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Serial: RA-15-0031
August 19, 2015

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: 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"

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, Inc., referred to henceforth as "Duke Energy", is submitting a request for an amendment 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, Duke Energy requests 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. This methodology will initially be used to perform reactor physics calculations as part of reload design analysis for SHNPP and HBRSEP, which is currently performed by AREVA using CASMO-3/PRISM. Approval of the new methodology will allow Duke Energy to perform the subject analysis, as opposed to utilizing contract services. The DPC-NE-1008-P method is an evolution of the CASMO-4/SIMULATE-3 methodology currently used for McGuire, Catawba, and Oconee nuclear stations. The evolution consists of the replacement of the CASMO-4 code with CASMO-5 to generate nuclear data for SIMULATE-3. Duke Energy and NRC staff participated in a pre-submittal meeting on April 14, 2015, regarding these changes.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed changes involve no significant hazards consideration. The bases for these determinations 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 provides the retyped TS pages. Note that because the proposed change to the TSs would be affected by an amendment request currently awaiting NRC approval (submitted March 5, 2014; ML15075A211), the TS mark-up and retyped pages also reflect the changes of that previously submitted request.

Attachment 5 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 5 be withheld from public disclosure. An affidavit is included (Attachment 1) attesting to the proprietary nature of the information. A non-proprietary version of the attachment is included in Attachment 6.

Approval of the proposed amendment is requested by December 31, 2016 in order to support the core design of SHNPP Cycle 22, which is expected to commence operation Spring 2018. The requested approval date allows sufficient time to establish the appropriate contract services to perform the analysis, if the amendment is not approved. An implementation period of 120 days is requested to allow for updating the TS and Facility Operating License.

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.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 19, 2015.

Sincerely,

Regis T. Repko

Regis T. Repko
Senior Vice President – Governance, Projects and Engineering

JBD

Attachments:
1. Affidavit of Regis T. Repko
2. Evaluation of the Proposed Change
3. Proposed Technical Specification Changes (Mark-Up)
4. Retyped Technical Specification Pages
5. DPC-NE-1008-P, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors” (Proprietary)
6. DPC-NE-1008, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors” (Redacted)

cc: (all with Attachments unless otherwise noted)

V. M. McCree, Regional Administrator USNRC Region II
J. D. Austin, USNRC Senior Resident Inspector – SHNPP
K. M. Ellis, USNRC Senior Resident Inspector – HBRSEP
M. C. Barillas, NRR Project Manager – SHNPP & HBRSEP
D. J. Galvin, NRR
W. L. Cox, III, Section Chief, NC DHSR (Without Attachment 5)
S. E. Jenkins, Manager, Radioactive and Infectious Waste Management Section (SC) (Without Attachment 5)
Attorney General (SC) (Without Attachment 5)
A. Gantt, Chief, Bureau of Radiological Health (SC) (without Attachment 5)

bcc: (all with Attachments unless otherwise noted)

Chris Nolan
Art Zaremba
Leo Martin
Bob Harvey
Lara Nichols
David Cummings
File: (Corporate)
Electronic Licensing Library (ELL)

Ben Waldrep
John Caves
Duncan Brewer
John Dufner
Paula Hamilton
Brian McCabe
Vijay D'Souza
Donald Griffith
Sean O'Connor
Frankie Womack
SHNPP NSRB
Lisa Owens
Cindy Hereford (For SHNPP Licensing/Nuclear Records Files)

Mike Glover
Richard Hightower
Heidi Walters (For HBRSEP Licensing/Nuclear Records Files)

Attachment 1

Affidavit of Regis T. Repko

AFFIDAVIT of Regis T. Repko

1. I am Senior Vice President of Governance, Projects, and 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.

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.

Rationale for holding this information in confidence 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 in the submittal is that which is marked in the proprietary version of the Duke methodology report DPC-NE-1008-P, *Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors*. This information enables Duke Energy to:

(a) Support license amendment requests for its Harris and Robinson reactors.

(b) Perform 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.

Regis T. Repko 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 August 19, 2015.

Regis T. Repko

Attachment 2

EVALUATION OF THE PROPOSED CHANGE

Subject: 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”

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 as part of the reload design process. 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,” and subsequent inclusion of DPC-NE-1008-P-A into the SHNPP and HBRSEP Technical Specifications (the “-A” is added to indicate an approved report, in accordance with the NRC process for topical reports).

2.0 DETAILED DESCRIPTION

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-A, “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 Attachments 3 and 4. Note that because the proposed change to the TSs would be affected by an amendment request currently awaiting NRC approval (submitted March 5, 2014; ML15075A211), the TS mark-up and retyped pages also reflect the changes of that previously submitted request . Because the current HBRSEP and SHNPP TSs are consistent with TSTF-363, “Revise Topical Report References in ITS 5.6.5, COLR [Core Operating Limits Report]” (References 4 and 5),

inclusion of revision dates for the topical report in the TS is not required, which is also consistent with NUREG-1431, Revision 4.

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. 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.

3.0 TECHNICAL EVALUATION

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 $F\Delta H$, Fq and Fz. 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 $F\Delta H$, Fq and Fz 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 with or without gadolinia.

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

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.

4.3 No Significant Hazards Consideration Determination

Duke Energy Progress, Inc., referred to henceforth as “Duke Energy”, requests NRC review and approval of a reactor core design methodology report DPC-NE-1008-P, Revision 0, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” and adoption of the methodology 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.

   Therefore, the proposed change does 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.

Therefore, the proposed change does 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.

   Therefore, the proposed change does 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. Carolina Power & Light Company letter, *Request for Technical Specification Change Revision to Core Operating Limits Report (COLR) References*, dated June 14, 2000 (ADAMS Accession No. ML003725331)
5. Carolina Power & Light Company letter, *Revised Technical Specification Pages for License Amendment Request – Addition of Methodology References to Core Operating Limits Report*, dated January 11, 2000 (ADAMS Accession No. ML003676878)

Attachment 3

Proposed Technical Specification Changes (Mark-up)

5.6 Reporting Requirements

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 ($F^N_{\Delta H}$) limit for Specification 3.2.2;

(continued)

5.6 Reporting Requirements (continued)

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)

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)

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.

Insert 1
(see next page)

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,

(continued)

Note: Item 28 is to be added pending NRC approval of LAR ML15075A211 submitted March 5, 2015

Insert 1:

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

29. DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” approved version as specified in the COLR.

No changes to this page. Included for information only.

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_Q^{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.

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).

No changes to this page. Included for information only.

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.

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.

Insert 2
(see next page)

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).

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.l. 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,

Note: Item p is to be added pending NRC approval of LAR ML15075A211 submitted March 5, 2015

Insert 2:

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).

q. DPC-NE-1008-P-A, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse 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).

Attachment 4

Retyped Technical Specification Pages

5.6 Reporting Requirements (continued)

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.

29. DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” approved version as specified in the COLR.

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d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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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,

(continued)

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).

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

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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.

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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).

q. DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse 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).

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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.l. 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,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)

Attachment 6

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

Proprietary Notice

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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 (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/FΔH)
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_2) 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 FΔH, Fq and Fz 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 UO2 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^{-5} 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 $F\Delta H$, Fq, and Fz 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^{10} 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^{10} depletion effects to ensure both the measured and predicted values were based on the same B^{10} abundance.

Table 3-2 compares measured to predicted BOC hot zero power (HZP) ARO critical boron concentrations for the benchmark cycles. The mean measured minus predicted deviation is [ ] ^{a,c} with a standard deviation of [ ] ^{a,c}. Figure 3-4 graphically presents deviations between measured and predicted HZP 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 [ ] ^{a,c} with a standard deviation of [ ] ^{a,c}. 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 [ ] ^{a,c} with a standard deviation of [ ] ^{a,c}.

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 [ ][a,c] with a standard deviation of [ ][a,c], 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 [ ][a,c] with a standard deviation of [ ][a,c].

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 FΔH, is simply the average relative power in each fuel assembly. The assembly maximum power, or assembly Fq, is the largest relative power in each fuel assembly. The assembly axial peak power, or assembly Fz, is the assembly Fq normalized to the assembly average power for each fuel assembly (Fz = Fq/FΔH). Measured assembly FΔH, Fq, and Fz 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 FΔH, maximum assembly Fq and assembly axial peak Fz 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.

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

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 FΔH, Fq and Fz 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 (FΔH) powers less than 1.0 are not considered in the ONRF calculation. Equation 3-2 is used to define each uncertainty factor.

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

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:

$$Bias = \frac{1}{n}\sum_{i}^{n}\frac{(P_i - M_i)}{M_i} \qquad \text{eq. 3-3}$$

where,

P_i = the i[th] predicted or calculated value
M_i = the i[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 σ_a and K_a are:

σ_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.

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

The E_{mth} term 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 by equation 3-5.

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

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 FΔH, Fq and Fz are summarized in Table 3-6. These assembly uncertainties are combined with pin uncertainties (developed in section 4) to calculate the FΔH, Fq and Fz 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 FΔH, Fq, and Fz 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

Cycle	Cycle Length (EFPD)	Power Level (MWt)	# Feed Assemblies	Feed Enrichments (w/o)	# Gad/IFBA Rods in Fresh	Gadolinia Concentrations (w/o Gd)	# Removable BP Rods
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

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

Harris Unit 1 Cycle 15

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

Harris Unit 1 Cycle 16

	Meas.	Pred.	Diff
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

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

Robinson Unit 2 Cycle 25

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

Robinson Unit 2 Cycle 26

	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 27

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

Robinson Unit 2 Cycle 28

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

a-c

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

McGuire Unit 1 Cycle 20

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

McGuire Unit 1 Cycle 21

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

a-c

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

McGuire Unit 1 Cycle 22

	Meas.	Pred.	Diff
EFPD	(ppm)	(ppm)	(M – P)

McGuire Unit 1 Cycle 23

	Meas.	Pred.	Diff
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	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
D' (P = 0.005)	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
D'	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
D' (P = 0.995)	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
Evaluation	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}

Assembly Uncertainty

Parameter	**Bias**	**Statistical Deviation ($K_a\sigma_a$)**	**Assembly Uncertainty Factor (ONRF)**
FΔH	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
Fq	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
Fz	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}

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)	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
HFP Critical Soluble Boron (ppmb)	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
BOC HZP Isothermal Temp. Coefficient (pcm/°F)	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
BOC HZP Individual Control Rod Worth (%)	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}

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

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1							28			15						1
2			19	SA		3 CB		51 CC		CB		SA				2
3					SD		SB	10	SB	30	SC	39		52		3
4		5 SA	36	CD				43 SE				CD		SA		4
5			SC		11				38		31		24 SD			5
6	17	CB	54			14 CC		6 CA		CC				8 CB		6
7			SB	44			32			16			47 SB			7
8	23	CC	58	SE	29	CA	46	CD		48 CA		50 SE	49	34 CC		8
9		57	SB						22		9		SB		56	9
10		CB			4	CC	1	CA		CC		12		CB		10
11	33		SD		40			26			21		SC		13	11
12		SA		CD		41		SE	55			7 CD		SA		12
13			45		35 SC		SB	20	SB		SD			25		13
14			18	SA		CB	27	CC		42 CB		37 SA		Incore Location RCCA Bank Location		14
15					53			2								15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

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

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1								48								1
2						CA		CD		39 CA						2
3				42			31 SA	27	SA		SC	43				3
4				CC	26	CB		15		22 CB		CC				4
5			35 SC		19 SB		12				20 SB	28		46		5
6		CA		CB	13	CD		4 CC		8 CD		CB		CA		6
7			30 SA				1 SB		2 SB			17	SA	37		7
8	47	CD	25		10	CC				5 CC			29	36 CD		8
9			SA		11		SB		3 SB	7			SA		50	9
10		CA	32	CB		CD	6	CC		CD		23 CB		40 CA		10
11					18 SB			9		14	21 SB		SC			11
12			41	CC		CB	16			CB		34 CC	44			12
13					SC		SA	24	SA	33						13
14					45	CA		CD	38	CA						14
15							49									15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

Incore Location
RCCA Bank Location

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

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1								18								1
2						CB				31 CB						2
3				1			9 SA	45	SA			10				3
4					13	CC		37 CD		3 CC						4
5			22		50 SB		24 CA		CA		49 SB	39		7		5
6		CB		CC	40			8 SB		15		CC		CB		6
7			33 SA		CA		28		20		CA	29	SA	17		7
8	36		46	CD	4	SB		CD		5 SB		CD	44	25		8
9			SA		11 CA				19	32	CA		SA		34	9
10		CB	26	CC			23	SB				41 CC		12 CB		10
11					48 SB		CA	38	CA	21	47 SB					11
12			35			CC	16	CD		CC		6	27			12
13							SA	43	SA	14						13
14					30	CB			42	CB						14
15							2									15
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	

Incore Location
RCCA Bank Location

Incore instrument at core location L-05 is used for RVLIS

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

a,c

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

a,c

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

a,c

Figure 3-11
Percent Difference in Individual Bank Worths

a,c

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

a-c

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

a-c

Figure 3-13
Robinson Unit 2 Cycle 28
Assembly Average Power Distribution Comparisons
99.7% FP, 7 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, 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

a-c

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

a-c

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

a-c

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 (UO_2+Gd_2O_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 UO_2+Gd_2O_3, Ag-In-Cd and B_4C 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.

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

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.

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

where,

P_i = the ith predicted CASMO-5 or SIMULATE-3 pin power

M_i = the ith 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.

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

where,

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

σ_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.

$$Bias = \frac{1}{n}\sum_{i}^{n}\frac{(P_i - M_i)}{M_i} \qquad \text{eq. 4-4}$$

where,

n = sample size

P_i = the ith predicted pin power

M_i = the ith measured pin power

The CASMO-5 and SIMULATE-3 95/95 LEU pin power uncertainties are [ ]^{a,c}, 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 [ ] a,c with a standard deviation of [ ]a,c. 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 [ ] a,c 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.

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

where,

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

$\overline{M}$ = Average of measured powers [ ] a,c

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

σ = standard deviation of the difference

An average gadolinia pin power of [ ] a,c was chosen instead of 1.0 in the above calculation because the presence of Gd_2O_3 in fuel results in a lower thermal conductivity and heat capacity relative to standard UO_2 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_2 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_2 fuel results in centerline fuel melt (CFM) limits for gadolinia fuel being challenged at lower power densities than UO_2 fuel. Duke has determined that the fractional decrease in CFM linear heat rate limits is approximately [ ] a,c 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 [ ] a,c 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 **[ ]** a,c below the assembly average power ($\overline{M}$ = [ ] a,c). This results in a CASMO-5 pin power uncertainty for gadolinia fuel of **[ ]** a,c based on a K value of **[ ]** a,c from Reference 19 **[**

] a,c. 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 the 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^{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 [ ] a,c with a bias of [ ] a,c. 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:

$$SIMULATE\ Gadolinia\ Pin\ Power\ \ Uncertainty$$
$$= -(Bias_c + Bias_{pr}) + \sqrt{(K_c\sigma_c)^2 + (K_{pr}\sigma_{pr})^2}$$
$$= [\qquad\qquad\qquad]^{a,c}$$

where, bias = [ ] a,c for Gad fuel, and

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

The resulting SIMULATE-3 gadolinia pin power uncertainty is [ ] a,c. 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 [ ] a,c, 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_2O_3/1.94 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_2O_3/1.94 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_2O_3/1.94 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 (ºF)	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_2O_3/1.94 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_2O_3/1.94 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_2O_3/1.94 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	[ ] a,c	[ ] a,c
D' (P = 0.005)	[ ] a,c	[ ] a,c
D'	[ ] a,c	[ ] a,c
D' (P = 0.995)	[ ] a,c	[ ] a,c
Evaluation	[ ] a,c	[ ] a,c

LEU Fuel Pin Power Uncertainty Calculation Results

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

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**
[ ] a,c	[ ] a,c

Normality Test Using the W Test for Normality

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

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

N	$\overline{M}$	Bias	Standard Deviation (%)	K	Uncertainty+
[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c	[ ] a,c

+ Based on Equation 4-5 where,

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

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

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	[ ] a,c
D' (P = 0.005)	[ ] a,c
D'	[ ] a,c
D' (P = 0.995)	[ ] a,c
Evaluation	[ ] a,c

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

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

Non Parametric Uncertainty = - E_{mth}
(E_{mth} is the m^{th} 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

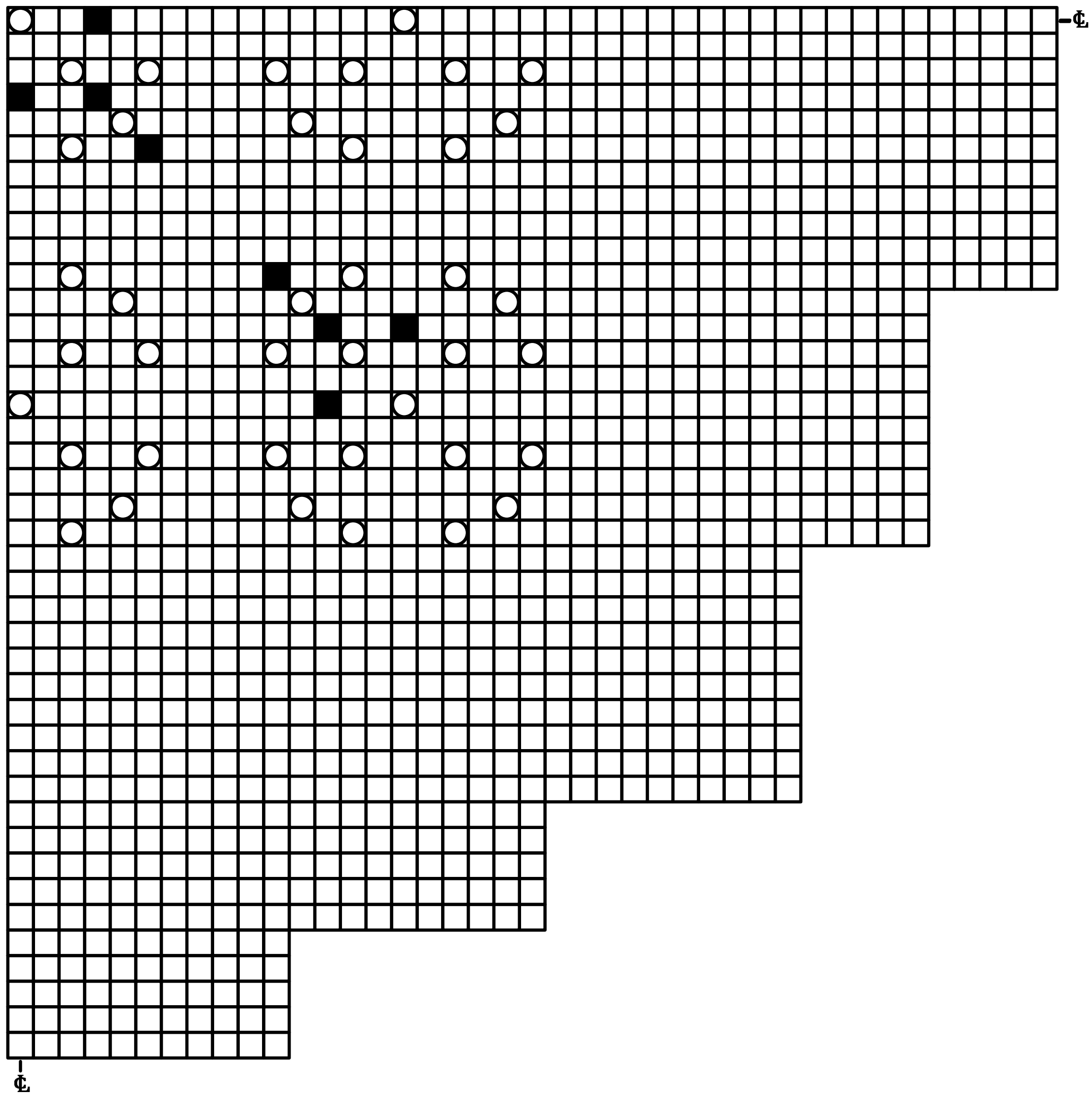
Table 4-9 cont'd
SIMULATE-3 to CASMO-5 Theoretical 2x2 Colorset Comparison Results

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

Mean = (SIMULATE-3 – CASMO-5)/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

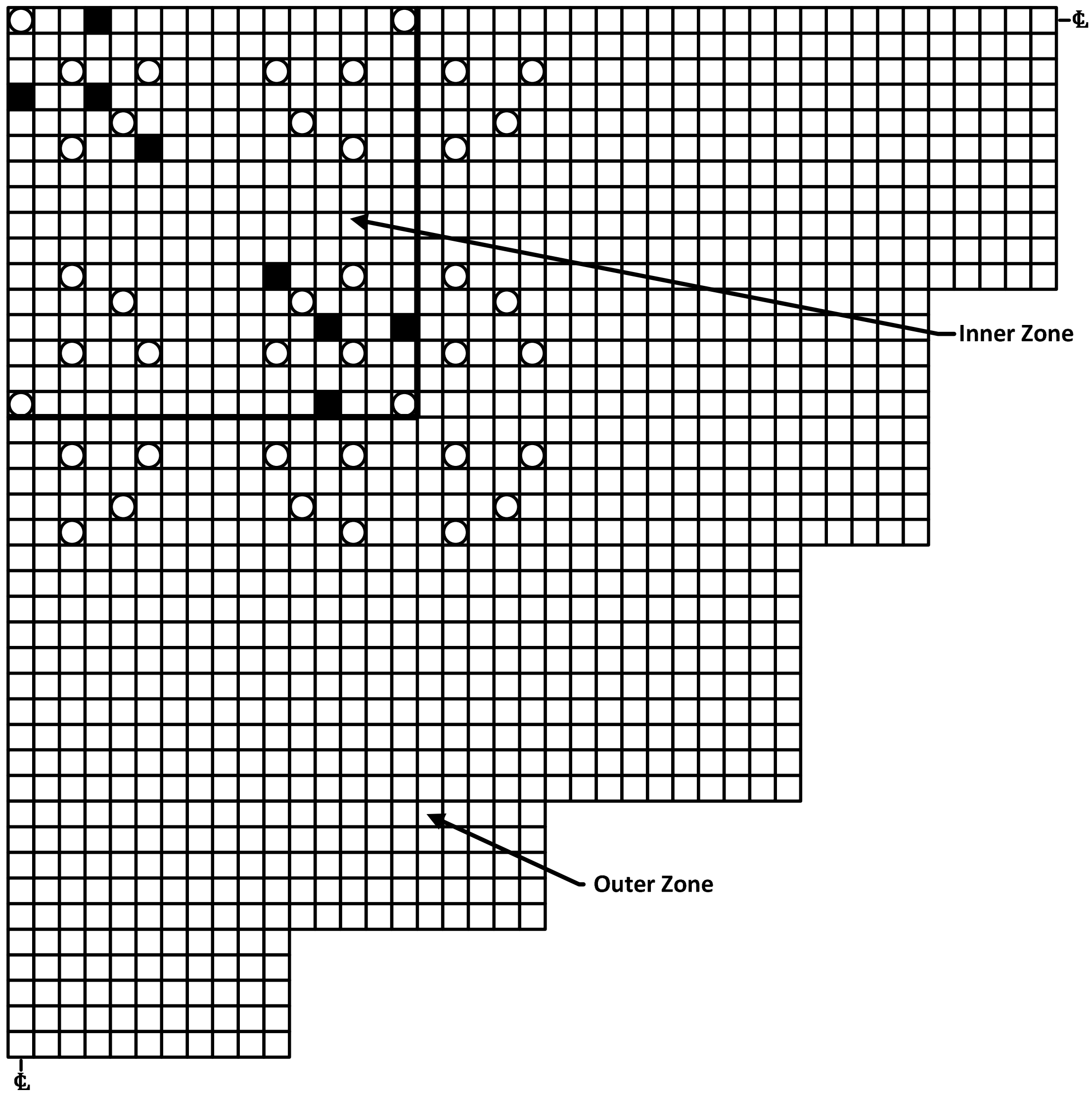
Core 5

VACANT WATER-FILLED POSITION

2.46 wt % U-235 ENRICHED FUEL

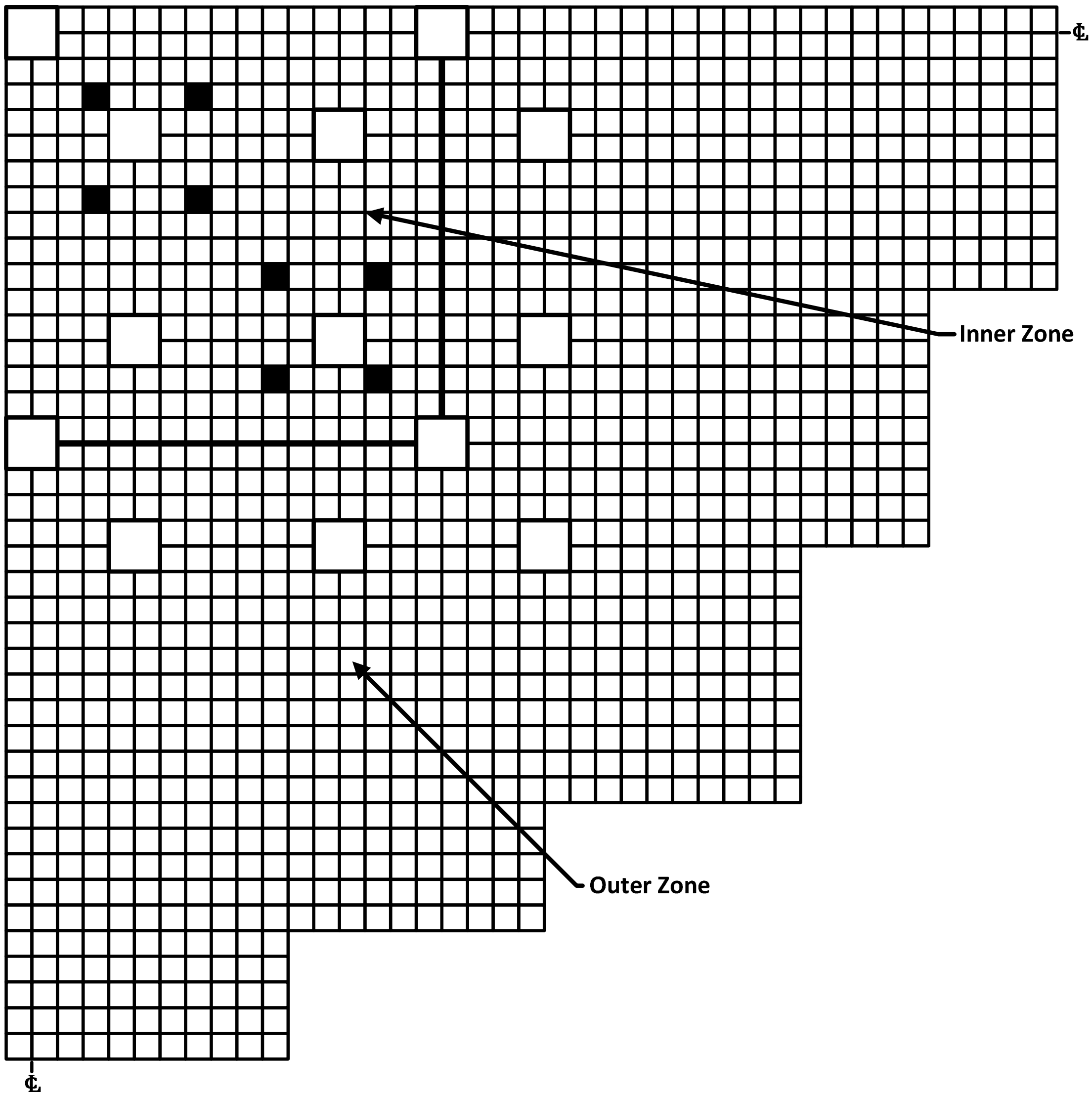
4.00 wt % Gd_2O_3/1.94 wt % U-235 ENRICHED FUEL

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



Core 12	**Core 14**
VACANT WATER-FILLED POSITION	VACANT WATER-FILLED POSITION
2.46 wt % U-235 ENRICHED FUEL: OUTER ZONE	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
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_20_3/1.94 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

a,c

SIMULATE-3 Versus Measured Pin Power Distribution

a,c

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

a,c

SIMULATE-3 Versus Measured Pin Power Distribution

a,c

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

a,c,

SIMULATE-3 Versus Measured Pin Power Distribution

a,c

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

a,c

SIMULATE-3 Versus Measured Pin Power Distribution

a,c

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

a,c

SIMULATE-3 Versus Measured Pin Power Distribution

a,c

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

a,c

SIMULATE-3 Versus Measured Pin Power Distribution

a,c

Figure 4-10
Theoretical 2x2 Colorset Case Matrix

a,c

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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 (σ_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:

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

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:

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

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 [ ]^{a,c} in Section 4.7. This uncertainty is composed of a [ ]^{a,c} bias term and [ ]^{a,c} 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	[ ] a,c (Table 3-6)	[ ] a,c (Table 4-5)	[ ] a,c (Table 3-6)	[ ] a,c (Table 4-5)	[ ] a,c
Fq	[ ] a,c (Table 3-6)	[ ] a,c (Table 4-5)	[ ] a,c (Table 3-6)	[ ] a,c (Table 4-5)	[ ] a,c
Fz	[ ] a,c (Table 3-6)	N/A	[ ] a,c (Table 3-6)	N/A	[ ] a,c

Gadolinia Fuel

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

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 FΔH, Fq and Fz 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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