10 CFR 50.90

Duke Energy®

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U.S. Nuclear Regulatory Commission
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Subject: Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke)
McGuire Nuclear Station, Units 1 and 2; Docket Nos. 50-369, 50-370
Catawba Nuclear Station, Units 1 and 2; Docket Nos. 50-413 and 50-414
License Amendment Request Revising Methodology Report
DPC-NE-1005-P-A, Revision 0, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX* (Proprietary)

References:
1) Letter, K. S. Canady (Duke) to U.S. Nuclear Regulatory Commission, Topical Report DPC-NE-1005-P, Revision 0, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,* August 3,2001.
2) Letter, Robert E. Martin (NRC) to H. B. Barron (Duke), Final Safety Evaluation for Duke Topical Report DPC-NE-1005-P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,* August 20, 2004.
3) Letter, Thomas C. Geer (Duke), to U.S. Nuclear Regulatory Commission, Revision 1 to DPC-NE-1005-P-A, Revision 0, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX* (Proprietary), May 4, 2007.

In accordance with the provisions of 10 CFR 50.90, Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke) is submitting a license amendment request (LAR) for the Renewed Facility Operating Licenses (FOLs) and Updated Final Safety Analysis Reports (UFSARs) for McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2. Specifically, Duke requests NRC review and approval of proposed changes to the FOLs and UFSARs based on Revision 1 to DPC-NE-1005-P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX.*

In Reference 1 Duke submitted methodology report, DPC-NE-1005-P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX,* for NRC staff review and approval. The NRC staff accepted this methodology report and transmitted their safety evaluation to Duke in Reference 2. The original approval of the CASMO-4/SIMULATE-3 MOX methodology was for performing nuclear design calculations for McGuire and Catawba reactor cores containing low enriched uranium fuel, and for the use of up to four MOX lead test assemblies (LTAs) in one of the Catawba units. Reference 3 submitted Revision 1 to this report for the purpose of extending the previously approved methodology to reactor cores containing gadolinia bearing fuel. This revision primarily consists of Appendix B to DPC-NE-1005-P which presents benchmark calculations to

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operating data from reactor cores with fuel containing gadolinia and to data from critical experiments with fuel containing gadolinia. In addition, changes are also made to the original content of this report to address the inclusion of the gadolinia methodology in the report, to correct typographical errors, and to perform editorial revisions to add clarity. Pursuant to 10 CFR 50.32, the information provided in Reference 3 dated May 4, 2007 is hereby incorporated by reference.

Duke is requesting review of Revision 1 of DPC-NE-1005-P to extend the application of the CASMO-4/SIMULATE-3 MOX methodology described in the Updated Final Safety Analysis Reports in order to perform reload core design calculations for reactor cores containing gadolinia. NRC review is requested since the NRC Safety Evaluation (SE) for DPC-NE-1005-P was written with the following contingency: "Introduction of significantly different fuel designs will require further validation of the above stated physics methods for application to Catawba and McGuire by the licensee and will require review by the NRC staff." The introduction of gadolinia bearing fuel rods is considered a significant fuel design change.

The methodology described in Appendix B to DPC-NE-1005-P will be used to perform nuclear design calculations for reactor cores containing gadolinia. The first application will be to support the Catawba 1 Cycle 19 core design with fuel receipt scheduled for November 19, 2009. The transition to the gadolinia burnable absorber design will occur simultaneously with the transition from the Westinghouse Robust Fuel Assembly (RFA) design to AREVA NP's Advanced Mark-BW (ABW) fuel design. This submittal is the first of three submittals required for the ABW fuel transition. The second submittal is the ABW fuel transition report titled: *"McGuire and Catawba Nuclear Stations Advanced Mark-BW Fuel Transition Methodology,"* DPC-NE-2016-P. This report consists of the methodologies to be used by Duke for performing core reload design, fuel assembly mechanical and thermal hydraulic analyses and UFSAR chapter 15 non-LOCA transient and accident analyses, for the transition to the AREVA NP ABW fuel design. The third submittal will contain the LAR to update technical specifications (TS) 2.1.1, 4.2.1, and 5.6.5, in addition to a 10 CFR 50.46 exemption request for M5 cladding to allow transition to the AREVA NP ABW fuel product.

The Enclosure provides Duke's evaluation of the LAR which contains a description of the proposed changes, the technical analysis, the determination that this LAR contains No Significant Hazards Consideration and the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

Attachments 1a and 1b provide marked copies of the affected FOL pages for McGuire and Catawba, showing the proposed changes.

Attachments 2a and 2b provide the retyped affected FOL pages for McGuire and Catawba.

Attachments 3a and 3b provide the existing UFSAR pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, marked-up to show the proposed changes.

Duke is requesting that the NRC review and approve this LAR by May 1, 2008, in order to support the Catawba Unit 1 Cycle 19 core design which will contain gadolinia bearing fuel.

Revisions to the McGuire and Catawba UFSARs necessary to reflect approval of this submittal will be made in accordance with 10 CFR 50.71(e).

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, the proposed amendment has been previously reviewed and approved by the McGuire and Catawba Plant Operations Review Committees and by the Duke Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this LAR has been forwarded to the appropriate State of North Carolina and State of South Carolina officials.

There are no regulatory commitments contained in this letter or its attachments.

If you have any questions or need additional information on this matter, please contact L. B. Jones at (704) 382-4753.

Sincerely,

Thomas C. Geer

Thomas C. Geer

Enclosure

Licensee Evaluation
- Attachment 1 – Licensee Markups
  - Attachment 1a McGuire Nuclear Station, Marked Change to Page 3 in Current FOLs NPF-9 and NPF-17
  - Attachment 1b Catawba Nuclear Station, Marked Change to Page 4 in Current FOLs NPF-35 and NPF-52
- Attachment 2 – Retyped License Pages
  - Attachment 2a McGuire Nuclear Station, Retyped License Page 3 to FOLs NPF-9 and NPF-17
  - Attachment 2b Catawba Nuclear Station, Retyped License Page 4 to FOLs NPF-35 and NPF-52
- Attachment 3 – UFSAR Markups
  - Attachment 3a McGuire Nuclear Station, Marked Pages to UFSAR Chapter 4
  - Attachment 3b Catawba Nuclear Station, Marked Pages to UFSAR Chapter 4

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Thomas C. Geer 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.

Thomas C. Geer
____
Thomas C. Geer

Subscribed and sworn to me: November 12, 2007
Date

Deborah S. Rome   Deborah S. Rome
Notary Public

My Commission Expires: December 19, 2009
Date

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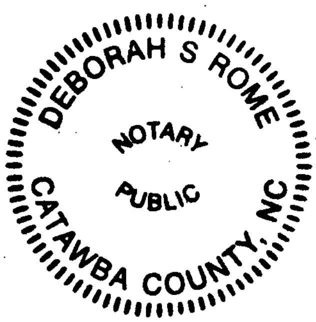

LICENSEE EVALUATION

Subject: License Amendment Request revising FOLs and UFSARs to extend the previously approved methodology to reactor cores containing gadolinia bearing fuel.

LICENSEE EVALUATION

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating Licenses (FOLs) NPF-9 and NPF-17 for McGuire Nuclear Station and NPF-35 and NPF-52 for Catawba Nuclear Station and the Updated Final Safety Analysis Reports (UFSARs) for McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2.

The proposed changes would revise the FOLs and UFSARs to include Revision 1 to DPC-NE-1005-P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX.*

2.0 DETAILED DESCRIPTION

DPC-NE-1005-P describes the analysis methodology used to calculate nuclear physics data and power distribution information for the McGuire and Catawba nuclear units. Revision 0 of the report developed a set of biases and uncertainty factors for use in reload design calculations. The application of this methodology was limited to the analysis of reactor cores containing low enriched uranium and a mixture of low enriched uranium fuel and up to four mixed oxide (MOX) lead test assemblies in one of the Catawba units. The burnable poison types evaluated in the benchmark calculations included lumped burnable poison designs and the zirconium diboride integral fuel burnable absorber design.

Duke intends to use the gadolinia integral fuel burnable absorber in future reload cores for peaking and reactivity control. The NRC Safety Evaluation for DPC-NE-1005-P stated that the "Introduction of significantly different fuel designs will require further validation of the above stated physics methods for application to Catawba and McGuire by the licensee and will require review by the NRC staff." The introduction of gadolinia bearing fuel rods is considered a significant fuel design change. As such, Revision 1 to DPC-NE-1005-P was provided for NRC review in a letter dated May 4, 2007 (Reference 3). This revision described the qualification of the CASMO-4/SIMULATE-3 MOX based core models for analyzing low enriched uranium reactor fuel containing gadolinia integral fuel burnable absorbers.

NRC approval of Revision 1 to DPC-NE-1005-P to extend the use of the CASMO-4/SIMULATE-3 MOX code system to analyze reactor cores containing gadolinia is requested. Subsequent to NRC approval, Section 4.3.3 of the McGuire UFSAR and Catawba UFSAR will be updated to reflect the extension of the CASMO-4/SIMULATE-3 MOX methodology to reactor cores containing gadolinia bearing fuel.

Proposed Changes to FOLs

- Page 3 of the McGuire Unit 1 and Unit 2 FOLs (NPF-9 and NPF-17) is modified to replace references to the latest amendment number.

- Page 4 of the Catawba Unit 1 and Unit 2 FOLs (NPF-35 and NPF-52) is modified to replace references to the latest amendment number.

Proposed Changes to UFSARs

- McGuire UFSAR Section 4.3.3 Analytical Methods – page 4.3-29 and 30, 1st paragraph, will be reworded (*changes in italics*) to read, "The CASMO-3/TABLES-3/*SIMULATE-3P* methodology and CASMO-4/CMS-*LINK/SIMULATE-3* methodology have been approved for use in the nuclear design of a reactor core. These codes were used to generate few group constants and reactor models which can accurately predict the behavior of the reactor core in either two or three dimensions. A description of the methodologies and computer codes used in the evaluation of the reactor core designs are described in the topical reports titled "Nuclear Physics Methodology for Reload Design (Reference 4), "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P" (Reference 53), "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX" (Reference 52), and also in References 48 and 64. These methodologies are NRC approved to perform nuclear design analyses and either CASMO-3P/SIMULATE-3P (Reference 53) or CASMO-4/SIMULATE-3 MOX (Reference 52) methods are applicable *to low enriched uranium (LEU) fuel analyses.* The transition to CASMO-4/SIMULATE-3 MOX method is required to model cores containing mixed oxide fuel (MOX) *within the limitations specified in the safety evaluation contained in Reference 52 or to model cores with fuel containing gadolinia*."

  A minor editorial change will be made above – 1st sentence, "The CASMO-3/TABLES-3/ methodology and CASMO-4/CMS-Link" corrected to "The CASMO-3/TABLES-3/*SIMULATE-3P* methodology and CASMO-4/CMS-*LINK/SIMULATE-3* ."

  A minor editorial change will be made in this paragraph to use only "LEU" instead of "low enriched uranium" since the phrase was defined previously.

- McGuire UFSAR Section 4.3.3.2, page 4.3-30, "Computer Codes for CASMO-3/P Methodology" will be corrected to the more descriptive "CASMO-3/*SIMULATE-3P* Methodology."

- McGuire UFSAR Section 4.3.3.3, Computer Codes for CASMO-4/SIMULATE-3 MOX Methodology – page 4.3-31, 1st paragraph, will be reworded (*insert in italics*) to read, "Another methodology used to perform reload design nuclear calculations is based on CASMO-4 and SIMULATE-3 MOX. This methodology is similar to CASMO-3/SIMULATE-3P methodology described in Section 4.3.3.2 with additional capabilities included to model mixed oxide (MOX) fuel. *The CASMO-4/SIMULATE-3 MOX methodology is also used to model reactor cores with fuel containing gadolinia.*"

- McGuire UFSAR Section 4.3.3.3 – page 4.3-31, 3rd paragraph, will be reworded (*insert in italics*) to read, "SIMULATE-3 MOX is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics model. SIMULATE-3 MOX includes enhancements to model the steep thermal flux gradient between MOX and LEU fuel and is applicable for analysis of ~~all LEU~~ *cores containing LEU fuel with and without gadolinia* or cores containing LEU and MOX LTA fuel. SIMULATE-3 MOX accounts for the effects of fuel and moderator temperature feedback using its nodal thermal-hydraulics model."

Minor editorial change above, 1st sentence "mode" will be changed to "model."

- McGuire UFSAR Section 4.3.3.3 – page 4.3-32, 2nd paragraph, will be reworded (*insert in italics*) to read, "The capability of the SIMULATE-3 MOX code to predict measured power distributions in LEU, *gadolinia* and MOX core designs has been demonstrated by comparisons between measured and predicted power distributions as described in Reference 52. The capability of SIMULATE-3 MOX to predict pin power distributions has been demonstrated through comparison of measured and predicted pin powers for the B&W critical experiments for LEU fuel, *LEU fuel containing gadolinia* and three MOX critical experiments for MOX fuel. These comparisons are described in Reference 52."

  Minor editorial change above, 1st sentence "bee" will be changed to "been."

- McGuire UFSAR Section 4.3.6 References (*changes in italics*) page 4.3-33 and 35

  Item 4, DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design", Rev. ~~1, Oct 1, 2002.~~ *2, June 24, 2003.*

  Item 52, DPC-NE-1005-P-A, Rev ~~0~~1, "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX", SER dated ~~August 20, 2004~~.

  replace with *Rev. 1 SER date when approved*

- Catawba UFSAR Section 4.3.3 Analytical Methods – page 4.3-31, 1st paragraph, will be reworded (*changes in italics*) to read, "A description of the methodologies and computer codes used in the evaluation of the reactor core designs are described in the topical reports titled "Nuclear Physics Methodology for Reload Design" (Reference 19), "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P" (Reference 10), "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX" (Reference 22), and also in References 45 and 55. These methodologies are NRC approved to perform nuclear design analyses and either CASMO-3P/SIMULATE-3P (Reference 10) or CASMO-4/SIMULATE-3 MOX (Reference 22) methods are applicable *to low enriched uranium (LEU) fuel analyses.* The transition to CASMO-4/SIMULATE-3 MOX methods is required to model cores containing mixed oxide fuel (MOX) *within the limitations specified in the safety evaluation contained in Reference 22 or to model cores with fuel containing gadolinia.*"

A minor editorial change will be made in this paragraph to use only "LEU" instead of "low enriched uranium" since the phrase was defined previously.

- Catawba UFSAR Section 4.3.3.2, Computer Codes for Method 2 – page 4.3-32, 1st paragraph, will be reworded (*insert in italics*) to read, "Another methodology used to perform reload design nuclear calculations is based on CASMO-4 and SIMULATE-3 MOX. This methodology is similar to CASMO-3/SIMULATE-3P methodology described in Section 4.3.3.1 with additional capabilities included to model mixed oxide (MOX) fuel. *The CASMO-4/SIMULATE-3 MOX methodology is also used to model reactor cores with fuel containing gadolinia.*"

- Catawba UFSAR Section 4.3.3.2 – page 4.3-32, 3rd paragraph, will be reworded (*insert in italics*) to read, "SIMULATE-3 MOX is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics model. SIMULATE-3 MOX includes enhancements to model the steep thermal flux gradient between MOX and LEU fuel and is applicable for analysis of ~~all LEU~~ *cores containing LEU fuel with and without gadolinia* or cores containing LEU and MOX LTA fuel."

- Catawba UFSAR Section 4.3.3.2 – page 4.3-32 and 33, 5th paragraph, will be reworded (*insert in italics*) to read, "The capability of the SIMULATE-3 MOX code to predict measured power distributions in LEU*, gadolinia* and MOX core designs has been demonstrated by comparisons between measured and predicted power distributions as described in Reference 22. The capability of SIMULATE-3 MOX to predict pin power distributions has been demonstrated through comparison of measured and predicted pin powers for the B&W critical experiments for LEU fuel*, LEU fuel containing gadolinia* and three MOX critical experiments for MOX fuel. These comparisons are described in Reference 22."

- Catawba Section 4.3.6 References (*changes in italics*) page 4.3-34

  Item 19, DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station, Catawba Nuclear Station Nuclear Physics Methodology for Reload Design", Rev. ~~1, Oct 1, 2002.~~ *2, June 24, 2003.*

  Item 22, DPC-NE-1005-P-A, Rev ~~0~~1, "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX", SER dated ~~August 20, 2004~~.

  replace with *Rev. 1 SER date when approved*

  Minor editorial change above, "SUMULATE" corrected to "SIMULATE."

3.0 TECHNICAL EVALUATION

McGuire and Catawba Section 4.3.3 Analytical Methods Changes

Section 4.3.3 of the McGuire UFSAR, Units 1 and 2 and Section 4.3.3 of the Catawba UFSAR, Units 1 and 2 describe the methodology used to perform reload design calculations. The current methodology is based on the CASMO-4/SIMULATE-3 MOX code system, and is applicable to reactor cores containing low enriched uranium fuel, and up to four MOX LTAs in one of the Catawba units. A description of both the CASMO-4 and SIMULATE-3 MOX codes is provided. The types of calculations that are performed and the validation of the code systems' performance in predicting core reactivity and power distributions are described. Finally, the accuracy of the analytical methods is provided.

The proposed UFSAR changes pertain to extending the application of the CASMO-4/SIMULATE-3 MOX code system for analysis of reactor cores with fuel containing gadolinia. The markups to these UFSAR sections are contained in Attachments 3a and 3b. The NRC approval of Revision 1 to DPC-NE-1005-P will provide the necessary technical justification for updating the UFSARs.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The applicable regulatory requirements for Reactor Design are defined in 10 CFR 50, Appendix A, Criterion 10. This LAR is being submitted in accordance with 10 CFR 50.90.

4.2 Precedent

The CASMO-4/SIMULATE-3 code system has been previously approved by the NRC for analyzing reactor cores with gadolinia in an analysis for the Prairie Island Nuclear Power Plant. The topical report, "Qualification of Reactor Physics Methods for Application to Prairie Island, NSPNAD-8101-A, Revision 2," was approved by an NRC Safety Evaluation dated September 13, 2000.

4.3 Significant Hazards Consideration

The proposed amendment would revise the Renewed Facility Operating Licenses (FOLs) and Updated Final Safety Analysis Reports (UFSARs) for McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2 to extend the previously approved methodology to reactor cores with fuel containing gadolinia.

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

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

Response: No.

The proposed UFSAR change to allow the use of the CASMO-4/SIMULATE-3 MOX reload design software to analyze reactor cores with fuel containing gadolinia does not involve a significant increase in the probability or consequences of an accident previously evaluated. The CASMO-4 and SIMULATE-3 MOX codes are used to perform reactivity and power distribution calculations to develop power distribution limits and provide confirmation of reactivity and power distribution input assumptions used in the evaluation of UFSAR Chapter 15 accidents. The SIMULATE-3 MOX code is also used to confirm the acceptability of thermal limits at post accident conditions. Since the CASMO-4/SIMULATE-3 MOX software is not used in the operation of any plant equipment, the probability of an accident previously evaluated in the UFSAR is not increased.

The benchmark calculations performed in Revision 1 to DPC-NE-1005-P verified the acceptability of the CASMO-4/SIMULATE-3 MOX codes for performing reload design calculations for reactor cores containing gadolinia. These calculations confirmed the accuracy of the codes and developed a methodology for calculating power distribution uncertainties for use in reload design calculations. The use of power distribution uncertainties applicable to gadolinia core designs in conjunction with predicted peaking factors ensures that thermal accident acceptance criteria are satisfied.

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

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

Response: No.

The extension of the reload design software to perform reload design calculations for reactor cores containing gadolinia will not create the possibility of a new or different kind of accident from any accident previously evaluated. The CASMO-4/SIMULATE-3 MOX 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 evaluation of UFSAR accidents and the associated acceptance criteria for these accidents remains unchanged.

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

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

Response: No.

The extension of the CASMO-4/SIMULATE-3 MOX reload design software to perform reload design calculations for reactor cores containing gadolinia will not involve a significant reduction in a margin of safety.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design function during and following an accident. These barriers include the fuel cladding, the reactor coolant system and the containment system. The reload design process assures the acceptability of thermal limits under normal, transient, and accident conditions. The CASMO-4/SIMULATE-3 MOX reload design software was qualified for the analysis of reactor cores containing gadolinia in Revision 1 to DPC-NE-1005-P and a methodology for developing appropriate power distribution uncertainties for application in reload design analyses was developed. The use of these uncertainties for analysis of reload cores with gadolinia ensures that design and safety limits are satisfied such that the fission product barriers perform their design function.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the preceding discussion, Duke concludes that the proposed amendment does not involve a 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

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations. Implementation of this amendment will have no adverse impact upon the McGuire or Catawba units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration.
2. No significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and
3. No significant increase in individual or cumulative occupational radiation exposure.

Therefore, this amendment to the McGuire and Catawba Nuclear Station Renewed Facility Operating Licenses meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

ATTACHMENT 1a

McGuire Nuclear Station, Marked Change to Page 3 in Current FOLs NPF-9 and NPF-17

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2, and;

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3411 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 236 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than June 12, 2021, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3411 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 218 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than March 3, 2023, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

ATTACHMENT 1b

Catawba Nuclear Station, Marked Change to Page 4 in Current FOLs NPF-35 and NPF-52

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No 236 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Power Company LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Power Company LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Power Company LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplement wherein this renewed license condition is discussed.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 232 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Power Company LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Power Company LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Power Company LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

ATTACHMENT 2a

McGuire Nuclear Station, Retyped License Page 3 to FOLs NPF-9 and NPF-17

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2, and;

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3411 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than June 12, 2021, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3411 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than March 3, 2023, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

ATTACHMENT 2b

Catawba Nuclear Station, Retyped License Page 4 to FOLs NPF-35 and NPF-52

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No which are attached hereto, are hereby incorporated into this renewed operating license. Duke Power Company LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Power Company LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Power Company LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplement wherein this renewed license condition is discussed.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No which are attached hereto, are hereby incorporated into this renewed operating license. Duke Power Company LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Power Company LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Power Company LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

ATTACHMENT 3a

McGuire Nuclear Station, Marked Pages to UFSAR Chapter 4

power, and was performed at cycle average burnups corresponding to BOC, MOC and EOC. The xenon transients shown in Figure 4-87 assume that control rods were held at their initial position for the duration of the transient.

2. Radial Power Distribution

   The core described herein has been calculated to be stable against X-Y xenon-induced oscillations at all times in life.

   The X-Y stability of large PWRs has been further verified as part of the startup physics test program for PWR cores with 193 fuel assemblies. The measured X-Y stability of the PWR core with 157 assemblies was in good agreement with the calculated stability as discussed in Sections 4.3.2.7.4 and 4.3.2.7.5. In the unlikely event that X-Y oscillations occur, actions can be taken to increase the natural stability of the core. This is based on the fact that several actions could be taken to make the moderator temperature coefficient more negative, which will increase the stability of the core in the X-Y plane.

   Provisions for protection against asymmetric perturbations in the X-Y power distribution that could result from equipment malfunctions are made in the protection system design. These include control rod drop, rod misalignment and asymmetric loss of coolant flow.

   A more detailed discussion of the power distribution control in PWR cores is presented in References 6 and 7.

4.3.2.8 Vessel Irradiation

A brief review of the methods and analyses used in the determination of neutron and gamma ray flux attenuation between the core and the pressure vessel is given below. A more complete discussion on the pressure vessel irradiation and surveillance program is given in Section 5.4.3.7.

The materials that serve to attenuate neutrons originating in the core and gamma rays from both the core and structural components consist of the core baffle, core barrel, neutron pads, and associated water annuli, all of which are within the region between the core and the pressure vessel.

In general, few group neutron diffusion theory codes or advanced nodal codes are used to determine fission power density distributions within the active core and the accuracy of these analyses is verified by incore measurements on operating reactors. Region and rodwise power sharing information from the core calculations is then used as source information in two-dimensional S_n transport calculations which compute the flux distributions throughout the reactor. Outside the active core, methods such as those which use multigroup space dependent slowing down codes described in Section 5.4.3.7 are used. Regionwise power sharing information from the core calculations is often used as reference source data for the multigroup codes.

The neutron flux distribution and spectrum in the various structural components varies significantly from the core to the pressure vessel.

As discussed in Section 5.4.3.7, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel.

4.3.3 Analytical Methods

SIMULATE-3P LINK/SIMULATE-3

The CASMO-3/TABLES-3/ methodology and CASMO-4/CMS-~~Link~~ methodology have been approved for use in the nuclear design of a reactor core. These codes were used to generate few group constants and reactor models which can accurately predict the behavior of the reactor core in either two or three dimensions. A description of the methodologies and computer codes used in the evaluation of the reactor core designs are described in the topical reports titled "Nuclear Physics Methodology for Reload Design" to low enriched uranium(LEU) fuel analyses,

(Reference 4), "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P" (Reference 53), "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX" (Reference 52), and also in References 48 and 64. These methodologies are NRC approved to perform nuclear design analyses and either CASMO-3P/SIMULATE-3P (Reference 53) or CASMO-4/SIMULATE-3 MOX (Reference 52) methods are applicable. The transition to CASMO-4/SIMULATE-3 MOX method is required to model cores containing mixed oxide fuel (MOX). This methodology is also applicable to cores containing only ~~low enriched uranium~~ (LEU) fuel. Since many generic analyses were performed with CASMO-3/SIMULATE-3P methods, this methodology will be retained in the UFSAR even after transition. An overview of the nuclear design analyses performed as part of the licensing basis of each reload core design follows. Details pertaining to the analyses performed can be found in the referenced topical reports.

The design of a reload core initially requires the development of a preliminary loading pattern which satisfies desired energy, feed batch size and enrichment requirements. Following this initial step, analyses are performed to ensure that applicable safety, fuel mechanical and thermal limits are also satisfied. Calculation of these limits are performed using NRC approved thermal hydraulic, system thermal hydraulic (e.g. RETRAN) and space-time kinetics transient analysis codes. A conservative set of safety, mechanical or thermal limits are determined and assured through the selection of conservative initial conditions, boundary conditions, code options, key physics parameters and core thermal hydraulic models. Key physics parameters, which are identified for each analysis, are calculated for each reload core and verified to be bounded by the values used in the licensing analysis. The confirmation of cycle-specific values of the key physics parameters relative to the values used in the licensing analysis ensures that the analyses performed to establish safety, mechanical and thermal limits bound the reload core. The method employed to select the key physics parameters important to each Chapter 15 event are described in References 48 and 51.

within the limitations specified in the safety evaluation contained in Reference 52 or to model cores with fuel containing gadolinia

4.3.3.1 Computer Codes for ARMP Methodology

SIMULATE-3P

4.3.3.2 Computer Codes for CASMO-3/P Methodology

The methodology used to perform reload design nuclear calculations is based on CASMO-3 and SIMULATE-3P. The computer codes used are described as follows.

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

SIMULATE-3P is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics model. SIMULATE-3P accounts for the effects of fuel and moderator temperature feedback using its nodal thermal-hydraulics model. The program explicitly models the baffle and reflector region. The program uses data from CASMO-3 for each pin in the fuel assembly and uses inter-assembly and intra-assembly data obtained from the coarse mesh solution to reconstruct the power distribution for each pin.

The primary uses of this program include the calculation of critical boron concentrations, control rod worths, reactivity coefficients, boron worths, kinetics data and the time dependent behavior of the xenon distribution following a change in reactor power, or perturbation in the three-dimensional power distribution. Shutdown margin, and ejected and stuck rod worth calculations are also performed with this code.

The capability of the SIMULATE-3P code to predict measured power distributions has been demonstrated by comparisons between measured and predicted power distributions as described in References 53 and 64. The capability of SIMULATE-3P to predict pin power distributions has been demonstrated through comparison of measured and predicted pin powers for the B&W critical experiments. These comparisons are also described and discussed in Reference 53.

Predicted versus measured reactivity comparisons are contained in Reference 53 and are also performed as part of the startup and physics testing program at the beginning of each cycle. The predictive capability of SIMULATE-3P is also assessed through core follow power distribution and critical boron concentration comparisons and the evaluation of startup conditions following a reactor trip.

Based on comparison with measured data, it is estimated that the accuracy of current analytical methods is:

±0.2 percent $\Delta\rho$ for the Doppler power defect

±2 pcm/°F for moderator temperature coefficient

±50 ppm for critical boron concentration with depletion

±3 percent for power distributions

±0.2 percent $\Delta\rho$ for rod bank worth

±4 pcm/step for the differential rod worth

±0.5 pcm/ppm for boron worth

±0.1 percent $\Delta\rho$ for the moderator defect

4.3.3.3 Computer Codes for CASMO-4/SIMULATE-3 MOX Methodology

Another methodology used to perform reload design nuclear calculations is based on CASMO-4 and SIMULATE-3 MOX. This methodology is similar to CASMO-3/SIMULATE-3P methodology described in Section 4.3.3.2 with additional capabilities included to model mixed oxide (MOX) fuel. The computer codes used are described as follows.

Insert A

CASMO-4 is a multigroup two-dimensional transport theory code for fuel assembly burnup ~~calculations~~. It uses a library of 70 energy group cross sections based primarily on the ENDF/B-IV data base. Certain data used in the CASMO-4 library, such as the Xe-135 yields, and fission spectra data for U-235 and Pu-239, as well as data for Ag, Gd, Er, and Tm are taken from ENDF/B-V. Data for Pu-241 was taken from JENDL-2. This code produces two-group cross sections, assembly discontinuity factors, fission product data, detector reaction rates, and pin power data. The data from CASMO-4 is reformatted into two- or three-dimensional tables using a data processing program, CMS-LINK, for input to the three-dimensional code SIMULATE-3 MOX. SIMULATE-3 MOX interpolates the data from CMS-LINK for the independent variables for certain core conditions that SIMULATE-3 MOX models.

model

SIMULATE-3 MOX is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics ~~mode~~. SIMULATE-3 MOX includes enhancements to model the steep thermal flux gradient between MOX and LEU fuel and is applicable for analysis of all ~~LEU~~ or cores containing LEU and MOX LTA fuel. SIMULATE-3 MOX accounts for the effects of fuel and moderator temperature feedback using its nodal thermal-hydraulics model. The program explicitly models the baffle and reflector region. The program uses data from CASMO-4 for each pin in the fuel assembly and uses inter-assembly and intra-assembly flux data obtained from the coarse mesh solution to reconstruct the power distribution for each pin.

cores containing LEU fuel with and without gadolinia

Insert A

The CASMO-4/SIMULATE-3 MOX methodology is also used to model reactor cores with fuel containing gadolinia.

The primary uses of this program include the calculation of critical boron concentrations, control rod worths, reactivity coefficients, boron worths, kinetics data and the time dependent behavior of the xenon distribution following a change in reactor power, or perturbation in the three-dimensional power distribution. Shutdown margin, and ejected and stuck rod worth calculations are also performed with this code.

been , gadolinia

The capability of the SIMULATE-3 MOX code to predict measured power distributions in LEU and MOX core designs has bee demonstrated by comparisons between measured and predicted power distributions as described in Reference 52. The capability of SIMULATE-3 MOX to predict pin power distributions has been demonstrated through comparison of measured and predicted pin powers for the B&W critical experiments for LEU fuel and ~~three MOX critical experiments for MOX fuel~~. These comparisons are described in Reference 52.

, LEU fuel containing gadolinia

Predicted versus measured reactivity comparisons are contained in Reference 52 and are also performed as part of the startup and physics testing program at the beginning of each cycle. The predictive capability of SIMULATE-3 MOX is also assessed through core follow power distribution and critical boron concentration comparisons and the evaluation of startup conditions following a reactor trip.

Based on comparison with measured data, it is estimated that the accuracy of current analytical methods is:

- ±0.2 percent $\Delta\rho$ for the Doppler power defect
- ±2 pcm/°F for moderator temperature coefficient
- ±50 ppm for critical boron concentration with depletion
- ±3 percent for power distributions
- ±0.2 percent $\Delta\rho$ for rod bank worth
- ±4 pcm/step for the differential rod worth
- ±0.5 pcm/ppm for boron worth
- ±0.1 percent $\Delta\rho$ for the moderator defect

4.3.4 Deleted Per 2003 Update

4.3.5 Changes

[Include a discussion of changes, as required by Reg Guide 1.70, Standard Format.]

Section 4.3.4 was removed for the following reasons:

1. It is not a major area of nuclear design, such as Section 4.3.1 – Design Bases, Section 4.3.2 – Description, and Section 4.3.3 – Analytical Methods.
2. The utilization of fuel temperature data in a reactor physics code should appropriately characterize core reactivity and feedback effects. The primary function of a fuel performance code is to conservatively characterize the fuel temperature from a mechanical point of view. Although a fuel performance code may be used to develop fuel temperature data for use in a reactor physics code, it is intended that the use of fuel temperature data will appropriately characterize neutronic behavior.
3. Topical Report DPC-NF-2010, contained a description of the generation of fuel temperature data used as input to the neutronics code. This statement was removed, and the NRC approved Revision 2 of this topical report by letter dated June 24, 2003.

4.3.6 References (HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED)

1. *"Westinghouse Anticipated Transients Without Reactor Trip Analysis", WCAP-8330, August 1974.*
2. *Spier, E. M. "Evaluation of Nuclear Hot Channel Factor Uncertainties", WCAP-7308-L-P-A (Proprietary) June 1988, and WCAP-7810-A, June 1988.*
3. Deleted Per 1999 Update.
4. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design", Rev. 1, ~~Oct~~ 1, ~~2002~~. 2, June 24, 2003.
5. Deleted Per 1999 Update.
6. *Moore, J. S., "Power Distribution Control of Westinghouse Pressurized Water Reactors", WCAP-7208 (Proprietary), September 1968 and WCAP-7811, December 1971.*
7. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", Rev. 1, Oct 1, 2002 (DPC Proprietary).
8. Deleted Per 1999 Update.
9. Deleted Per 1994 Update.
10. *McFarlane, A. F., "Topical Report, Power Peaking Factors", WCAP-7912-P-A (Proprietary) and WCAP-7912-A, January 1975.*
11. *Altomare, S. and Barry, R. G., "The TURTLE 24.0 Diffusion Depletion Code", WCAP-7213-P-A (Proprietary) and WCAP-7758-A, February, 1975.*
12. *Cermak, J. O., et al., "Pressurized Water Reactor pH - Reactivity Effect Final Report", WCAP-3696-8 (EURAEC-2074), October 1968.*
13. Deleted Per 1999 Update.
14. Deleted Per 1999 Update.
15. *Poncelet, C. G. and Christie, A. M, "Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors", WCAP-3680-20, (EURAEC-1974), March 1968.*
16. *Skogen, F. B. and McFarlane, A. F., "Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors", WCAP-3680-21, (EURAEC-2111), February 1969.*
17. *Skogen, F. B. and McFarlane, A. F., "Xenon-Induced Spatial Instabilities in Three-Dimensions", WCAP-3680-22 (EURAEC-2116), September 1969.*
18. *Lee, J. C., et al., "Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor", WCAP-7964, June 1971.*
19. *Altomare, S. and Minton, G., et al., "The PANDA Code", WCAP-7048-P-A (Proprietary) and WCAP-7757-A, Febrary 1975.*
20. *Barry, R. F., "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094", WCAP-3269-26, September 1963.*
21. *England, T. R., "CINDER - A One-Point Depletion and Fission Product Program", WAPD-TM-334, August 1962.*
22. *Eggleston, F. T., "Topical Report, Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Spring 1976", WCAP-8768, June 1976.*
23. Deleted Per 1998 Update.

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52. DPC-NE-1005-P-A, Rev 0 "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX", SER dated August 20, 2004. Replace with Rev. 1 SER date when approved

53. DPC-NE-1004A, Rev 1, "Nuclear Design Methodology Using CASMO-3/P," SER dated April 26, 1996.

54. Duke Power Company, "DETECTOR User's Manual", COM-0204.C6-10-0197, March 1993.

55. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3", SER dated April 3, 1995 (DPC Proprietary).

56. Duke Power Company, "Core Monitoring and Evaluation of Technical Specifications", Documented in NSD-800, Controlled Program List.

57. Deleted Per 2002 Update.

58. Deleted Per 2002 Update.

59. Deleted Per 2002 Update.

60. "Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.31 Wt% ^{235}U Enriched UO_2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6," Pacific Northwest Laboratory, NUREG/CR-1547, PNL-3314, July 1980.

61. "Critical Separation Between Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," Pacific Northwest Laboratory, PNL-2438, October 1977.

62. "Critical Experiments to Provide Benchmark Data on Neutron Flux Traps," S. R. Bierman, PNL-6205, June 1988.

63. Deleted Per 2006 Update.

64. DPC-NE-2009 P-A, "Westinghouse Fuel Transition Report," Rev 2, Dec 18, 2002.

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66. Oak Ridge National Laboratory, SCALE 4.4 – A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200 (Rev. 5), CCC-545, March 1997.

67. Letter from G. Peterson (Duke) to U.S. NRC, "McGuire Nuclear Station Units 1 and 2, Proposed Technical Specification (TS) Amendments, TS 3.7.15 – Spent Fuel Assembly Storage, and TS 4.3 – Fuel Storage" – September 29, 2003.

68. Letter from J. Shea (U.S. NRC) to G. Peterson (Duke), "McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos MC0945 and MC0946)" – March 17, 2005.

THIS IS THE LAST PAGE OF THE TEXT SECTION 4.3.

ATTACHMENT 3b

Catawba Nuclear Station, Marked Pages to UFSAR Chapter 4

to low enriched uranium (LEU) fuel analyses,

As discussed in Section 5.3, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel. within the limitations specified in the safety evaluation contained in Reference 22 or to model cores with fuel containing gadolinia.

4.3.3 Analytical Methods

A description of the methodologies and computer codes used in the evaluation of the reactor core designs are described in the topical reports titled "Nuclear Physics Methodology for Reload Design" (Reference 19), "Nuclear Design Methodology Using CASMO-3/SIMULATE -3P" (Reference 10), "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX" (Reference 22), and also in Reference 45 and 55. These methodologies are NRC approved to perform nuclear design analyses and either CASMO-~~3P/SIMULATE-3P~~ (Reference 10) or CASMO-4/SIMULATE-3 MOX (Reference 22) methods are applicable. The transition to CASMO-4/SIMULATE-3 MOX methods is required to model cores containing mixed oxide fuel (MOX). This methodology is also applicable to cores containing only ~~low enriched uranium~~ (LEU) fuel. Since many generic analyses were performed with CASMO-3/SIMULATE-3P methods, this methodology will be retained in the UFSAR even after transition. An overview of the nuclear design analyses performed as part of the licensing basis of each reload core design follows. Details pertaining to the analyses performed can be found in the referenced topical reports.

The design of a reload core initially requires the development of a preliminary loading pattern which satisfies desired energy, feed batch size and enrichment requirements. Following this initial step, analyses are performed to ensure that applicable safety, fuel mechanical and thermal limits are also satisfied. Calculation of these limits are performed using NRC approved thermal hydraulic, system thermal hydraulic (e.g. RETRAN) and space-time kinetics transient analysis codes. A conservative set of safety, mechanical or thermal limits are determined and assured through the selection of conservative initial conditions, boundary conditions, code options, key physics parameters and core thermal hydraulic models. Key physics parameters, which are identified for each analysis, are calculated for each reload core and verified to be bounded by the values used in the licensing analysis. The confirmation cycle-specific values of the key physics parameters relative to the values used in the licensing analysis ensures that the analyses performed to establish safety, mechanical and thermal limits bound the reload core. The method employed to select the key physics parameters important to each Chapter 15 event are described in References 45 and 46.

4.3.3.1 Computer Codes

The methodology used to perform reload design nuclear calculations is based on CASMO-3 and SIMULATE-3. The computer codes used are described as follows.

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

SIMULATE-3 is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics model. SIMULATE-3 accounts for the effects of fuel and moderator temperature feedback using a nodal thermal-hydraulics model. The program explicitly models the baffle and reflector region. The program uses data from CASMO-3 for each pin in the fuel assembly and uses inter-assembly and intra-assembly data obtained from the coarse mesh solution to reconstruct the power distribution for each pin.

The primary uses of this program include the calculation of critical boron concentrations, control rod worths, reactivity coefficients, boron worths, kinetics data and the time dependent behavior of the xenon distribution following a change in reactor power, or perturbation in the three-dimensional power distribution. Shutdown margin, and ejected and stuck rod worth calculations are also performed with SIMULATE-3.

The capability of the SIMULATE-3 code to predict measured power distributions has been demonstrated by comparisons between measured and predicted power distributions as described in Reference 10. The capability of SIMULATE-3 to predict pin power distributions has been demonstrated through comparison of measured and predicted pin powers for the B&W critical experiments. These comparisons are also described and discussed in Reference 10.

Predicted versus measured reactivity comparisons are contained in Reference 10 and are also performed as part of the startup and physics testing program at the beginning of each cycle. The predictive capability of SIMULATE-3 is also assessed through core follow power distribution and critical boron concentration comparisons and the evaluation of startup conditions following a reactor trip.

The estimated accuracy of these analytical methods are described in the appropriate Topical Reports (References 10, 22, and 55.)

4.3.3.2 Computer Codes For Method 2

Another methodology used to perform reload design nuclear calculations is based on CASMO-4 and SIMULATE-3 MOX. This methodology is similar to CASMO-3/SIMULATE-3P methodology described in Section 4.3.3.1 with additional capabilities included to model mixed oxide (MOX) fuel. The computer codes used are described as follows.

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CASMO-4 is a multigroup two-dimensional transport theory code for fuel assembly burnup calculations. It uses a library of 70 energy group cross sections based primarily on the ENDF/B-IV data base. Certain data used in the CASMO-4 library, such as the Xe-135 yields, and fission spectra data for U-235 and Pu-239, as well as data for Ag, Gd, Er, and Tm are taken from ENDF/B-V. Data for Pu-241 was taken from JENDL-2. This code produces two-group cross sections, assembly discontinuity factors, fission product data, detector reaction rates, and pin power data. The data from CASMO-4 is reformatted into two- or three-dimensional tables using a data processing program, CMS-LINK, for input to the three-dimensional code SIMULATE-3 MOX. SIMULATE-3 MOX interpolates the data from CMS-LINK for the independent variables for certain core conditions that SIMULATE-3 MOX models.

SIMULATE-3 MOX is a two-group three-dimensional coarse mesh diffusion theory code based on the QPANDA neutronics model. SIMULATE-3 MOX includes enhancements to model the steep thermal flux gradient between MOX and LEU fuel and is applicable for analysis of all LEU or cores containing LEU fuel and MOX LTA fuel. SIMULATE-3 MOX accounts for the effects of fuel and moderator temperature feedback using its nodal thermal-hydraulics model. The program explicitly models the baffle and reflector region. The program uses data from CASMO-4 for each pin in the fuel assembly and uses inter-assembly and intra-assembly flux data obtained from the coarse mesh solution to reconstruct the power distribution for each pin.

cores containing LEU fuel with and without gadolinia

The primary uses of this program include the calculation of critical boron concentrations, control rod worths, reactivity coefficients, boron worths, kinetics data and the time dependent behavior of the xenon distribution following a change in reactor power, or perturbation in the three-dimensional power distribution. Shutdown margin, and ejected and stuck rod worth calculations are also ~~performed with this~~ code.

, gadolinia

The capability of the SIMULATE-3 MOX code to predict measured power distributions in LEU and MOX core designs has been demonstrated by comparisons between measured and predicted power distributions as described in Reference 22. The capability of SIMULATE-3 MOX to predict pin power

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The CASMO-4/SIMULATE-3 MOX methodology is also used to model reactor cores with fuel containing gadolinia.

LEU fuel containing gadolina

distributions has been demonstrated through comparison of measured and predicted pin powers for the B&W critical experiments for LEU fuel and three MOX critical experiments for MOX fuel. These comparisons are described in Reference 22.

Predicted versus measured reactivity comparisons are contained in Reference 22 and are also performed as part of the startup and physics testing program at the beginning of each cycle. The predictive capability of SIMULATE-3 MOX is also assessed through core follow power distribution and critical boron concentration comparisons and the evaluation of startup conditions following a reactor trip.

Based on comparison with measured data, it is estimated that the accuracy of current analytical methods is:

±0.2 percent Δρ for the Doppler power defect
±2 pcm/°F for moderator temperature coefficient
±50 ppm for critical boron concentration with depletion
±3 percentg for power distributions
±0.2 percent Δρ for rod bank worth
±4 pcm/step for the differential rod worth
±0.5 pcm/ppm for boron worth
±0.1 percent Δρ for the moderator defect

4.3.4 Deleted Per 2004 Update

4.3.5 Changes

Section 4.3.4 was removed for the following reasons:

1. It is not a major area of nuclear design, such as Section 4.3.1 – Design Bases, Section 4.3.2 – Description, and Section 4.3.3 – Analytical Methods.
2. The utilization of fuel temperature data in a reactor physics code should appropriately characterize core reactivity and feedback effects. The primary function of a fuel performance code is to conservatively characterize the fuel temperature from a mechanical point of view. Although a fuel performance code may be used to develop fuel temperature data for use in a reactor physics code, it is intended that the use of fuel temperature data will appropriately characterize neutronic behavior.
3. Topical Report DPC-NF-2010, contained a description of the generation of fuel temperature data used as input to the neutronics code. This statement was removed, and the NRC approved Revision 2 of this topical report by letter dated June 24, 2003.

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SIMULATE

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