UCR	= Uncertainty factor on the pin radial peak, including engineering hot channel factor and rod bow if not included in the DNB analysis (see Section 3.1). An assembly bow uncertainty may also be included in UCR.
AMF	= Additional margin Factor, optionally used to incorporate additional design margin.
TILT	= Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).

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

#### 4.4. RPS DNB Margin Calculations

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

#### 4.5. Centerline Fuel Melt Margin Calculations

The centerline fuel melt limit is also used to validate the RPS limits, so the operating limits restrictions on power distributions are not applied in the calculation. Since there usually is a positive margin for centerline fuel melting, only the power distributions at **the power level assumed in the development of**

**the over-power and over-temperature RPS limits (typically 118% power)** are used for the centerline fuel melt margin calculations. A positive margin at ~~118%~~ **this** power level will preclude negative margins at lower power levels. If the ~~118%~~ **results at this** power level ~~results~~ show negative margins, lower power levels will be analyzed to fully define the **centerline fuel melt** AFD - power level limit. If low margins exist at the ~~118%~~ power level **assumed in the development of the over-power and over-temperature RPS limits**, lower power levels may be evaluated to quantify the trade-off between peaking margin and power level. The equations below show how the margin for centerline fuel melt is calculated. Note that the linear heat rate is calculated similarly to the LOCA margin calculation. Each node in the core model is analyzed, but only the minimum margin for a power distribution is used to determine the **centerline fuel melt** AFD - power level limits **used to develop the OPAT f(ΔI) penalty**. CFM limits are calculated using NRC approved fuel performance codes.

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

Where:

MAXLHR	= Maximum allowable linear heat rate in kw/ft.
LHR(x,y,z)	= NP(x,y,z) * FP * AVGLHR * TILT * RPF * UCT * AMF
NP(x,y,z)	= Normalized peak local power for each axial elevation, z, from the power distribution calculation.
FP	= Fraction of core power level, including power level uncertainty.
AVGLHR	= Total core power divided by the total length of fuel rods in the core, kw/ft, accounting for fuel densification and thermal expansion.
UCT	= Uncertainty factor on the pin total peak, including engineering hot channel factor and rod bow (see Section 3.1).
TILT	= Factor to account for a peaking increase due to an allowable quadrant tilt (see Section 3.2).
RPF	= Factor to account for the power generated in the fuel rod.
AMF	= Additional Margin Factor, optionally used to incorporate additional design margin.

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

#### 4.6. Determining the AFD - Power Level Limits

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

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

The DNB margin calculations performed for the RPS OTΔT AFD Trip penalty,  $f(\Delta I)$ , provide AFD limits [

]<sup>a,c</sup> The power - AFD penalty is determined by selecting the limiting breakpoints and slopes defined by the [ <sup>a,c</sup> The uncertainty associated with the  $f(\Delta I)$  function is combined with the uncertainties of the other OTΔT function input parameters in determining the adjusted  $K_1$  constant in the setpoint equation (References 2, 3, 6 and 7), or the  $f(\Delta I)$  function is adjusted to account for the AFD uncertainties.

The centerline fuel melt protection criterion is associated with the OPΔT Trip  $f(\Delta I)$  penalty function. ~~Since the OPΔT  $f(\Delta I)$  function is usually zero, the check~~ **If the centerline fuel melt check performed at the upper 118% power limit (typically 118%) shows positive margin, the OPΔT  $f(\Delta I)$  is adequate to verify that the penalty is not required.** **For this case, additional checks at a lower power level are not required. However,** ~~should the centerline fuel melt margin calculations result in an AFD limit at the~~ **upper 118% power limit (negative centerline fuel melt margins),** lower power levels would be

analyzed in order to define the **centerline fuel melt** power - AFD penalty. The penalty could then be incorporated into the OPΔT trip function or the required protection could be provided by the OTΔT function.

Appendix B to this report describes the methodology used to develop the OTΔT and OPΔT  $f(\Delta I)$  penalties **for McGuire and Catawba. This same methodology is applicable to Harris and Robinson with the exception that the transient analysis methodologies described in DPC-NE-3000, DPC-NE-3001 and DPC-NE-3002 (References 8, 9 and 10, respectively) are replaced with similar NRC-approved transient analysis methodologies as described in the Harris and Robinson COLRs.**

#### **4.7. Control Rod Insertion Limits**

The rod insertion limits are assumed when the operational AFD - power level limits are set. However, further iteration on the limits may be necessary depending upon results from the evaluation of Updated Final Safety Analysis Chapter 15 accident analyses. Adjustments are made to the rod insertion limits and AFD - power level limits as necessary.

**Table 3**

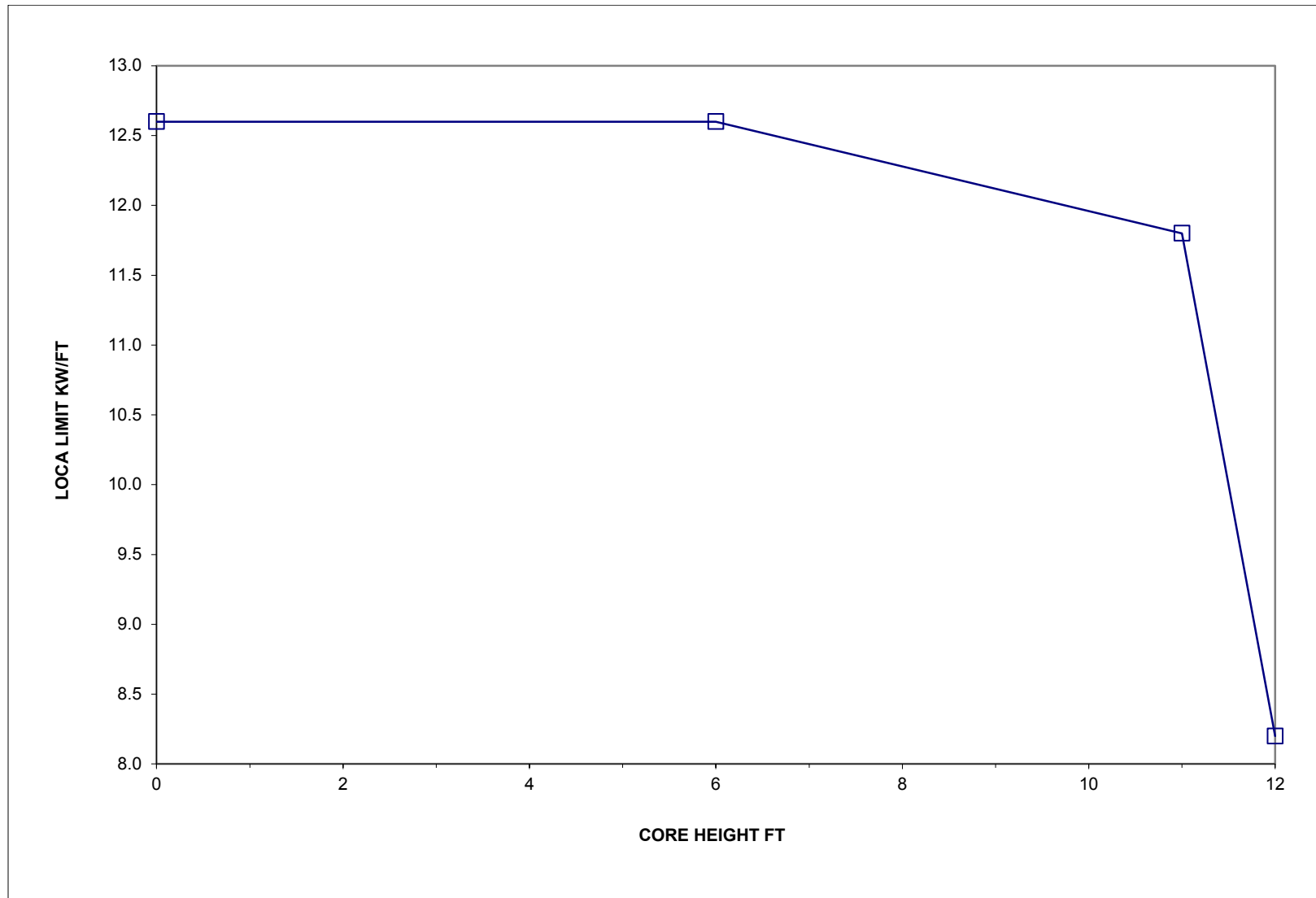
**Typical RPS MATP Curve Conditions  
and Conditions of the Power Distributions  
used for each set of MATP Curves**

[

]

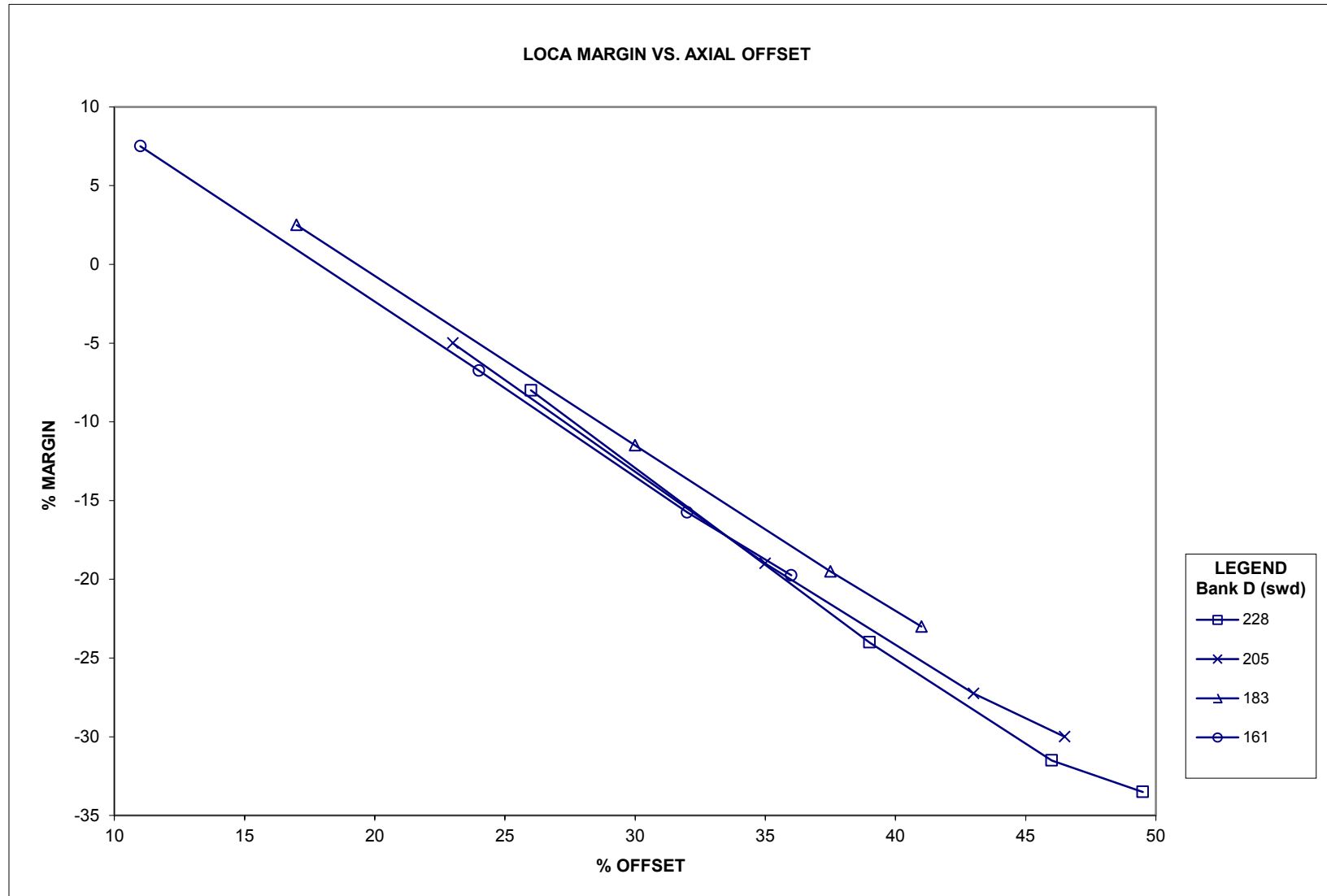
a, c

**Figure 6**  
**Typical LOCA Linear Heat Rate Limits Vs Core Height**

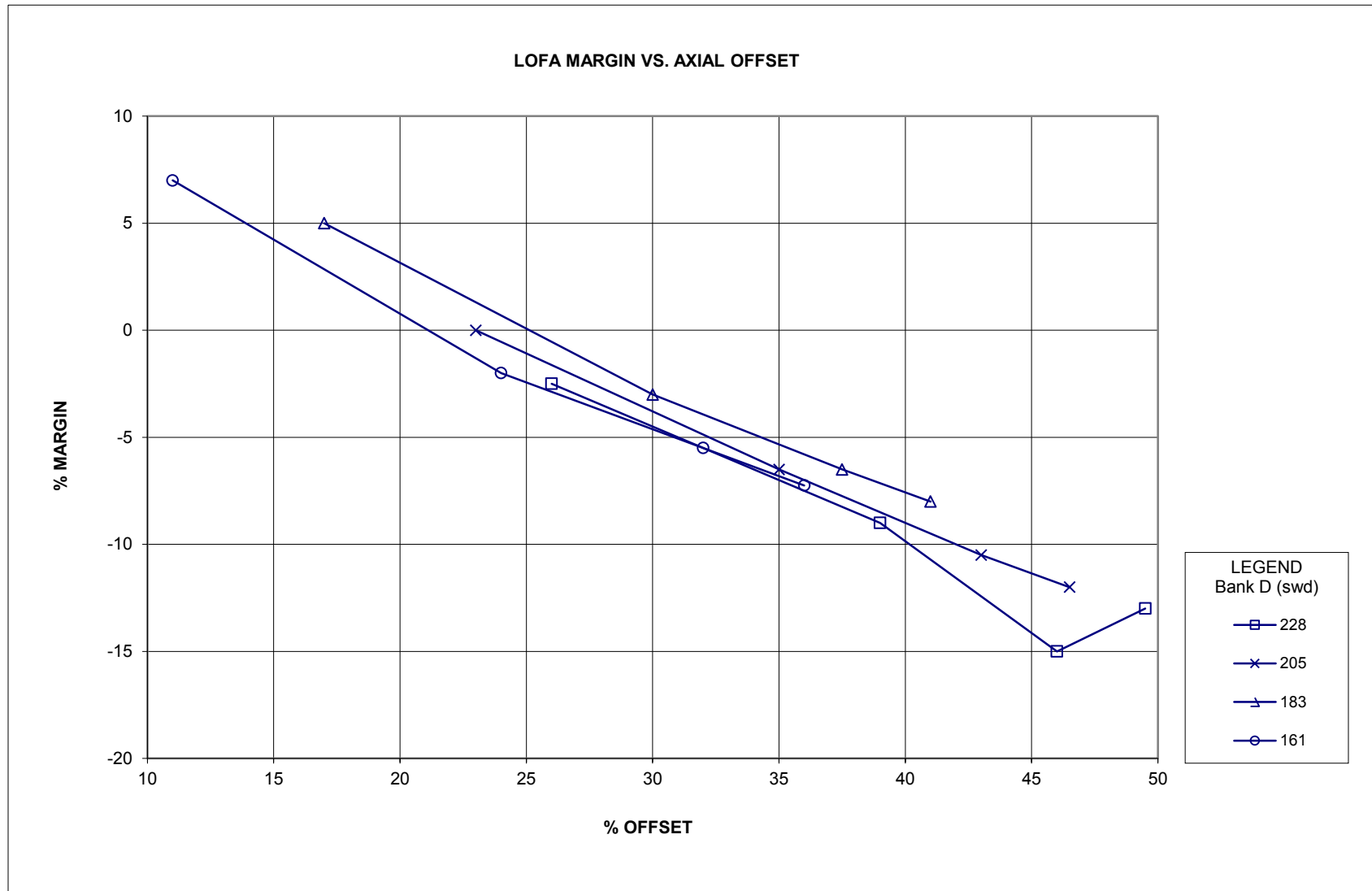




**Figure 7**  
**Sample LOCA Margin Plot**



**Figure 8**  
**Sample LOFA Margin Plot**



## 5. BASE LOAD LCO LIMITS

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

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

**Base load operation option removed in Revision 2 of the report**

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## 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. Similarly,  $f(\Delta I)$  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 LOCA  $F_Q$  limit that must be satisfied within the AFD – power level operating limits is:

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

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

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) specified in the Core Operating Limits Report (COLR). **This If the LOCA limit may be is burnup dependent, the normalized burnup dependency of the limit is specified by the factor  $K(BU)$  in the COLR.**

This criterion is a Technical Specification (**References 2, 3, 6 and 7**) limiting condition for operation (LCO).

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

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

Where:

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

UMT = The measurement uncertainty factor on the total peak, provided in the Core Operating Limits Report (COLR).

MT = Manufacturing tolerance factor (or engineering hot channel factor), provided in the COLR.

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

$F_Q^D(x, y, z)$  =  $NP^D(x, y, z) * UCT$ , design power distribution for  $F_Q$ . **(UCT is defined in section 3.1)**

$NP^D(x, y, z)$  = Design normalized peak local power for each node at axial elevation, z, at equilibrium conditions.

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

a, c

a, c

**6.1.1 F<sub>Q</sub>(x, y, z) Margin Decrease Penalty Factor Methodology**

F<sub>Q</sub>(x, y, z) is measured periodically using the incore detector system to ensure that the value of the total peaking factor, F<sub>Q</sub><sup>RTP</sup>, assumed in the accident analysis is bounding. The frequency requirement for this measurement is 31 effective full power days (EFPD). In order to account for the possibility that F<sub>Q</sub>(x, y, z) may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If extrapolation of the measurement indicates that the F<sub>Q</sub>(x, y, z) measurement would exceed the F<sub>Q</sub>(x, y, z) limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on the available margin, or the F<sub>Q</sub>(x, y, z) measurement is increased by an appropriate penalty and compared against the F<sub>Q</sub>(x, y, z) operational and RPS surveillance limits to ensure allowable total peaking limits are not exceeded.

The F<sub>Q</sub>(x, y, z) penalty factor is calculated by projecting the change in the [

] a, c

a, c

[

] <sup>a,c</sup> The  $F_Q$  margin decrease factor may be applied directly to the measured  $F_Q$  or may be incorporated into the  $M_Q(x, y, z)$  and  $M_C(x, y, z)$  margin factors. For burnup ranges where the  $F_Q$  margin decrease factor is less than 1.02, a value of 1.02 will be maintained. **Cycle-specific  $F_Q$  margin decrease penalty factors are specified in the COLR.**

## 6.2. CFM $F_Q$ Surveillance Methodology

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

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

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

<sup>a, c</sup>



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

The  $F_{\Delta H}$  limits that must be satisfied within AFD – power level operating limits are:

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

Where P is the relative thermal power and RRH is the thermal power reduction required to compensate for each 1% that the measured radial peak exceeds its limit.  $\text{MARP}(x, y)$  is the Maximum Allowed Radial Peak which is derived from the MATP curves (see Figure 10) by dividing the MATP by the axial peak term. MARP limits along with RRH are specified in cycle-specific COLRs. This criterion is a Technical Specification ([References 2, 3, 6 and 7](#)) LCO.

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

$$F_{\Delta H}^M(x, y) * \text{UMR} * \text{TILT} \leq [F_{\Delta H}^D(x, y) * M_{\Delta H}(x, y)]$$

Where:

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

UMR = Uncertainty factor on the measured radial peaks, provided in the Core Operating Limits Report (COLR).

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

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### 6.3.1 $F_{\Delta H}(x, y)$ Margin Decrease Peaking Penalty Factor Methodology

The nuclear enthalpy rise hot channel factor,  $F_{\Delta H}(x, y)$ , is measured periodically using the incore detector system to ensure that fuel design criteria are not violated and accident analysis assumptions are not violated. The frequency requirement for this measurement is 31 effective full power days (EFPD) per Technical Specifications (**References 2, 3, 6 and 7**). In order to account for the possibility that  $F_{\Delta H}(x, y)$  may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If extrapolation of the measurement indicates that the  $F_{\Delta H}(x, y)$  measurement would exceed the  $F_{\Delta H}(x, y)$  surveillance limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval is decreased based on the available margin, or the  $F_{\Delta H}(x, y)$  measurement is increased by an appropriate penalty and compared against the  $F_{\Delta H}(x, y)$  surveillance limit to ensure allowable peaking limits are not exceeded.

The  $F_{\Delta H}(x, y)$  penalty factor is calculated by projecting the change in the [

$$\left[ \frac{F_{\Delta H}(x, y) - F_{\Delta H}(x, y)_{a,c}}{F_{\Delta H}(x, y)_{a,c}} \right]^{a,c}$$

[<sup>a,c</sup> The  $F_{\Delta H}$  margin decrease factor may be applied directly to the measured  $F_{\Delta H}$  or may be incorporated into the  $M_{\Delta H}(x, y)$  margin factors. For burnup ranges where the  $F_{\Delta H}$  margin decrease factor is less than 1.02, a value of 1.02 will be maintained. **Cycle-specific  $F_{\Delta H}$  margin decrease penalty factors are specified in the COLR.**

#### **6.4. Monitoring of Plant Measured Parameters**

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

##### **6.4.1. AFD – Power Level Limits**

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

**Operating AFD – power level limits are specified in the COLR.**

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

#### 6.4.2. Control Rod Insertion Limits

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

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

The in-core instrumentation system is used periodically to measure  $F_Q^M(x, y, z)$ , which must always be within applicable limits. The LOCA limit is specified in the Core Operating Limits Report (COLR).

This limit on  $F_Q^M(x, y, z)$  is a Limiting Condition of Operation, so operation outside of the limit is allowed for a short period of time to allow the operator to bring the reactor back within the limits ~~without a reactor trip~~ **or be placed in MODE 2.**

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

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

$$F_Q^M(x, y, z) \leq F_Q^L(x, y, z)^{RPS}, \text{ or}$$

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

$F_Q^L(x, y, z)^{OP}$  and  $F_Q^L(x, y, z)^{RPS}$  are generated in the maneuvering analysis. These limits are specified in the COLR and are not imposed on the top or bottom **10 - 15%** of the core. The limits on  $F_Q^M(x, y, z)$  account for an appropriate measurement uncertainty, which is provided in the COLR.

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

limits. If  $F_Q^M(x, y, z)$  exceeds  $F_Q^L(x, y, z)^{RPS}$  (CFM limits), then a reduction is made to the OTΔT trip setpoints, or the  $f_1(\Delta I)$  or  $f_2(\Delta I)$  breakpoints are adjusted.

~~$F_Q^{Max BL}(x, y, z)$  for base load operation is calculated using the same equations as  $F_Q^L(x, y, z)^{OP}$  except that the monitor factor,  $M_Q(x, y, z)$ , is calculated at each point in core life over a small AFD band about an AFD target and within a limited power range.~~

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

$$MBLP = \frac{\text{Min Over } (x, y, z)}{F_Q^M(x, y, z)} \frac{F_Q^{MAX BL}(x, y, z) * 100\%}{F_Q^M(x, y, z)}$$

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

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

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

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

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

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

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

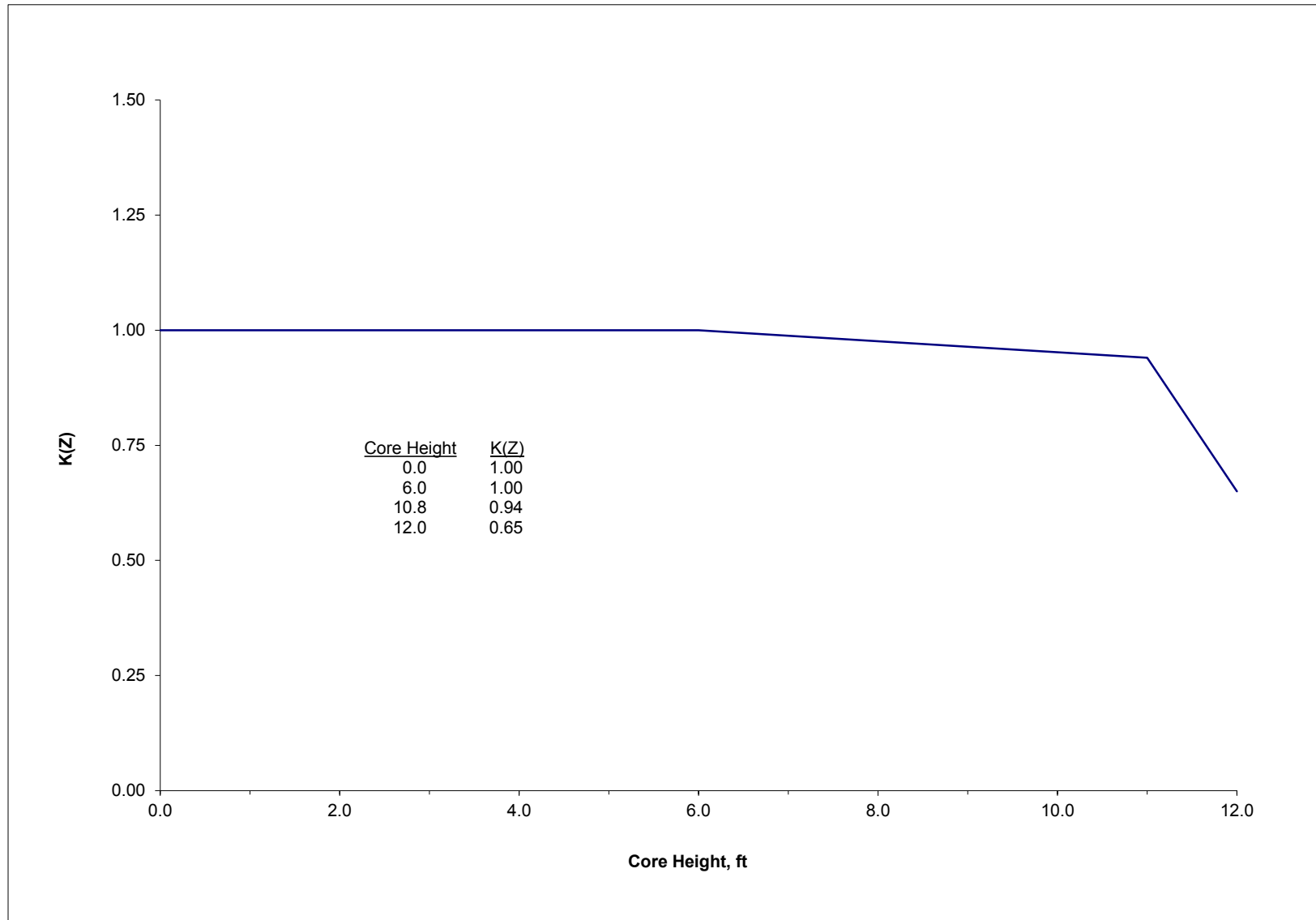
~~$F_{\Delta H}^{\text{Max BL}}(x, y)$  for base load operation is calculated using the same equations as  $F_{\Delta H}^L(x, y)^{\text{SURV}}$  except that the monitor factor,  $M_{\Delta H}(x, y)$ , is calculated at each point in core life at over a small AFD band about an AFD target, and within a limited power range.~~

#### 6.4.5. Quadrant Power Tilt

An allowance for a 2% quadrant power tilt was made in the AFD - power level operating limits **and in the verification of DNB, CFM and LOCA thermal limits.** ~~and in the values of  $F_Q^L(x, y, z)^{\text{OP}}$ ,~~

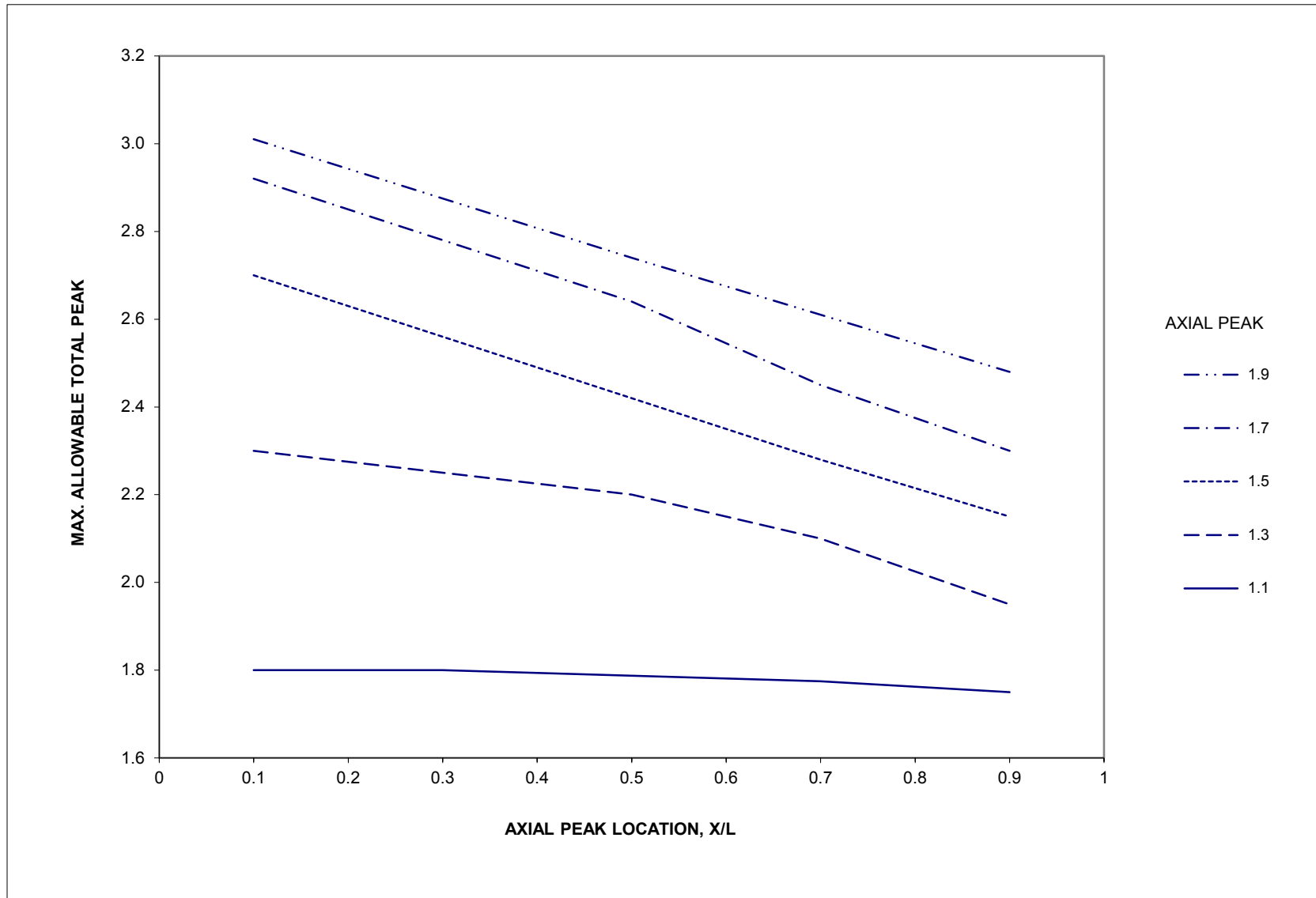
~~$F_Q^L(x, y, z)^{\text{RPS}}$ ,  $F_Q^{\text{Max BL}}(x, y, z)$ ,  $F_{\Delta H}^L(x, y)^{\text{SURV}}$ , and  $F_{\Delta H}^{\text{Max BL}}(x, y)$ .~~ Thus, no action is required for an indicated quadrant power tilt of up to 2%. A quadrant power tilt larger than 2% is a Limiting Condition of Operation, so operation of the plant is allowed to continue for a period of time while the operator attempts to correct the condition.

**Figure 9**  
**Sample  $K(Z)$  – Normalized  $F_Q(Z)$  as a Function of Core Height**

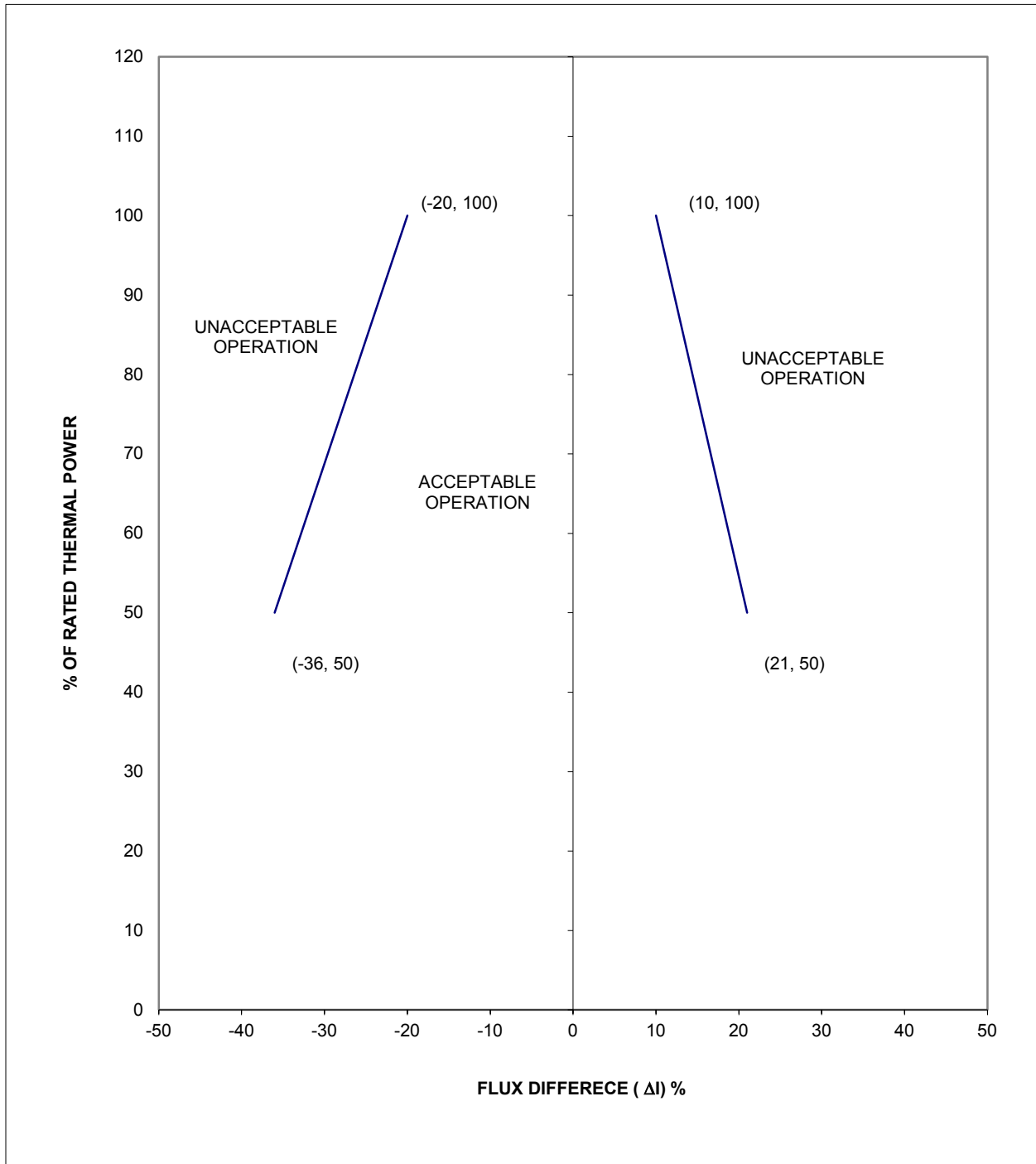




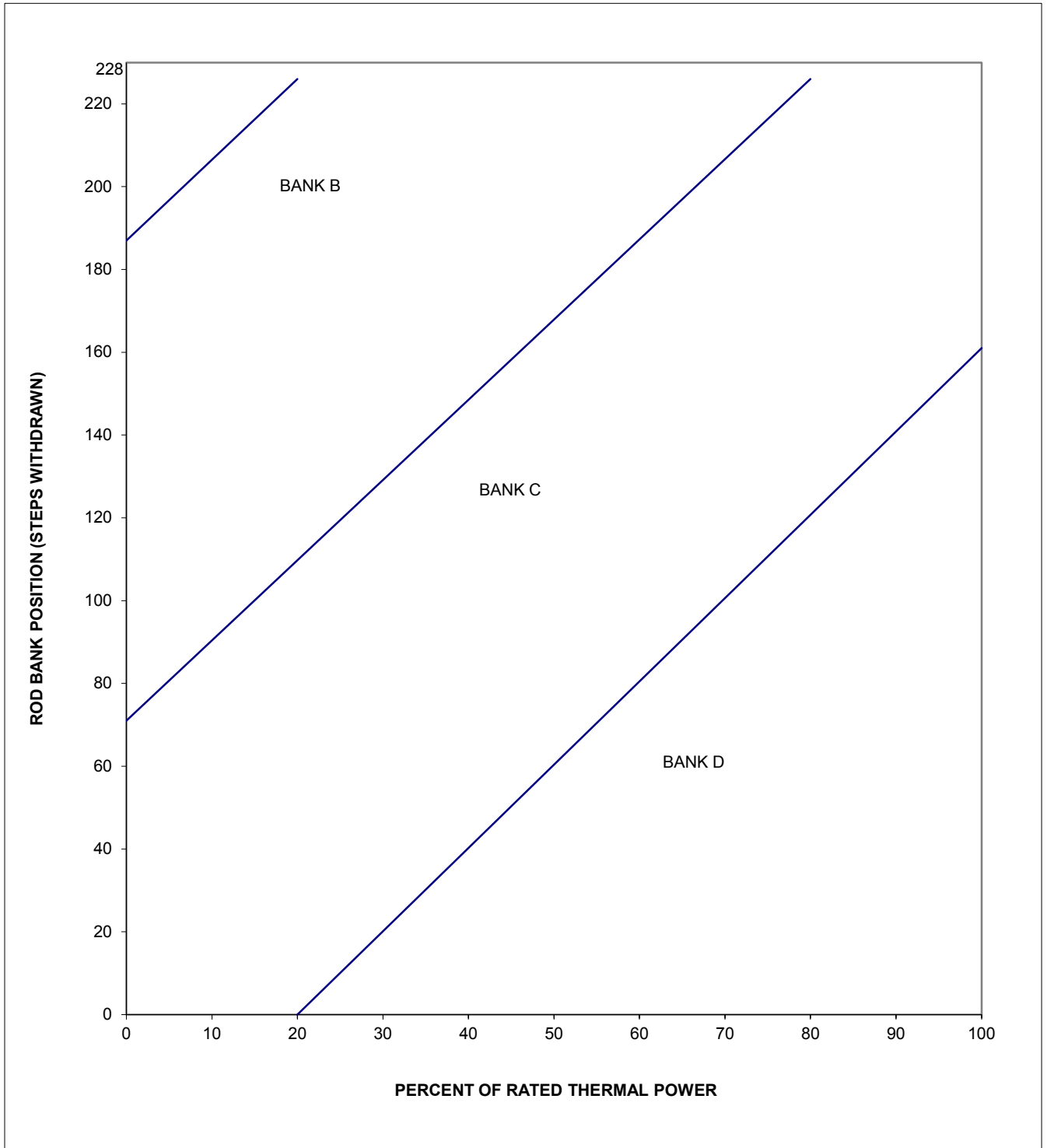
**Figure 10**  
**Sample LOFA DNB MATP Curves for 100% Power**



**Figure 11**  
**Sample AFD – Power Level Operating Space**



**Figure 12**  
**Sample Control Rod Insertion Limits vs. Thermal Power**



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## 7. REFERENCES

1. "Relaxation of Constant Axial Offset Control, F(q) Surveillance Technical Specification", WCAP-10216-PA, June 1983.
2. Technical Specifications for McGuire Nuclear Station Units No. 1 and 2, Docket Nos. 50-369/370.
3. Technical Specifications, Catawba Nuclear Station, Unit Nos. 1 and 2, Docket Nos. 50-413/414.
4. "Duke Power Company, Thermal-Hydraulic Statistical Core Design Methodology", DPC-NE-2005P-A, Revision 35, September 2002 (**Revision 5 Submitted to NRC for Review and Approval**)
5. T. C. McMeekin (Duke) to U. S. Nuclear Regulation Commission, "Supplement to Technical Specification Amendment Relocation of Cycle –Specific Limits to the Core Operating Limits Report (COLR)", April 26, 1993. (Approved in Letter from Victor Nerses (NRC) to T. C. McMeekin (Duke), 'Issuance of Amendments – McGuire Nuclear Station, Units 1 and 2 (TAC NOS. M85474 and M85475)', May 31, 1994.
6. ~~Intentionally Left Blank.~~ **Technical Specifications for Shearon Harris Nuclear Power Plant Unit 1, Docket No. 50-400.**
7. ~~Intentionally Left Blank.~~ **Technical Specifications for H. B. Robinson Steam Electric Plant Unit No. 2, Docket No. 50-261.**
8. ~~Intentionally Left Blank.~~ **"Thermal-Hydraulic Transient Analysis Methodology", DPC-NE-3000-PA, Rev. 5a**
9. ~~Intentionally Left Blank.~~ **"Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology", DPC-NE-3001-PA, Rev. 1**
10. ~~Intentionally Left Blank.~~ **"UFSAR Chapter 15 System Transient Analysis Methodology", DPC-NE-3002-A, Rev. 4b**
11. ~~Intentionally Left Blank.~~ **Robert E. Martin (NRC) to G. R. Paterson (Duke), "McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC NOS. MB8361 and MB8362)", January 14, 2004.**

12. ~~"Computer Code Certification for SMARGINS," DPC internal document.~~ **Sean Peters (NRC) to D. Jamil (Duke), "Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC NOS. MB8359 and MB8360)", December 19, 2003.**
13. ~~"Computer Code Certification for MARGINPLOT," DPC internal document.~~ **Intentionally Left Blank.**
14. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Core Thermal-Hydraulic Methodology using VIPRE-01", DPC-NE-2004-PA, Revision 4**2a**, **December 2008**~~SER~~  
~~Dated February 20, 1997 (DPC Proprietary).~~
15. "Duke Power Company, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX", DPC-NE-1005-P-A, Revision 1, November 2008.
16. **Duke Energy Carolinas, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors", Rev. 0, DPC-NE-1008-P, August 2015. (Submitted to NRC for Review and Approval)**

## APPENDIX A

### Computer Program Description

#### SMARGINS

SMARGINS (12) is a program written by DPC that computes the margin to thermal limits for LOCA  $F_Q$ , DNB and centerline fuel melt. SMARGINS requires three dimensional power distribution data for input. The output of MARGINS is a file that contains one entry per power distribution; the entry contains the case and limit type identifiers, the core axial offset and the core margin to the thermal limit evaluated. The code is also use to generate the  $F_Q$  and FAH power peaking increases that occur over a 31 EFPD Surveillance period.

#### MARGINPLOT

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

#### SIMULATE-3

SIMULATE ([References 15 and 16](#)) is an advanced two-group nodal code written by Studsvik based on the QPANDA neutronics model. SIMULATE computes three dimensional nodal and pin power distributions accounting for fuel and moderator temperature, fuel burnup, xenon distributions, control rods, burnable absorbers, and soluble boron. Cross-section input to SIMULATE is provided from CASMO.

**APPENDIX B**  
**OPDT and OTDT Setpoint Methodology**  
**(NRC Approved From Reference 5)**



**Excerpt from Reference 5**  
**Request for Additional Information**  
**Application to Transfer TS Values to COLR**  
**For Catawba and McGuire Stations**

1. Identify the report providing the methodology for calculating the value of each of the parameters proposed to be transferred to the COLR. If the report discusses the parameters implicitly, provide a reference for the definition of each parameter and a specific discussion of how the parameter values are determined. These parameters should include  $\tau_1, \tau_2, \tau_3, \tau_4, \tau_5, \tau_6, K_1, K_2, K_3, K_4, K_5, K_6, f_1(\Delta I), f_2(\Delta I)$ , the breakpoints and slopes for  $f(\Delta I)$ , BAST volume and concentration, RWST volume and concentration, reactor water makeup pump flow rate, and accumulator boron concentration.

Answer: The methodology described below is how Duke Power Company currently arrives at values for the OT $\Delta$ T and OP $\Delta$ T parameters. This is one of many equally valid methods for determining these parameters. Once a new preliminary set of over temperature and overpower setpoint equation parameters is selected, they must be evaluated by reanalyzing the appropriate transient analyses with the new setpoint parameters. The transient analyses utilized to validate these new setpoints are performed using the NRC approved methodology documented in Duke Power Company topical reports DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," DPC-NE-3001-A, "McGuire/Catawba Nuclear Station Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," and DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology," approved by the NRC for McGuire/Catawba use in November 1991. Once the analysis is performed and the new setpoint constants demonstrate they are capable of protecting the plant during the appropriate transients, the new setpoint parameters may be used. In other words, there are many possible methods for selecting these setpoint parameters and regardless of the method used they are not considered valid parameters until they are proven capable of protecting the plant under transient conditions with the NRC approved methodology described in the topical's above.

The OP $\Delta$ T parameter  $K_5$  is not being relocated to the COLR since it currently is not calculated as part of the reload design methodology described below. **(The  $K_5$  constant was subsequently relocated from Technical Specifications to the COLR for McGuire and Catawba in References 11 and 12.)**

The purpose of the OT $\Delta$ T trip function is to protect the reactor core against DNB and hot leg boiling for any combination of power, pressure and temperature during normal operation and transient conditions. The parameter values for  $K_1, K_2, K_3, K_4, K_6$ , and  $f_1(\Delta I)$  breakpoints and slopes are calculated using as inputs the DNB core limit lines at different pressures, axial offset versus power limits, and various nominal operating condition parameters. For steady state conditions and a reference power shape these inputs are used to calculate the overtemperature and overpower  $\Delta T$  setpoints based on the following constraints:

- Thermal over temperature limits, which provide protection against DNB and hot leg boiling.
- Pressurizes low pressure and high pressure safety limits, which limit the range of pressures over which the overtemperature  $\Delta T$  and overpower  $\Delta T$  trips must function.
- The locus of conditions where the steam generator safety valves open, which places a limit on the primary side temperature based on the steam generator design pressure and the primary to secondary heat transfer capacity.

- Thermal overpower limit, which protects against centerline fuel melt.

Multiple  $K_1$  and  $K_4$  pairs and corresponding  $K_2$ ,  $K_3$ , and  $K_6$  are calculated such that they meet the above constraints. Using engineering judgment and plant operating experience a set of  $K$ s is then chosen from these allowable sets. The chosen set is then evaluated in the transient analysis using the methodology described in topical reports DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology" and DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology," to determine whether the  $K$  parameter values are capable of protecting the plant during the appropriate transients.

The thermal overtemperature limits described above are calculated for zero axial imbalances. Therefore, once the setpoint constants are calculated, the  $f_1(\Delta I)$  trip reset function for the OT $\Delta$ T equation is determined using two axial offset versus power envelopes (typically 100% and 118% power) supplied by nuclear design analyses. This function is determined as described in the DPC-NE-2011-P-A. A value of imbalance,  $\Delta I$ , and a point on the DNB line for a given pressure which is not bounded by the exit boiling line, steam generator safety valve line, or OP $\Delta$ T setpoint equation is selected. This point is compared with the OT $\Delta$ T setpoint and the amount the setpoint must be lowered, if at all, to bound this point is calculated. This process is repeated for this  $\Delta I$  for the other non-bounded DNB points at this and other pressures. The largest reduction in the OT $\Delta$ T setpoint equation required to bound the imbalance corrected DNB points becomes the  $f_1(\Delta I)$  penalty for this particular value of  $\Delta I$ . This process is repeated for a range of  $\Delta I$ s that will envelope all the expected skewed axial power distributions. The  $f_1(\Delta I)$  breakpoints and slopes are then selected in a manner such that they bound the calculated  $f_1(\Delta I)$  penalties which were determined from the two axial offset envelopes.

The purpose of the OP $\Delta$ T trip function is to provide protection against fuel center-line melt (CFM) during normal operation and Condition II Transients. The  $\Delta T$  trip setpoint for this trip function is typically set at 118%FP and is determined as described above. The trip reset portion of this trip function,  $f_2(\Delta I)$ , is designed to lower the trip setpoint when measured imbalances exceed predetermined values. Since highly skewed power distributions lead to high kw/ft values, a  $f_2(\Delta I)$  trip function can be developed to prevent CFM limits from being exceeded at large imbalances, or to increase the available margin to the CFM limit for highly skewed power distributions.

Current core designs do not challenge to CFM limits and therefore a  $f_2(\Delta I)$  penalty is not required. However, from an operational and design standpoint, it is desirable to eliminate from consideration power distributions with high imbalances. Therefore, a  $f_2(\Delta I)$  trip reset function was established to trip the reactor at high imbalances. The breakpoints and slopes of this function were arbitrarily chosen to limit the power distributions that need to be considered during the design of the reactor core and to increase the margin to the CFM limit and therefore reducing the probability of the CFM Technical Specification surveillance limits being violated.

In the event that it would be necessary to establish a  $f_2(\Delta I)$  trip reset function because CFM limits were being exceeded, one possible method of determining this trip function would be to develop a kw/ft versus imbalance envelope based on the analysis of Condition II transient such that this envelope would conservatively bound expected transient peaks. This envelope would next be used to determine the  $f_2(\Delta I)$  penalty as a function of imbalance by comparing the CFM kw/ft limit against the maximum expected peak for a given imbalance. The  $f_2(\Delta I)$  trip reset function would then be developed such that the breakpoints and slopes bound the  $f_2(\Delta I)$  penalties developed from the kw/ft versus imbalance envelope which is generated in a manner similar to the  $f_1(\Delta I)$  reset function described above.

The dynamic terms ( $\tau_1, \tau_2, \tau_3, \tau_4, \tau_5, \tau_6$ ) in the OT $\Delta$ T and OP $\Delta$ T setpoint equations compensate for inherent instrument delays and piping lags between the reactor core and the temperature sensors. Lead-lag and rate-lag compensations are required for the following reasons:

- To offset measured RTD instrumentation time delays.
- To ensure the protection system response time is within the limits required by the accident analyses.

In addition, the dynamic terms are used as noise filters and to decrease the likelihood of an unnecessary reactor trip following a large load rejection.

Models have been created to examine the effects of different sets of  $\tau$  values used in the lead-lag, lag, and rate lag functions of the over temperature and overpower equations. These models are the same as those given in EPRI NP-1850-CCM-A, "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems." Using these models the  $\tau$  values are selected in a manner such that the optimum response of the OT $\Delta$ T and OP $\Delta$ T setpoints to changes in plant variables is obtained while satisfying the transient analyses acceptance criteria. The acceptability of the chosen  $\tau$  values is determined utilizing these same mathematical models, which are also contained in the transient analysis models described in DPC-NE-3000, "Thermal Hydraulic Transient Analysis Methodology."

#### **OP $\Delta$ T and OT $\Delta$ T Setpoint Methodology for Harris and Robinson Nuclear Plants**

**Application of NRC-approved codes and methods for Harris Nuclear Plant (HNP) and Robinson Nuclear Plant (RNP) for some (U)FSAR Chapter 15 transients rely on the OP $\Delta$ T and OT $\Delta$ T reactor trip functions. As with the McGuire and Catawba methodology described in the RAI response above, the methodology for determining the K constants and tau ( $\tau$ ) values used in the OP $\Delta$ T and OT $\Delta$ T trip functions is to choose a set that results in acceptable transient analysis results. There are many possible methods for selecting these setpoint parameters but those parameters are not valid if they are not capable of protecting against specified acceptable fuel design limits (SAFDLs) appropriate to the transient conditions. Since HNP and RNP are Westinghouse designed NSSS similar to MNS and CNS, the overall description above of the delta-T trip functions for MNS and CNS is also pertinent to HNP and RNP. NRC-approved models (eg. RETRAN-3D) are used to evaluate the effects of different set of  $\tau$  values used in the lead-lag, lag and rate lag functions of the over-temperature and over-power equations. The effects of different  $\tau$  values are evaluated to determine the optimum response of the OP $\Delta$ T and OT $\Delta$ T trip functions while satisfying transient analysis acceptance criteria.**

**Technical Justification of Changes for Revision 2 (Redacted)**

## **DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)**

### **DPC-NE-2011-P Changes Constituting Revision 2**

Revision 2 to methodology report DPC-NE-2011-P, “Duke Energy Carolinas Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors”, describes the analysis methodology used to develop core operational axial flux difference (AFD) limits, rod insertion limits (RILs) and  $f(\Delta I)$  limits for the OPDT trip function for Westinghouse nuclear units. The changes made to this report to extend this methodology to Harris and Robinson and present several methodology changes.

The format of the report was updated to the current version of WORD 2010 including a font change from “Courier” to “Times New Roman”. These changes resulted in pagination changes and consequently changes to page numbers included in the Table of Contents. Each of the revisions are described along with each changes technical justification. The identification of the report location of each change *refers back to the Revision 1a page numbering*.

**DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)****1.0 Administrative Changes (Report Cover, Table of Contents, etc)****Change Intro-1: Report Cover and New Admin Page**

Description of Change: The report date, revision number and engineering division name is updated. Duke Energy Progress, INC is added as an applicable company affiliated with this report. The proprietary notice on the cover page is also revised and a new page with additional proprietary information is added as page “i”.

Technical Justification: Editorial.

**Change Intro-2: Statement of Disclaimer (page ii)**

Description of Change: Duke Energy Carolinas, LCC is replaced with Duke Energy and its subsidiaries to remove the specificity that the disclaimer is only applicable to Duke Energy Carolinas, LLC

There are no warranties expressed, and no claims of content accuracy implied. Duke Energy ~~Carolinas, LLC~~ **and its subsidiaries** disclaims any loss or liability, either directly or indirectly as a consequence of applying the information presented herein, or in regard to the use and application of the before mentioned material. The user assumes the entire risk as to the accuracy and the use of this document.

Technical Justification: Editorial.

**Change Intro-3: Revision History (pages iii - iv)**

Description of Change: Content that is no longer applicable is deleted from the Revision 1 description.

Technical Justification: Editorial.

**Change Intro-4: Revision History (page iv)**

Description of Change: The purpose of revision 2 of this report is added to the revision history.

Technical Justification: Editorial.

**Change Intro-5: Table of Contents, List of Figures and List of Tables**

Description of Change: Page numbers are updated as needed.

Technical Justification: Editorial.

## DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)

### 2.0 Section 1 Changes

#### Change 1-1: Purpose (Section 1.1, paragraph 1, page 1-1)

Description of Change: Specified that the report methodology (following NRC approval) is applicable to the McGuire, Catawba, Harris and Robinson nuclear units.

This report describes the methodology for performing a maneuvering analysis for ~~four-loop, 193-fuel assembly~~ **Duke Energy's** Westinghouse reactors, **encompassing the** ~~such as McGuire, and Catawba, Harris and Robinson Nuclear Stations~~ **units.**

Technical Justification: Editorial.

#### Change 1-2: Purpose (Section 1.1, paragraph 1, page 1-1)

Description of Change: Remove reference to Duke Energy Carolinas.

Duke Energy ~~Carolinas~~ has developed this methodology as an alternative to the existing Relaxed Axial Offset Control (RAOC) Methodology (1). This maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of Duke Energy's ~~Carolinas~~ nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques.

Technical Justification: Editorial.

#### Change 1-3: Summary of Methods (Section 1.2, first paragraph, page 1-1)

Description of Change: A clarification is made that the limiting Condition II transient may not be the loss of flow accident.

The operating limits define the AFD - power level space and rod insertion limits which provide assurance that the peak local power in the core is not greater than that assumed in the analysis of design basis accidents or transients (loss of coolant accident (LOCA), ~~or~~ loss of flow accident (LOFA)<sup>1</sup> **or other limiting Condition II transient where the initial condition peaking does not change as the result of the event**). The loss of flow accident is **typically** the limiting departure from nucleate boiling Condition II transient (transient of moderate frequency). Operating the reactor within the allowed AFD - power level window and rod insertion limits satisfies the power peaking assumptions of the LOCA and LOFA analyses.

<sup>1</sup>**Referral to the LOFA transient or LOFA limits in the remainder of the report is implied to mean the limiting Condition II transient where the initial condition peaking does not change as a result of the event.**

Technical Justification: The loss of flow accident (LOFA) has historically been the limiting Condition II transient for McGuire and Catawba. However, current AREVA analyses indicate that the loss of internal load transient is slightly more limiting than the LOFA. As a result, the write-up is modified to indicate the LOFA may not be the limiting Condition II transient. Development of the AFD - power level window and rod insertion limits would be performed using the limiting Condition II transient where initial condition peaking does not change as a result of the event.

## DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)

### Change 1-4: Summary of Methods (Section 1.2, first paragraph, page 1-2)

Description of Change: The word “centerline” is added to describe the type of fuel melting that RPS limits protect against.

The RPS limits, among other functions, provide protection against fuel failure due to **centerline** fuel melting (CFM) or departure from nucleate boiling (DNB) during anticipated transients.

Technical Justification: Editorial.

### Change 1-5: Summary of Methods (Section 1.2, second paragraph, page 1-2)

Description of Change: A clarification was added to the fourth sentence to define the report location of appropriate uncertainties.

Appropriate uncertainty factors **as described in section 4 of this report** are applied to the calculated power distributions which are then evaluated against the various thermal limits

Technical Justification: Clarification.

### Change 1-6: Applicability of the Method (Section 1.3, first paragraph, page 1-2)

Description of Change: The description of the analytical method is modified to extend its applicability to both 3-loop and 4-loop reactors comprising the McGuire, Catawba, Harris and Robinson units.

The maneuvering analysis presented in this report applies to Westinghouse four loop **and three loop,** ~~193 assembly~~ reactors **at the McGuire, Catawba, Harris and Robinson nuclear sites.** This method is ~~intended to be~~ used to set or validate the AFD - power level operating limits, the control rod insertion limits, and the RPS trip limits.

Technical Justification: Editorial.

### Change 1-7: Applicability of the Method (Section 1.3, second paragraph, page 1-2), References (Section 7), Appendix A,

Description of Changes: (1) Deleted the SMARGINS and MARGINPLOT codes from the list of computer codes. Deleted References 12 and 13.

(2) Reference 16 (DPC-NE-1008-P, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors”) is added to the list of NRC approved design codes.

*Section 1.3, Applicability of the Method:*

**Implementation of the method requires the use of a NRC-approved three-dimensional core simulator capable of calculating assembly average and pin power distributions. The CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 code is used for this purpose.** ~~A system of computer programs is used to implement this method. A description of the computer programs currently in use is contained in Appendix A. This list includes both the major design~~ **These** codes



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were approved by the NRC (15) in References 15 and 16 and minor codes that are used for post-processing data. A description of the SIMULATE-3 code is contained in Appendix A, and in References 15 and 16. Post processing codes are used to process data generated by the core simulator to calculate margins to thermal limits.

### Section 7, References:

12. ~~"Computer Code Certification for SMARGINS," DPC internal document.~~ **Intentionally Left Blank.**
13. ~~"Computer Code Certification for MARGINPLOT," DPC internal document.~~ **Intentionally Left Blank.**

### Appendix A:

#### SMARGINS

SMARGINS (12) is a program written by DPC that computes the margin to thermal limits for LOCA  $F_Q$ , DNB and centerline fuel melt. SMARGINS requires three dimensional power distribution data for input. The output of MARGINS is a file that contains one entry per power distribution; the entry contains the case and limit type identifiers, the core axial offset and the core margin to the thermal limit evaluated. The code is also use to generate the  $F_Q$  and FAH power peaking increases that occur over a 31 EFPD Surveillance period.

#### MARGINPLOT

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

#### SIMULATE-3

SIMULATE (References 15 and 16) is an advanced two-group nodal code written by Studsvik based on the QPANDA neutronics model. SIMULATE computes three dimensional nodal and pin power distributions accounting for fuel and moderator temperature, fuel burnup, xenon distributions, control rods, burnable absorbers, and soluble boron. Cross-section input to SIMULATE is provided from CASMO.

### Technical Justification:

- (1) Implementation of the method requires a three-dimensional core simulator capable of calculating assembly average and pin power distributions and some auxiliary post processing codes to process and evaluate core power distribution information and to calculate margin to thermal limits. The post processing codes used to process, calculate thermal margins and plot data are simply a tool to efficiently implement the methodology described in this report. They are not part of the methodology and are therefore removed from the licensing basis.
- (2) Reference 16 describes the nuclear design analysis methodology using CASMO-5/SIMULATE-3. This is the code system that will be used to perform neutronics calculations for Harris and Robinson. Future use of this code system for McGuire and Catawba is planned. The methodology report referenced (DPC-NE-1008-P, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors") is currently under NRC review. The DPC-NE-2011-P methodology for Harris and Robinson requires approval of the DPC-NE-1008-P report prior to its use.

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**Change 1-8: Definition of Terms (Section 1.4, MARP definition)**

Description of Change: The following clarification is added to the MARP definition: “Axial peak is defined as the ratio of  $F_Q$  divided by  $F\Delta H$ ”.

Maximum Allowed Radial Peak values derived from MATP values by dividing the MATP by the axial peak. **Axial peak is defined as the ratio of  $F_Q$  divided by  $F\Delta H$ .**

Technical Justification: Clarification. The axial peak term was not previously defined.

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### 3.0 Section 2 Changes

#### Change 2-1: Description of Models Used (Section 2.1, first paragraph, page 2-1)

Description of Change: Reference 16 is added to the list of NRC approved nuclear design codes.

The three dimensional xenon and local peak pin power distributions are generated with SIMULATE-3. The SIMULATE-3 model was approved for use in reload core design analyses in References 15 and 16. A description of the SIMULATE-3 model is presented in these References 15.

Technical Justification: See change 1-7.

#### Change 2-2: Time in Core Life (Section 2.2, first paragraph, page 2-1)

Description of Change: The times in life where the maneuvering analysis is performed is modified.

The maneuvering analysis is typically performed at [ ]<sup>a,c</sup>

Technical Justification: The approach specified in the original methodology stated that the maneuvering analysis is typically performed [ ]

[ ]<sup>a,c</sup> This allows the analysis to capture reactivity peaks associated with burnable poison burnout which depending upon the burnable poison type occurs early in cycle (e.g. Westinghouse's zirconium diboride integral fuel burnable absorbers) and true middle of cycle conditions.

#### Change 2-3: Generation of Abnormal Xenon Distributions (Section 2.3.1 third paragraph, page 2-2)

Description of Change: The use of control rod insertion limits in the generation of the xenon transients is updated based on additional experience with applying the methodology.

The control rod positions for the xenon transients were chosen to be at or near the expected rod insertion limits. **Control rods are inserted to the control rod insertion limit [ ]**

[ ]<sup>a,c</sup> ~~The final control rod insertion limits may be different from the positions used in the xenon transients and the analysis will still be valid. This is because the xenon transients are so severe that the maneuvering analysis results are not sensitive to the control rod motions that drive the xenon transients.~~

Technical Justification: The proposed change states that control rod positions chosen for the xenon transients will be inserted to the control rod insertion limit [ ]

[ ]<sup>a,c</sup> Core operational AFD limits are the limits established to preclude core power distributions from exceeding thermal limits during normal operation and anticipated transients. The AFD space provides a bound on power distributions used as initial conditions for analysis of Condition II, III and IV transients. [ ]

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$I^{a,c}$

Figure 1 shows results from a sensitivity study where the Control Bank D position used to establish the initial condition equilibrium xenon condition is varied [

$I^{a,c}$

a.c

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a,c

### Change 2-4: Generation of Power Distributions (Section 2.4 first paragraph, page 2-3)

Description of Change: A clarification is made that the limiting Condition II transient may not be the loss of flow accident.

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

] <sup>a,c</sup> The operating limits are pre-conditions that would prevent exceeding the peak local power in the core assumed in the loss of coolant accident (LOCA) analysis or the loss of flow accident (LOFA, or ~~a primary coolant pump trip~~ **another limiting Condition II transient where the initial condition peaking does not change as the result of the event**) analysis.

Technical Justification: Refer to the technical justification for change 1-3.

### Change 2-5: Generation of Power Distributions (Section 2.4, paragraph 3, page 2-3)

Description of Change: The referral to control rod motion in 50% overlap mode is removed.

The limit of the control rod motion for [

] <sup>a,c</sup>

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Technical Justification: Bank overlap between the 3-loop and 4-loop control rod systems are slightly different – 116 SWD for McGuire and Catawba and 128 SWD for Harris and Robinson. Removal of the overlap value avoids implication that control banks move exactly in 50% overlap.

### **Change 2-6: Generation of Power Distributions (Section 2.4, first paragraph, page 2-4)**

Description of Change: Two clarifications are made. The boron dilution event described is for MODE 1 and the basis for termination is updated.

[

] <sup>a,c</sup>

Technical Justification: The above change is made because the licensing basis for the termination of the boron dilution event is dependent upon each plants licensing basis.

### **Change 2-7: Generation of Power Distributions (Section 2.4, second paragraph, page 2-4)**

Description of Change: The specificity of the power level where the overcooling transient is terminated is removed.

During an [

] <sup>a,c</sup>

Technical Justification: The reference to the specific power level where the RPS trips the reactor varies from plant to plant and for this reason is deleted. The power level where the RPS would trip the reactor is based on the power level assumed in the development of the over-temperature and over-power delta-T RPS trip functions. This power level is 118% for McGuire and Catawba, but is 116% for Harris and Robinson.

## DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)

### 4.0 Section 3.0 Changes

#### Change 3-1: Power Distributions (Section 3.1, second paragraph, page 3-1)

Description of Change: Peaking uncertainties produced from the CASMO-5/SIMULATE-3 benchmarks performed in DPC-NE-1008-P (Ref. 16) are added.

References 15 and 16 presents calculation~~ed~~ peaking uncertainties based on the benchmarking analysis of measured to predicted power distribution **for the CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 methodologies**. The peaking uncertainty factor is calculated as described below.

Technical Justification: CASMO-5/SIMULTE-3 benchmark calculations were performed in DPC-NE-1008-P, Reference 16. DPC-NE-1008-P contains the peaking factor uncertainties that are appropriate for use with this code system. Reference 15 contains the peaking uncertainties applicable to the CASMO-4/SIMULATE-3 codes system.

#### Change 3-2: Quadrant Tilt (Section 3.2, first paragraph, page 3-2)

Description of Change: The Harris and Robinson Technical Specifications (References 6 and 7) are added as allowing operation with an excore quadrant power tilt of up to 2.0%.

The excore detector system is used to monitor gross changes in the core power distribution. The primary purpose of the excore detectors with respect to quadrant power tilts is to detect changes in tilt from the previous calibration. Since the Technical Specifications (**References 2, 3, 6 and 7**) allow reactor operations with excore quadrant power tilts up to 2%, the relationship between excore quadrant power tilt and a penalty to apply to the thermal limits calculations had to be determined.

Technical Justification: Both the Harris and Robinson Technical Specifications (Reference 6 and 7), like McGuire and Catawba Technical Specifications, allow quadrant power tilts of up to 2% prior to requiring reactor operators to perform actions.

#### Change 3-3: Quadrant Tilt (Section 3.2, second paragraph, page 3-2)

Description of Change: A clarification is made that the excore quadrant power tilt penalty is specified in the core operating limits report.

This relationship was determined by evaluating various tilt causing mechanisms for several reactor cores using a full core 3-dimensional core model. The ~~results showed that a { }<sup>a,c</sup> power peaking allowance~~ penalty is required to account for **corresponding to** the allowed 2% excore quadrant power tilt **is contained in the COLR**. This penalty will be applied as TILT to the LOCA, DNB and centerline fuel melt margin calculations in Section 4.

Technical Justification: The value of the excore quadrant power tilt allowance is specified in cycle-specific core operating limits reports. This calculation input is used to account for a 2% excore quadrant power tilt allowed by Technical Specifications.

## DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)

### 5.0 Section 4.0 Changes

#### Change 4-1: General Methodology (Section 4.1, first paragraph, page 4-1), Centerline Fuel Melt Margin Calculations (Section 4.5, first paragraph, page 4-4)

Description of Change: The reason for the maximum power level assumed in the Maneuvering Analysis is provided.

##### *Section 4.1 Change:*

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

##### *Section 4.5 Change:*

The centerline fuel melt limit is also used to validate the RPS limits, so the operating limits restrictions on power distributions are not applied in the calculation. Since there usually is a positive margin for centerline fuel melting, only the power distributions at **the power level assumed in the development of the over-power and over-temperature RPS limits (typically 118% power)** are used for the centerline fuel melt margin calculations. A positive margin at ~~118%~~ **this power level** will preclude negative margins at lower power levels. If the ~~118%~~ **results at this power level** results show negative margins, lower power levels will be analyzed to fully define the **centerline fuel melt AFD - power level limit**. If low margins exist at the ~~118%~~ power level **assumed in the development of the over-power and over-temperature RPS limits**, lower power levels may be evaluated to quantify the trade-off between peaking margin and power level. The equations below show how the margin for centerline fuel melt is calculated. Note that the linear heat rate is calculated similarly to the LOCA margin calculation. Each node in the core model is analyzed, but only the minimum margin for a power distribution is used to determine the **centerline fuel melt AFD - power level limits used to develop the OPΔT f(ΔI) penalty**. CFM limits are calculated using NRC approved fuel performance codes.

Technical Justification: The specific reference to 118% power level for CFM calculations is removed. The upper power level evaluated in the verification of RPS limits is based on the power level assumed as the basis for the over-power and over-temperature delta temperature (OPΔT and OTΔT) trip functions. This power level varies based on reactor type. Any change to the power level assumed in the development of these trip functions would require a commensurate change to the upper power level assumed in the maneuvering analysis.

The section 4.5 change also has a clarification to define the type of AFD versus power level limits (i.e. centerline fuel melt AFD - power level limits) to delineate between operational limits and those used for development of the f(ΔI) trip reset penalty associated with the over-power trip function.



## DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)

### Change 4-2: Determine the AFD – Power Level Limits (Section 4.6, last sentence in second paragraph on page 4-6)

Description of Change: References for the Harris and Robinson Technical Specifications (Reference 6 and 7) are added.

The uncertainty associated with the  $f(\Delta I)$  function is combined with the uncertainties of the other OTAT function input parameters in determining the adjusted  $K_1$  constant in the setpoint equation (References 2, 3, 6 and 7), or the  $f(\Delta I)$  function is adjusted to account for the AFD uncertainties.

Technical Justification: Extension of the DPC-NE-2011-P methodology to the Harris and Robinson reactors requires addition of the Harris and Robinson Technical Specifications for the OTAT setpoint equation.

### Change 4-3: Determine the AFD – Power Level Limits (Section 4.6, third paragraph on page 4-6)

Description of Change: A clarification is made that the OPAT  $f(\Delta I)$  function is usually not zero.

The centerline fuel melt protection criterion is associated with the OPAT Trip  $f(\Delta I)$  penalty function. ~~Since the OPAT  $f(\Delta I)$  function is usually zero, the check~~ **If the centerline fuel melt check** performed at ~~the upper 118% power limit (typically 118%) shows positive margin, the OPAT  $f(\Delta I)$  is adequate to verify that the~~ **penalty is not required; For this case, additional checks at a lower power level are not required. However,** should the centerline fuel melt margin calculations result in an AFD limit **(negative centerline fuel melt margins)** at ~~the upper 118% power limit~~, lower power levels would be analyzed in order to define the **centerline fuel melt** power - AFD penalty. The penalty could then be incorporated into the OPAT trip function or the required protection could be provided by the OTAT function

Technical Justification: The OPAT  $f(\Delta I)$  function at the inception of this methodology was not typically required to provide centerline fuel melt (CFM) protection. However, with changes in fuel design and fuel management strategies, and because of fuel thermal conductivity degradation, a non-zero OPAT  $f(\Delta I)$  function is typically used. The use of the OPAT  $f(\Delta I)$  function eliminates axial power shapes at high imbalances which increases margin to CFM limit, and reduces the probability of exceeding CFM Technical Specification surveillance limits. An increase in DNB margin is also realized if the OPAT  $f(\Delta I)$  penalty is more restrictive than the OTAT  $f(\Delta I)$  penalty.

References to the 118% power level are removed for the reasons stated in change 2-7.

### Change 4-4: Determine the AFD – Power Level Limits (Section 4.6, last paragraph on page 4-6), Appendix B, last page

Description of Change: The Duke McGuire and Catawba methodology used to calculate the following OTAT and OPAT parameters is extended to Harris and Robinson:  $\tau_1$ ,  $\tau_2$ ,  $\tau_3$ ,  $\tau_4$ ,  $\tau_5$ ,  $\tau_6$ ,  $K_1$ ,  $K_2$ ,  $K_3$ ,  $K_4$ ,  $K_5$ ,  $K_6$ ,  $f_1(\Delta I)$ ,  $f_2(\Delta I)$ , the breakpoints and slopes for  $f(\Delta I)$ . References 8 through 10 were added to Section 7. A short write-up was also added to the end of Appendix B describing applicability to Harris and Robinson.

*Section 4.6 Change:*

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Appendix B to this report describes the methodology used to develop the OTΔT and OPΔT  $f(\Delta I)$  penalties for McGuire and Catawba. This same methodology is applicable to Harris and Robinson with the exception that the transient analysis methodologies described in DPC-NE-3000, DPC-NE-3001 and DPC-NE-3002 (References 8, 9 and 10, respectively) are replaced with similar NRC-approved transient analysis methodologies as described in the Harris and Robinson COLRs.

### *Appendix B Addition:*

#### **OPΔT and OTΔT Setpoint Methodology for Harris and Robinson Nuclear Plants**

Application of NRC-approved codes and methods for Harris Nuclear Plant (HNP) and Robinson Nuclear Plant (RNP) for some (U)FSAR Chapter 15 transients rely on the OPΔT and OTΔT reactor trip functions. As with the McGuire and Catawba methodology described in the RAI response above, the methodology for determining the K constants and tau ( $\tau$ ) values used in the OPΔT and OTΔT trip functions is to choose a set that results in acceptable transient analysis results. There are many possible methods for selecting these setpoint parameters but those parameters are not valid if they are not capable of protecting against specified acceptable fuel design limits (SAFDLs) appropriate to the transient conditions. Since HNP and RNP are Westinghouse designed NSSS similar to MNS and CNS, the overall description above of the delta-T trip functions for MNS and CNS is also pertinent to HNP and RNP. NRC-approved models (eg. RETRAN-3D) are used to evaluate the effects of different set of  $\tau$  values used in the lead-lag, lag and rate lag functions of the over-temperature and over-power equations. The effects of different  $\tau$  values are evaluated to determine the optimum response of the OPΔT and OTΔT trip functions while satisfying transient analysis acceptance criteria.

Technical Justification: Information contained in Appendix B of DPC-NE-2011-P is from a NRC staff request for additional information and clarifications associated with Duke's request to relocate cycle-specific limits to the COLR. Duke's responses are included in the following letter:

T. C. McMeekin to U.S. Nuclear Regulatory Commission, "McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2 Docket Nos. 50-413 and 50-414, Supplement to Technical Specification Amendment Relocation of Cycle-specific Limits to Core Operating Limits Report (COLR) ", April 26, 1993

The information in the NRC response included in Appendix B of DPC-NE-2011-P is applicable to the Harris and Robinson nuclear units with the exception that the methodologies documented DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology", DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology", and DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology", will be replaced with similar NRC-approved Methods.

**DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)****6.0 Section 5.0 Changes****Change 5-1 Base Load LCO Limits (Section 5.0, page 5-1)**

Description of Change: The base load operational mode is removed.

Technical Justification: The base load limiting condition of operation (LCO) operating philosophy is a constant axial offset mode of operation about a target predicted target AFD. The margin benefit from this mode of operation can alternately be accomplished by decreasing the severity of the xenon transients and rod insertion limits to allow operation within a smaller AFD band. In addition, current Technical Specifications at McGuire and Catawba and those proposed to be submitted to the NRC for Harris and Robinson do not support this mode of operation. For these reasons, the base load operational mode is being deleted.

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### 7.0 Section 6.0 Changes

#### Change 6-1: LOCA $F_Q$ Surveillance Methodology (Section 6.1, $F_Q^{RTP}$ definition, page 6-1)

Description of Change: The factor  $K(BU)$  is defined.

$F_Q^{RTP}$  = the LOCA limit at rated thermal power (RTP) specified in the Core Operating Limits Report (COLR). ~~This~~ **If the LOCA limit may be is burnup dependent, the normalized burnup dependency of the limit is specified by the factor  $K(BU)$  in the COLR.**

Technical Justification: Fuel thermal conductivity degradation has resulted in the LOCA limit being burnup dependent if the effect of TCD cannot be accommodated by margin in the fuel vendor's LOCA analysis. The burnup dependency of the LOCA limit is specified as the factor  $K(BU)$ , which is the normalized LOCA limit,  $F_Q^{RTP}$ , as a function of burnup.  $K(BU)$  is applied as a direct multiplier to  $F_Q^{RTP}$ , and is specified in the Core Operating Limits Report.

#### Change 6-2: LOCA $F_Q$ Surveillance Methodology (Section 6.1, second paragraph, page 6-1), LOFA DNB $F_{\Delta H}$ Surveillance Methodology (Section 6.3, second paragraph, page 6-5), and $F_{\Delta H}(x, y)$ Margin Decrease Peaking Penalty Factor Methodology (Section 6.3.1, first paragraph, page 6-7)

Description of Change: References for the Harris and Robinson Technical Specifications (References 6 and 7) are added.

##### *Section 6.1 Change:*

This criterion is a Technical Specification (**References 2, 3, 6 and 7**) limiting condition for operation (LCO).

##### *Section 6.3 Change:*

This criterion is a Technical Specification (**References 2, 3, 6 and 7**) LCO.

##### *Section 6.3.1 Change:*

The frequency requirement for this measurement is 31 effective full power days (EFPD) per Technical Specifications (**References 2, 3, 6 and 7**).

Technical Justification: Extension of the DPC-NE-2011-P methodology to the Harris and Robinson reactors requires the addition of the Harris and Robinson Technical Specifications for the  $F_Q$  and  $F_{\Delta H}$  LCO criteria (sections 6.1 and 6.3), and for the frequency requirement for  $F_{\Delta H}$  measurements of 31 effective full power days (section 6.3.1).

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**Change 6-3: LOCA  $F_Q$  Surveillance Methodology (Section 6.1, pages 6-1 to 6-3), CFM  $F_Q$  Surveillance Methodology (Section 6.2, pages 6-4 and 6-5), LOFA DNB  $F_{\Delta H}$  Surveillance Methodology (Section 6.3, pages 6-6 and 6-7)**

Description of Change: The TILT parameter is removed from the LOCA  $F_Q$ , CFM  $F_Q$  and LOFA  $F_{\Delta H}$  surveillance equations.

*Section 6.1, equation on page 6-1 and removal of TILT definition:*

$$F_Q^M(x, y, z) * UMT * MT * \cancel{TILT} \leq [F_Q^D(x, y, z) * M_Q(x, y, z)]$$

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

*Section 6.1, text and equation on page 6-3:*

In application to power distribution surveillance,  $F_Q^D$  and  $M_Q$  are combined, along with UMT, **and** MT ~~and TILT~~, to make a maximum allowed  $F_Q^L(x, y, z)^{OP}$  term, which will be interpolated for each x,y,z location based on cycle burnup and power level:

$$F_Q^L(x, y, z)^{OP} = F_Q^D(x, y, z) * M_Q(x, y, z) / (UMT * MT * \cancel{TILT})$$

*Section 6.2, equation on page 6-4:*

$$F_Q^M(x, y, z) * UMT * MT * \cancel{TILT} \leq [F_Q^D(x, y, z) * M_C(x, y, z)]$$

*Section 6.2, text and equation on page 6-5:*

In application to power distribution surveillance,  $F_Q^D$  and  $M_C$  are combined, along with UMT, **and** MT ~~and TILT~~, to make a maximum allowed  $F_Q^L(x, y, z)^{RPS}$  term, which will be interpolated for each x,y,z location based on cycle burnup:

$$F_Q^L(x, y, z)^{RPS} = F_Q^D(x, y, z) * M_C(x, y, z) / (UMT * MT * \cancel{TILT})$$

*Section 6.3, equation on page 6-6 and removal of TILT definition:*

$$F_{\Delta H}^M(x, y) * UMR * \cancel{TILT} \leq [F_{\Delta H}^D(x, y) * M_{\Delta H}(x, y)]$$

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

*Section 6.3, text and equation on page 6-7:*

In application to power distribution surveillance,  $F_{\Delta H}^D$  and  $M_{\Delta H}(x, y)$  are combined, along with UMR ~~and TILT~~ to yield a maximum allowed  $F_{\Delta H}^L(x, y)^{SURV}$  term, which will be interpolated for each x,y location based on cycle burnup and power level:

$$F_{\Delta H}^L(x, y)^{SURV} = F_{\Delta H}^D(x, y) * M_{\Delta H}(x, y) / (UMR * \cancel{TILT})$$

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Technical Justification: The Technical Specification limiting condition for operation for quadrant power tilt requires that the quadrant power tilt ratio (QPTR) be less than or equal to a value of 1.02. Actions are required by the reactor operator if the 1.02 limit is exceeded. The purpose of the specification is to provide assurance that the reactor core's gross radial power distribution remains consistent with the power distribution assumed in the safety analysis. Signals from excore nuclear detectors provide continuous monitoring of the state of the neutron flux distribution in the reactor core. They are used to detect gross changes in the radial power distribution as indicated by excore detector quadrant power tilt relative to the previous detector calibration. The QPTR may be recalibrated (set to 1.0) following performance of a flux map. This specification, along with the axial flux difference, and control rod insertion limits specifications, provides the reactor operator with the indications needed to ensure the reactor core is operating within the bounds of the safety analysis.

Design analyses include a quadrant power tilt penalty which accounts for a change in the QPTR of 1.02 in the verification of the acceptability of loss of coolant accident (LOCA), departure from nucleate boiling (DNB) and centerline fuel melt (CFM) thermal limits. As a result, the following limits include the effects of the quadrant power tilt penalty in their determination.

- axial flux difference limits
- rod insertion limits
- $f(\Delta I)$  limits

Sections 4.2 (LOCA Margin Calculations), 4.3 (LOFA DNB Margin Calculations) and 4.5 (Centerline Fuel Melt Margin Calculations) describe the methodology used to evaluate each of these limits.

The LOCA  $F_Q$ , CFM  $F_Q$  and DNB  $F_{\Delta H}$  surveillance methodology described in Sections 6.1, 6.2 and 6.3 apply a quadrant power tilt penalty in the confirmation of  $F_Q$  LOCA and CFM surveillance limits, and  $F_{\Delta H}$  DNB limits. The proposed change requests removal of this penalty from the  $F_Q$  and  $F_{\Delta H}$  power distribution surveillances. A QPTR peaking penalty will continue to be applied in the thermal limits verification and AFD, RIL and  $f(\Delta I)$  limit determinations.

Technical Specifications require the periodic measurement of the reactor core power distribution. These measurements and the accompanying heat flux hot channel factor ( $F_Q$ ) and nuclear enthalpy rise hot channel factor ( $F_{\Delta H}$ ) surveillance requirements ensure that the measured  $F_Q$ 's and  $F_{\Delta H}$ 's are within their respective limits assumed in the safety analysis. Implicit in the power distribution measurement is the effect of any real core tilt. As a result, application of a quadrant power tilt penalty at the time of the measurement is not appropriate. However, the current Duke methodology increases the measured  $F_Q$  and  $F_{\Delta H}$  by the quadrant power tilt penalty as shown in the  $F_Q$  and  $F_{\Delta H}$  surveillance equations repeated below from sections 6.1, 6.2 and 6.3 of the report.

$$F_Q^L(x, y, z)^{OP} = F_Q^D(x, y, z) * M_Q(x, y, z) / (UMT * MT * TILT)$$

$$F_Q^L(x, y, z)^{RPS} = F_Q^D(x, y, z) * M_C(x, y, z) / (UMT * MT * TILT)$$

$$F_{\Delta H}^L(x, y)^{SURV} = F_{\Delta H}^D(x, y) * M_{\Delta H}(x, y) / (UMR * TILT)$$

The quadrant power tilt penalty was included in the above equations to account for the potential of a core perturbation that occurs after the current measurement but prior to the next power distribution surveillance. The tilt penalty accounts for an excore quadrant power tilt up to 2.0% as allowed by Technical Specifications. Inclusion of this penalty is not required for the current measurement since the

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measured  $F_{\Delta H}$  and  $F_Q$  includes the effects of any tilt. Inclusion of this penalty in the time period between power distribution surveillances is unnecessary based on the following rationale.

- a. A QPTR peaking penalty will continue to be applied in the thermal limits verification and in the determination of AFD, RIL and  $f(\Delta I)$  limits.
- b. Continuous QPTR indications are available to reactor operators to readily identify the presence of a core tilt and to initiate an investigation of cause. Additionally, rod insertion limit alarms and RCS loop flows and temperatures are available to identify abnormal conditions that may initiate a quadrant power tilt and allow remediation of the causal factor.
- c.  $F_Q$  and  $F_{\Delta H}$  measured power distribution data is trended per the requirements of the DPC-NE-2011-P surveillance methodology to capture as-built operating characteristics of the core. This trending is used to project margin out to the next surveillance providing indication of actual core behavior.
- d. Conservatism in the power distribution analysis methodology provides additional assurance that  $F_{\Delta H}$  and  $F_Q$  limits will not be exceeded prior to the next surveillance interval. These include the development of AFD, rod insertion and  $f(\Delta I)$  limits using an appropriate tilt penalty corresponding to the Technical Specification QPTR limit.

Power distribution surveillances, via an incore flux map, are currently performed every 31 effective full power days (EFPD). Each flux map is reviewed to determine if the core is operating consistent with the design analyses used to confirm the acceptability of the safety analysis. Small incore power tilts are not unexpected and are a consequence of as-built fuel deviations, fuel shuffling and differences in reactor coolant system (RCS) loop flows and temperature. The nuclear engineering hot channel factor applied in design analyses accounts for differences in design and as-built fuel, while historical impacts of shuffling and RCS loops differences are implicit in the development of  $F_Q$  and  $F_{\Delta H}$  peaking factor uncertainties. A unique aspect of the DPC-NE-2011-P surveillance methodology relative to the current Harris and Robinson Technical Specification power distribution surveillance methodology and NUREG-1431 is trending of measured  $F_Q$  and  $F_{\Delta H}$  peaking factors. Power distribution information from the previous measurement and the current measurement are extrapolated. Trending of the measurement is performed to determine the point in time (burnup) where the measured  $F_Q$  and  $F_{\Delta H}$  would exceed allowable limits prior to the next surveillance. This trending accounts for normal depletion effects and abnormal peaking trends that could be produced from deviations in as-built versus design fuel values, and asymmetric inlet flow or temperature distributions that potentially could produce quadrant power tilts. If extrapolation of the measured peaking factors indicates that allowable limits would be exceeded prior to 31 EFPD beyond the most recent measurement, then either a power distribution measurement would be performed prior to the surveillance limits being exceeded or the measurement is increased by an appropriate penalty and compared against LOCA, DNB or CFM limits. These requirements provide assurance that both  $F_Q$  and  $F_{\Delta H}$  will not exceed their respective limits for any significant period of time without detection. Separate  $F_Q$  and  $F_{\Delta H}$  penalty factors are calculated (described in sections 6.1.1 and 6.3.1) to account for the change in peaking over the 31 EFPD surveillance interval. A minimum factor of 1.02 is always applied even if the penalty factor is less than 1.02. This conservatism in the surveillance methodology is for the times in life where the peaking factors are constant or decreasing with burnup.

As described previously, the purpose of the QPTR specification and the 1.02 limit is to identify and limit the magnitude of changes to the radial power distribution between surveillance periods. Because the QPTR is based on signals from the excore detectors, only gross changes in the radial power distribution are indicated. Severe quadrant power tilts are not expected during normal operation. If they were to

## **DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)**

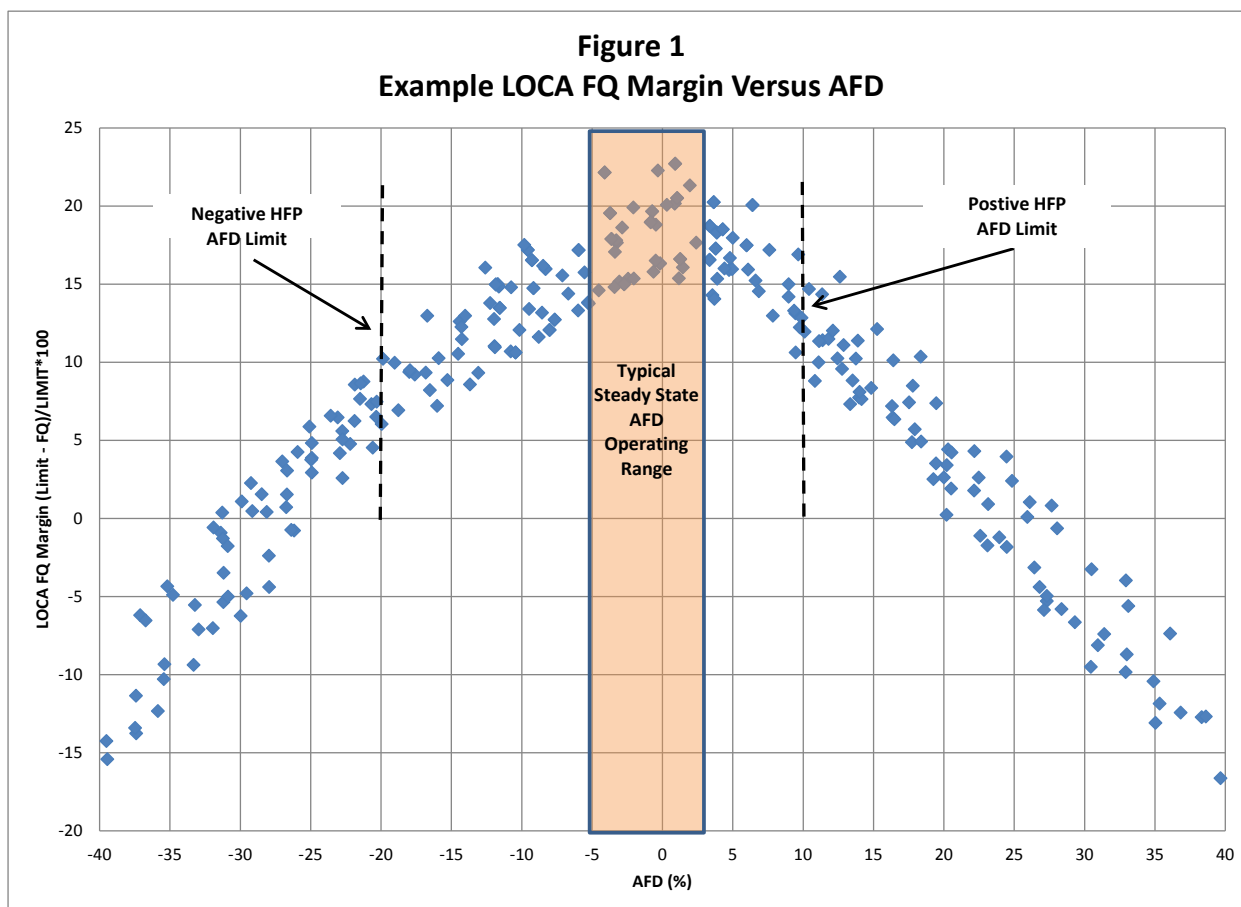
occur, the most likely cause would be from dropped or misaligned control rods. These types of events would be promptly identified to the operator by rod insertion limit alarms and/or by a step change in the indicated QPTR. A step change in QPTR would result in an investigation and correction of the initiating event. Additionally, these types of events are evaluated as part of the Chapter 15 accident analyses.

For quadrant power tilts up 2.0% (QPTR or 1.02), there is reasonable assurance that  $F_Q$  and  $F_{\Delta H}$  power peaking limits would not be exceeded prior to the next surveillance interval. The bases for this conclusion is conservatism in the power distribution evaluation methodology associated with the generation xenon distributions more severe than expected to occur, the application of peaking factor, rod position and AFD uncertainties all in the adverse direction during the thermal limit evaluation process, and the availability of continuous core monitoring via the excore nuclear detectors. Operationally, the reactor core is not routinely operated at the edge of the AFD envelope which is where limiting  $F_{\Delta H}$  and  $F_Q$  margins occur. This provides additional peaking margin available to offset adverse effects from unexpected core tilts. Figure 1 conceptually illustrates the peaking margin between steady state and transient xenon conditions. This figure plots  $F_Q$  margin versus AFD for an example core design and includes the AFD range associated with normal steady state operation. The AFD space beyond the normal operating AFD condition can only be achieved if the core is in a transient state as the result of a power maneuver. In this example, there is approximately 13% steady state LOCA  $F_Q$  margin (minimum margin associated with the steady state operating band) and 4% transient LOCA  $F_Q$  margins (minimum margin at the positive and negative HFP AFD limit – denoted by the vertical dashed lines). Since the core normally operates at steady state conditions, there is available peaking margin between steady state and transient conditions to accommodate quadrant power tilts. This margin varies depending upon the size of the operational AFD limits and the thermal limits being evaluated (i.e. LOCA, DNB or CFM).

In summary, extrapolation of the power distribution measurements provides indication of quadrant power tilts that could be produced from as-built versus design deviations, fuel shuffles and RCS loop flow and temperature differences. Conservatism in the power distribution methodology, peaking margin between steady state and transient xenon conditions, and continuous rod alignment and QPTR indications to the reactor operator provide assurance that  $F_Q$  and  $F_{\Delta H}$  power peaking limits are not exceeded prior to the next surveillance interval. For these reasons, removal of the QPTR penalty from the heat flux hot channel factor  $F_Q$  and nuclear enthalpy rise hot channel factor  $F_{\Delta H}$  surveillance requirements is acceptable. This change would also bring the DPC-NE-2011-P surveillance methodology in line with the current Harris and Robinson  $F_{\Delta H}$  and  $F_Q$  surveillance methodology, and standard Technical Specifications for Westinghouse plants (NUREG-1431, “Standard Technical Specifications for Westinghouse Plants”) where a quadrant power tilt penalty is not applied in heat flux hot channel factor  $F_Q$  and nuclear enthalpy rise hot channel factor  $F_{\Delta H}$  surveillances. Application of a QPTR penalty in design analyses will continue to be performed to verify the acceptability of LOCA, DNB and CFM thermal limits which forms the basis for AFD, rod insertion and  $f(\Delta I)$  limits used for power distribution control.



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**Change 6-4: LOCA  $F_Q$  Surveillance Methodology (Section 6.1,  $F_Q^D(x, y, z)$  definition, page 6-2)**

Description of Change: Clarification is added to  $F_Q^D(x, y, z)$  the definition to denote where the UCT term is defined.

$$F_Q^D(x, y, z) = NP^D(x, y, z) * UCT, \text{ design power distribution for } F_Q. \text{ (UCT is defined in section 3.1)}$$

Technical Justification: Clarification.

**Change 6-5:  $F_Q(x, y, z)$  Margin Decrease Penalty Factor Methodology (Section 6.1.1, last paragraph, page 6-4),  $F_{\Delta H}(x, y)$  Margin Decrease Peaking Penalty Factor Methodology Section 6.3.1, last paragraph, page 6-8), AFD – Power Level Limits (Section 6.4.1, first paragraph, page 6-9), Control Rod Insertion Limits(Section 6.4.2, first paragraph, page 6-9)**

Description of Change: A clarification is added stating the  $F_Q(x, y, z)$  and the  $F_{\Delta H}(x, y)$  margin decrease penalty factors, axial flux difference limits and control rod insertion limits are specified in the core operating limits report (COLR).

*Section 6.1.1 Change:*

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For burnup ranges where the  $F_Q$  margin decrease factor is less than 1.02, a value of 1.02 will be maintained. **Cycle-specific  $F_Q$  margin decrease penalty factors are specified in the COLR.**

### Section 6.3.1 Change:

For burnup ranges where the  $F_{\Delta H}$  margin decrease factor is less than 1.02, a value of 1.02 will be maintained. **Cycle-specific  $F_{\Delta H}$  margin decrease penalty factors are specified in the COLR.**

### Section 6.4.1 Change:

This allowance is meant to be used to increase the plant availability during transient situations and is not meant to be used for normal operation. **Operating AFD – power level limits are specified in the COLR.**

### Section 6.4.2 Change:

These limits are a Limiting Condition of Operation, so operation outside of these limits is allowed for short periods of time. **Control rod insertion limits are specified in the COLR.**

Technical Justification:  $F_Q(x, y, z)$  and the  $F_{\Delta H}(x, y)$  margin decrease penalty factors are calculated for each fuel cycle and the values for each parameters are specified in the core operating limits report. Operating AFD – power level limits and control rod limits assumed in the verification of thermal limits are also specified in the core operating limits report.

## Change 6-6: LOFA DNB $F_{\Delta H}$ Surveillance Methodology (Section 6.3, $F_{\Delta H}^D(x, y)$ definition, page 6-6)

Description of Change: Clarification is added to the  $F_{\Delta H}^D(x, y)$  definition to define the RNP term, and to denote where the UCR term is defined.

$F_{\Delta H}^D(x, y) = \text{RNP}^D(x, y) * \text{UCR}$ , **design radial power distribution for  $F_{\Delta H}$ . (RNP is defined in section 4.3 and UCR is defined in section 3.1)**

Technical Justification: Clarification.

## Change 6-7: AFD - Power Level Limits (Section 6.4.1, second paragraph, page 6-9), Heat Flux Hot Channel Factor - $F_Q(x, y, z)$ (Section 6.4.3, base load discussion, pp, 6-10 – 6-11), Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(x, y)$ (Section 6.4.4, page 6-11), Quadrant Power Tilt (Section 6.4.5, page 6-12)

Description of Change: Equations and discussions pertaining to base load operation are deleted.

Technical Justification: Refer to the discussion for change 5-1.

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### Change 6-8: Heat Flux Hot Channel Factor - $F_Q(x,y,z)$ (Section 6.4.3, second paragraph, page 6-9)

Description of Change: The implication that the reactor is tripped if the measured  $F_Q$  exceeds the limiting condition of operation is removed.

This limit on  $F_Q^M(x,y,z)$  is a Limiting Condition of Operation, so operation outside of the limit is allowed for a short period of time to allow the operator to bring the reactor back within the limits ~~without a reactor trip~~ **or be placed in MODE 2.**

Technical Justification: If  $F_Q^M(x,y,z)$  exceeds its limiting condition of operation limit and the compensating actions cannot be performed within the Technical Specification completion time, reactor power is reduced to at least MODE 2. The reactor is shutdown in an orderly manner to MODE 2. A reactor trip is not typically performed.

### Change 6-9: Heat Flux Hot Channel Factor - $F_Q(x,y,z)$ (Section 6.4.3, second paragraph, page 6-10)

Description of Change: The specific value of the top and bottom regions of the core excluded from surveillance is changed to 10 - 15%.

$F_Q^L(x,y,z)^{OP}$  and  $F_Q^L(x,y,z)^{RPS}$  are generated in the maneuvering analysis. These limits are specified in the COLR and are not imposed on the top or bottom **10 - 15%** of the core. The limits on  $F_Q^M(x,y,z)$  account for an appropriate measurement uncertainty, which is provided in the COLR.

Technical Justification: The top and bottom 15% of the core is excluded from surveillance at Harris, McGuire and Catawba. The top and bottom 10% of the core is excluded from surveillance at Robinson. The exclusion region is dictated by both the fuel design and fuel management strategy. The top and bottom regions are excluded from evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making precise measurements in these regions.

### Change 6-10: Heat Flux Hot Channel Factor - $F_Q(x,y,z)$ (Section 6.4.3, third paragraph, page 6-10)

Description of Change: Compensating actions for the condition where  $F_Q^M(x,y,z)$  exceeds  $F_Q^L(x,y,z)^{RPS}$  are added.

If  $F_Q^M(x,y,z)$  exceeds  $F_Q^L(x,y,z)^{OP}$  (LOCA limits), the AFD - power level limits must be adjusted by reducing the allowed AFD span (move the negative and positive AFD limits closer to the zero AFD point), so that positive margin would be maintained at the extremes of the AFD - power level operating limits. If  $F_Q^M(x,y,z)$  exceeds  $F_Q^L(x,y,z)^{RPS}$  (CFM limits), then a reduction is made to the OTAT trip setpoints, or the  **$f_1(\Delta I)$  or  $f_2(\Delta I)$**  breakpoints are adjusted.

Technical Justification: The option to adjust the breakpoints of the OTAT  $f_1(\Delta I)$  trip reset penalty function is added because the OPAT  $f_2(\Delta I)$  trip reset penalty function at Harris is not active. As a result, the OTAT  $f_1(\Delta I)$  trip reset function breakpoints must be adjusted to compensate for the condition where  $F_Q^M(x,y,z)$  exceeds  $F_Q^L(x,y,z)^{RPS}$  limit. The OPAT  $f_2(\Delta I)$  trip reset penalty function is active at McGuire, Catawba and Robinson.

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### Change 6-11: Quadrant Power Tilt (Section 6.4.5, page 6-12)

Description of Change: The bases for not requiring a penalty for quadrant power tilts up to 2.0% is clarified.

An allowance for a 2% quadrant power tilt was made in the AFD - power level operating limits **and in the verification of DNB, CFM and LOCA thermal limits.** and in the values of  $F_Q^L(x, y, z)^{OP}$ ,  $F_Q^L(x, y, z)^{RPS}$ ,  $F_Q^{Max-BL}(x, y, z)$ ,  $F_{\Delta H}^L(x, y)^{SURV}$ , and  $F_{\Delta H}^{Max-BL}(x, y)$ . Thus, no action is required for an indicated quadrant power tilt of up to 2%.

Technical Justification: Design analyses include a quadrant power tilt penalty which accounts for a change in the QPTR of 1.02 in the verification of the acceptability of loss of coolant accident, departure from nucleate boiling and centerline fuel melt thermal limits. Therefore, no action is required for an indicated quadrant power tilt of up to 2.0%. The important aspect is a quadrant power tilt allowance is considered in the determination of the plant axial flux difference limits, rod insertion limits and RPS setpoints. Refer to change 6-3 for additional details.

### Change 6-12: Figures 9 and 12

Description of Change: The titles of Figures 9 and 12 are changed to denote the K(Z) and rod insertion limits specified are sample limits.

**Figure 9: Sample K(Z) – Normalized  $F_Q(Z)$  as a Function of Core Height**

**Figure 12: Sample Control Rod Insertion Limits vs. Thermal Power**

Technical Justification: Actual control rod insertion limits and K(Z) limits used in the maneuvering analysis are specified in the core operating limits report (COLR).

## DPC-NE-2011-P Changes And Technical Justifications (Non-Proprietary Version)

### 8.0 Section 7.0 Changes

Description of Change: Updated References 4 and 14, added References 6, 7, 8, 9, 10 and 16, and deleted References 12 and 13.

4. "Duke Power Company, Thermal-Hydraulic Statistical Core Design Methodology", DPC-NE-2005P-A, Revision ~~35~~, ~~September 2002~~ (**Revision 5 Submitted to NRC for Review and Approval**)
6. ~~Intentionally Left Blank.~~ **Technical Specifications for Shearon Harris Nuclear Power Plant Unit 1, Docket No. 50-400.**
7. ~~Intentionally Left Blank.~~ **Technical Specifications for H. B. Robinson Steam Electric Plant Unit No. 2, Docket No. 50-261.**
8. ~~Intentionally Left Blank.~~ **"Thermal-Hydraulic Transient Analysis Methodology", DPC-NE-3000-PA, Rev. 5a**
9. ~~Intentionally Left Blank.~~ **"Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology", DPC-NE-3001-PA, Rev. 1**
10. ~~Intentionally Left Blank.~~ **"UFSAR Chapter 15 System Transient Analysis Methodology", DPC-NE-3002-A, Rev. 4b**
11. ~~Intentionally Left Blank.~~ **Robert E. Martin (NRC) to G. R. Paterson (Duke), "McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC NOS. MB8361 and MB8362)", January 14, 2004.**
12. ~~"Computer Code Certification for SMARGINS," DPC internal document. Sean Peters (NRC) to D. Jamil (Duke), "Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC NOS. MB8359 and MB8360)", December 19, 2003.~~
13. ~~"Computer Code Certification for MARGINPLOT," DPC internal document.~~ **Intentionally Left Blank.**
14. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Core Thermal-Hydraulic Methodology using VIPRE-01", DPC-NE-2004-PA, Revision ~~42a~~, **December 2008** ~~SER Dated February 20, 1997 (DPC Proprietary).~~
16. **Duke Energy Carolinas, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors", Rev. 0, August 2015. (Submitted to NRC for Review and Approval)**

Technical Justification: References 4 is currently under NRC review. Reference 14 was updated to the current NRC-approved version of the report.

References 6 and 7 were added and refer to the Harris and Robinson Technical Specifications.

References 8, 9 and 10 were added as the result of change 4-4.

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Reference 11 was added as the result of change B-1.

Reference 12 was deleted as the result of change 1-7 and subsequently replaced as the result of change B-1.

References 13 was deleted as the result of change 1-7.

Reference 16 was added to note that the Harris and Robinson nuclear analysis methods are based on CASMO-5/SIMULATE-3, while the current approved nuclear analysis methodology is based on CASMO-4/SIMULATE-3. Refer to change 1-7 for additional details.

***Change B-1: Appendix B Changes, third paragraph, page B-2:***

Description of Change: A note is added stating the OPAT  $K_5$  constant was relocated from Technical Specifications to the COLR in References 11 and 12.

The OPAT parameter  $K_5$  is not being relocated to the COLR since it currently is not calculated as part of the reload design methodology described below. **(The  $K_5$  constant was subsequently relocated from Technical Specifications to the COLR for McGuire and Catawba in References 11 and 12.)**

Technical Justification: This clarification was made to convey that the OPAT constant  $K_5$  no longer resides in Technical Specifications. This constant was relocated from Technical Specifications to the COLR as described in References 11 and 12.

**Attachment 9**

**Application of the DPC-NE-2004-PA Operating Limits and Maximum Allowable Total Peak  
Methodology to Harris and Robinson Nuclear Plants**

## **Application of the DPC-NE-2004-PA Operating Limits and Maximum Allowable Total Peak Methodology to Harris and Robinson Nuclear Plants**

### **1.0 Background**

DPC-NE-2004-PA, “Core Thermal-Hydraulic Methodology Using VIPRE-01”, Reference 1, describes Duke Energy’s NRC-approved steady state thermal hydraulics analysis methodology for McGuire and Catawba Nuclear Stations using the VIPRE-01 computer code. Key elements of the report include a description of the code inputs and correlations used to develop McGuire/Catawba VIPRE-01 models. The methodology used to determine allowable thermal-hydraulic operating limits in terms of core power level, reactor coolant system temperature and pressure, and three dimensional core power distributions is also presented. This methodology is used to determine allowable operating limits that provide DNB protection for nuclear plants that utilize a Westinghouse NSSS control and protection system. This method is described in Section 5 of Reference 1. Section 5 is applicable to the Harris and Robinson nuclear units as well as McGuire and Catawba.

### **2.0 Applicability**

The thermal-hydraulic methodology in Reference 1 is intended to interface with a Westinghouse plant reactor protection system (RPS) to ensure the required DNB design basis is satisfied. McGuire, Catawba, Harris and Robinson nuclear units all employ a Westinghouse RPS system to prevent fuel damage from occurring during Condition I and II events. DNB protection is accomplished through the use of the over-temperature delta-temperature (OT $\Delta$ T) and over-power delta-temperature (OP $\Delta$ T) trip functions, in combination with the following trips which limit the applicable range over which the OT $\Delta$ T and OP $\Delta$ T trip functions must provide DNB protection.

- High pressurizer pressure trip
- Low pressurizer pressure trip
- Low reactor coolant flow trip

Core DNB limits are determined for a range of operating conditions to ensure that the DNB design basis is satisfied. The protected space in terms of power level and pressure is defined by the OT $\Delta$ T/OP $\Delta$ T trip, high pressurizer pressure trip, and low pressurizer pressure trip. An upper range on temperature is set by a combination of the Hot Leg boiling limit and the steam generator safety valve actuation. The low reactor coolant system (RCS) flow trip provides DNB protection by limiting the reactor coolant flow that must be considered.

The methodology described in Reference 1 requires the following three elements to be in place.

- A NRC-approved VIPRE-01 model for the unit being evaluated
- A NRC-approved critical heat flux correlation for the fuel type
- A NRC-approved statistical DNBR limit for the fuel type

VIPRE-01 models have been developed to represent the Harris and Robinson core geometry and AREVA’s HTP fuel product in current operation. The VIPRE-01 model developed, along with the code inputs and correlations used to the represent AREVA’s Advanced W 17x17 HTP and 15x15 HTP fuel products are described in Appendices H and I of DPC-NE-2005-P-A, “Duke Energy Thermal-Hydraulic Statistical Core Design Methodology”, Reference 2.



The critical heat flux (CHF) correlation applicable to AREVA's Advanced W 15x15 and 17x17 HTP fuel designs is the HTP CHF correlation developed in Reference 3. The applicability of this correlation with the VIPRE-01 computer code was also demonstrated in Appendices H and I of Reference 2. In addition, Reference 2 describes the statistical core design (SCD) DNBR limit for each reactor considering plant specific uncertainties.

All three methodology components (VIPRE-01 model, CHF correlation and statistical DNBR limit) in Reference 2 are used to calculate core DNB limits. These limits are determined as a function of power level, reactor coolant system temperature, and reactor coolant system pressure using a reference radial peak power,  $F_{AH}$ , and the pin-by-pin power distribution specific to the applicable fuel design evaluated in Reference 2 Appendices H and I. Each point along the line corresponds to a constant DNBR value based on the SCD DNBR limit. DNB limit lines are provided in the core safety limits figure presented in Technical Specifications Figure 2.1-1 or provided in the cycle specific Core Operating Limits Report (COLR).

In summary, the Duke Energy thermal-hydraulic methodology used to provide DNB protection for Westinghouse plant instrumentation systems is not dependent upon the plant where it is applied. As a result, the method is as applicable to the Harris and Robinson nuclear units as well as McGuire and Catawba. It does however, rely on the use of NRC-approved VIPRE-01 model, CHF correlation, and DNBR limit as specified in Reference 2 for each plant.

### 3.0 Maximum Allowable Peaking Limits

The Duke Energy maximum allowable total peaking (MATP) methodology is described in Section 5 of Reference 1. This method is independent of the reactor design or fuel type and consists of DNB calculations performed with an NRC-approved model, fuel design, critical heat flux correlation, and DNBR limit. The methodology is used to develop local fuel peaking limits to prevent DNB for a fixed reactor core operating condition defined by a thermal power level, pressure, RCS flow and reactor coolant temperature. The MATP limits developed are lines of constant MDNBR for a range of axial peak magnitudes with the location of the peak varied from the bottom to the top of the core. For a given axial peak magnitude ( $F_z$ ) and axial location, the total peak ( $F_{AH} * F_z$ ) that yields a MDNBR equal to the DNBR limit defines a single point along a MATP curve. An example MATP curve is shown in Figure 14 of Reference 1.

The MATP limits are used to ensure the DNB design basis for Conditions I and II transients is satisfied by comparing calculated total peaking factors from a core nodal simulator (e.g. SIMULATE-3) against the MATP limits for a series of power distributions. Core power distributions are generated (by the core nodal simulator) as a function of rod position, burnup, xenon concentration, inlet temperature, and core power level. The method to generate these power distributions is described in Reference 4. DNBR margin is computed for each fuel assembly to verify positive margin exists for the allowable core power versus axial flux difference (AFD) limit used to establish the  $F(\Delta I)$  portion of the OTAT trip function. The same methodology is used at a different state point condition to develop the core operational AFD limits. The operational AFD limits protect the core against the limiting Condition II DNB transient, typically the Loss of Flow transient.

In summary, the Duke Energy MATP methodology is a generic methodology that relies on the NRC-approved VIPRE-01 model, fuel design, CHF correlation, and DNBR limit. It is applicable to any reactor or fuel type provided the NRC approvals for the unit specific methodology components are obtained. Since NRC approval has been granted for the VIPRE-01 model, CHF correlation, and DNBR limits in

References 2 and 3, application of the Duke Energy Core DNB limit and MATP methodology to the Harris and Robinson plants is appropriate.

#### **4.0 Conclusion**

The methodology described in DPC-NE-2004-P for development of limits to verify the required DNB design basis is satisfied is independent of the Westinghouse unit analyzed. The methodology does require an NRC approved:

- VIPRE-01 model for the unit being evaluated
- Critical Heat Flux (CHF) correlation for the fuel type
- Statistical DNBR limit for the fuel type

Satisfying these three criteria allow use of the Duke Energy thermal-hydraulic analysis methodology, including the Core Safety Limit generation and MATP limit approach described in Reference 1 for plants using the Westinghouse reactor control and protection system.

#### **5.0 References**

1. DPC-NE-2004-PA, Rev. 2a, "Core Thermal-Hydraulic Methodology Using VIPRE-01"
2. DPC-NE-2005-P-A, Rev. 5, "Duke Energy Thermal-Hydraulic Statistical Core Design Methodology"
3. EMF-92-153(P)(A), Rev. 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"
4. DPC-NE-2011-PA, Rev. 1a, "Nuclear Design Methodology Report For Core Operating Limits of Westinghouse Reactors"