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October 19, 2009

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

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Proposed License Amendment Request to Revise the Technical Specifications
Pursuant to the Use of Gadolinia Integral Burnable Absorber
License Amendment Request Number 2009-12

Duke Energy Carolinas, LLC (Duke) hereby submits a license amendment request (LAR) for the Oconee Nuclear Station (ONS) Renewed Facility Operating License (FOL). Specifically, Duke requests Nuclear Regulatory Commission (NRC) review and approval for usage of gadolinia as an integral burnable neutron absorber in the uranium oxide fuel matrix. The proposed change would revise Technical Specification (TS) 2.1.1, Reactor Core Safety Limits and TS 5.6.5.b, the Core Operating Limit Report and Duke's NRC-approved methodology reports for reload design and non-Loss Of Coolant Accident (LOCA) safety analyses to allow use of gadolinia. The associated TS Bases are also provided.

Enclosure 3 provides the proprietary evaluation of the proposed TS and methodology revisions and associated technical justification. Other methodology revisions are also included to enhance the existing methodologies, to correct errors, and for editorial clarification. Enclosure 4 provides the non-proprietary evaluation (redacted version) of the proposed change. Attachments 1 and 2 contain the TS and TS Bases Mark-Ups and Reprinted Pages, respectively. Revisions to the six methodology reports are included in Attachments 3-8.

Published versions of the above reports are available upon request. It is Duke's intent to publish the revised version of each of the above methodology reports following NRC approval.

DPC-NE-1006 was submitted to the NRC in an LAR dated, June 10, 2009 (ML091630712) and is required for modeling gadolinia. That methodology validates the code suite and calculates pin power uncertainties for fuel containing gadolinia. Approval of this report is required prior to loading fuel containing gadolinia and is added by reference in some of the methodology reports included in Attachments 3-8 of this LAR. Currently, AREVA NP methodology report, BAW-10192, Revision 2 (ML083460314 and ML083460315) is being reviewed by the NRC. Approval of this report is also required prior to loading fuel containing gadolinia.

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NRR

This report contains information that is proprietary to Duke and AREVA NP. Duke requests that this information be withheld from public disclosure in accordance with 10 CFR 2.390. Affidavits are included (Enclosures 1 and 2) from each organization attesting to the proprietary nature of the information in Enclosure 3 and the methodology reports. The specific information that is proprietary is identified in Enclosure 3 and the reports included in Attachments 5-8 (Attachments 3 and 4 are non-proprietary). The non-proprietary version of the evaluation is provided in Enclosure 4. Revisions to the non-proprietary versions of the methodology reports included in Attachments 5-8 are provided in Attachments 9-12.

Duke requests approval of this LAR by September 30, 2010. Duke will also update the UFSAR to include the revised methodologies and the new analysis results. These revisions will be submitted per 10 CFR 50.71(e). There are no new commitments being made as a result of this proposed change.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the Plant Operations Review Committee and Nuclear Safety Review Board. Additionally, a copy of this LAR is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Inquiries on this proposed amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 873-3364.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 19, 2009.

Sincerely.

Ronald A. Jones
Senior Vice President, Nuclear Operations

Enclosures:

1. Notarized Affidavit of T. C. Geer
2. Notarized Affidavit of Gayle F. Elliott
3. Proprietary Evaluation of Proposed Changes
4. Non-proprietary Evaluation of Proposed Changes

Attachments:

1. Technical Specification and Technical Specifications Bases - Mark Up
2. Technical Specification and Technical Specifications Bases - Reprinted Pages
3. NFS-1001-A - Oconee Nuclear Station Reload Design Methodology (Revision 6a) – Mark Up

4. DPC-NE-1002-A - Oconee Nuclear Station Reload Design Methodology II (Revision 3b) – Mark Up
5. DPC-NE-2003-PA - Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01 (Revision 2a) – Mark Up
6. DPC-NE-2008-PA -Fuel Mechanical Reload Analysis Methodology Using TACO3 (Revision 1a) – Mark Up
7. DPC-NE-3000-PA - Thermal-Hydraulic Transient Analysis Methodology (Revision 4a) – Mark Up
8. DPC-NE-3005-PA - UFSAR Chapter 15 Transient Analysis Methodology (Revision 3b) – Mark Up
9. DPC-NE-2003-A - Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01 (Revision 2a) – Mark Up
10. DPC-NE-2008-A -Fuel Mechanical Reload Analysis Methodology Using TACO3 (Revision 1a) – Mark Up
11. DPC-NE-3000-A - Thermal-Hydraulic Transient Analysis Methodology (Revision 4a) – Mark Up
12. DPC-NE-3005-A - UFSAR Chapter 15 Transient Analysis Methodology (Revision 3b) – Mark Up

bc w/enclosures and attachments:

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ENCLOSURE 1

T. C. GEER AFFIDAVIT

AFFIDAVIT OF T. C. GEER

1. I am Vice President of Duke Energy Carolinas, LLC (Duke) 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.

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's application for withholding which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b) (4) of 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 and has been held in confidence by Duke and its consultants.

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

   (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 Duke proprietary information sought to be withheld in the submittal is that which is marked in the proprietary evaluation of the Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber.

T. C. Geer

(Continued)

This information enables Duke to:

(a) Support license amendment and Technical Specification revision request for its Oconee reactors.

(b) Perform nuclear design calculations on Oconee reactor cores.

(c) Perform transient and accident analysis calculations for Oconee.

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

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

(b) Duke 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.

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

(Continued)

T. C. Geer

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. G

T. C. Geer

Subscribed and sworn to me: October 15, 2009
Date

Freda K. Crump
Notary Public

My Commission Expires: August 17, 2011

FREDA K CRUMP
NOTARY
PUBLIC
LINCOLN COUNTY, NC

SEAL

ENCLOSURE 2

AREVA NP AFFIDAVIT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the attachments to a letter from D.A. Baxter (Duke Energy Carolinas, LLC), to the U.S. Nuclear Regulatory Commission with subject "Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber," Oconee Nuclear Site, Unites 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, dated October 2009 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available,

on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 17th day of September 2009.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

ENCLOSURE 4

NON-PROPRIETARY EVALUATION OF PROPOSED CHANGE

Subject: Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber

1.0 SUMMARY DESCRIPTION

Duke Energy Carolinas, LLC (Duke) requests review and approval from the Nuclear Regulatory Commission (NRC) for a request to amend Renewed Facility Operating Licenses (FOLs) DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station, Units 1, 2, and 3.

The proposed change would revise the Renewed FOLs and Technical Specifications (TS) 2.1.1, Reactor Core Safety Limits and TS 5.6.5.b, Core Operating Limit Report, to allow usage of gadolinia as an integral burnable neutron absorber in the uranium oxide (UO_2) fuel matrix. The associated TS Bases will be revised and is also provided.

Supporting revisions to previously approved Duke methodology reports are provided in the attachments to this submittal. NRC review and approval of the methodology reports is also requested.

2.0 DETAILED DESCRIPTION

2.1 Description of Analysis

Gadolinia has been used as an integral burnable absorber in several nuclear units around the world. It is homogeneously mixed with low enriched uranium (LEU) to form the gadolinia-LEU fuel pellets. The gadolinia-LEU pellets are loaded in the axial middle of the fuel rods with the top and bottom 9" span (approximate) containing non-gadolinia LEU pellets. There are several different gadolinia concentrations that can be used with the weight percent (w/o) limited to between 2 and 8 w/o. The gadolinia bearing fuel rods are arranged symmetrically within the fuel assembly (Oconee will be using AREVA NP Mark-B-HTP fuel for the first batch of gadolinia fuel) with a minimum number of rods being considered of 4 and a maximum number of 24 (approximately 10%). Each fresh assembly that contains gadolinia will contain some combination of gadolinia rods and concentration. Consequently, a full core of gadolinia bearing fuel assemblies will contain no more than approximately 10% gadolinia fuel rods at no more than 8 w/o gadolinia.

As discussed in Section 3.1 below, gadolinia is added to fuel pellets for beginning-of-cycle reactivity hold down and for peaking control. Gadolinia also affects the physical characteristics of the fuel pellets, the most significant being it reduces the thermal conductivity of the fuel pellets and alters the neutron flux profile radially across the pellet such that the power is higher at the pellet periphery relative to conventional uranium fuel. Consequently, the use of gadolinia affects both the technical specifications as described in Section 2.2 below and the methods given in various Duke methodology reports as described in Section 2.3 below.

2.2 Description of TS and Bases Changes

TS 2.1.1.1, Reactor Core Safety Limits (SLs)

Current TS 2.1.1.1 establishes the maximum local fuel pin centerline temperature shall be $\leq 4642 - (5.8 \times 10^{-3}$ (Burnup, MWD/MTU))°F for MODES 1 and 2. This TS will be revised to add the fuel melt limit for fuel containing gadolinia and to revise the existing UO_2 fuel melt limit to include the dependence on the oxygen-to-uranium ratio.

The fuel melt equation currently in TS 2.1.1.1 is from BAW-10162P-A (TACO3) and assumes an oxygen-to-uranium ratio of 2.02. That ratio for Mark-B-HTP is 2.01, which would change the resultant equation. By including the oxygen-to-uranium dependency, the TACO3 equation will be valid regardless of the actual value of that ratio. The gadolinia equation is from BAW-10184P-A (GDTACO), is slightly different, and is therefore also supplied.

TS 5.6.5.b, Core Operating Limits Report (COLR)

TS 5.6.5.b, COLR, provides the previously approved analytical methods used to determine core operating limits. TS 5.6.5.b will be revised to reflect GDTACO in the title of DPC-NE-2008 and to add DPC-NE-1006-P-A to the list of references in anticipation of NRC review and approval of the methodology report. DPC-NE-1006 was submitted to the NRC in a license amendment request (LAR) dated June 10, 2009 (ML091630712) and is required for modeling gadolinia. TS 5.6.5.b was not revised in that LAR. The revision is included in this LAR.

Following NRC approval of this LAR, DPC-NE-2008 will contain GDTACO in the title. The report list in 5.6.5.b is updated to reflect that. Additionally, once the NRC reviews and approves DPC-NE-1006-P, it will be utilized in Oconee core designs and must also be included in the list.

TS Bases 2.1.1, Reactor Core SLs

With the revision of the above, the Reactor Core SLs associated TS Bases 2.1.1 will also be revised to reflect the change associated with adding the fuel melt limit for fuel containing gadolinia and to revise the existing UO_2 fuel melt limit to include the dependence on the oxygen-to-uranium ratio.

UO_2 fuel is modeled with TACO3 and the melt equation contains a dependency on the oxygen-to-uranium ratio while gadolinia fuel is modeled with GDTACO and the melt equation does not contain that dependency.

2.3 Description of Methodology Report Changes

Revisions are made to six NRC reviewed and approved methodology reports that are necessary for modeling gadolinia at Oconee. Other methodology revisions are also included to enhance the existing methodologies, to correct errors, and for editorial clarification. The methodology reports revision numbers will be incremented by one as shown on the cover pages in the attachments. Revisions to the following six methodology reports are described below and included in the attachments:

NFS-1001-A - Oconee Nuclear Station Reload Design Methodology (Revision 6a)

- Figure 1-1 of NFS-1001-A will be revised to add DPC-NE-1006 alongside DPC-NE-1004 and to revise the title of DPC-NE-2008. Additionally, a note will be added to state that DPC-NE-1004 is only included while the core designs transition to DPC-NE-1006.

  Oconee core designs are currently performed and will continue to be performed using CASMO-3/SIMULATE-3 methods documented in DPC-NE-1004 until such a time as they can be transitioned to CASMO-4/SIMULATE-3 methods documented in DPC-NE-1006. DPC-NE-1006 describes the CASMO-4/SIMULATE-3 methods that are required for modeling gadolinia fuel. Therefore, reference to both methodologies is made. The title change to DPC-NE-2008 is made to be consistent with the title change to DPC-NE-2008 as submitted with the LAR.

- Add new Reference 17 for DPC-NE-1006 to reference section. Wherever DPC-NE-1004 is referenced in NFS-1001, a reference to Reference 17 is added. This occurs in Sections 3.1.1, 3.2.3, 3.2.4, 3.2.5, 3.2.8, 5.0, 7.2.2.1, 7.4.1, and 9.0.

  Oconee core designs will be performed using CASMO-3/SIMULATE-3 methods documented in DPC-NE-1004 until such a time as they can be transitioned to CASMO-4/SIMULATE-3 methods documented in DPC-NE-1006. Therefore, reference to both methodologies is made. DPC-NE-1006 is under NRC review and requires approval prior to transitioning core designs containing gadolinia fuel.

- Add gadolinia terms to the uncertainty descriptions in sections 7.2.2.1, 7.2.2.2, and 7.4.1. Specifically, change "lumped burnable poison manufacturing tolerance" to "burnable poison manufacturing tolerance". Additionally, add the gadolinia hot channel factor (HCF) value wherever the UO_2 HCF value appears.

  Since the effect of the lumped burnable poison (LBP) and gadolinia manufacturing tolerance on the pin power uncertainty can be combined as appropriate into one factor representing the burnable poison manufacturing tolerance penalty, the factor is renamed to be more generic. Since the UO_2 HCF is supplied, and since the gadolinia HCF is different, the gadolinia HCF is added.

- Update References 3, 4, 5, 15, and 16 of NFS-1001-A to reflect the anticipated NRC approval and subsequent publication date.

- An editorial change to revise the revision history in Appendix B of NFS-1001 and the cover page is being made.

These changes are included in Attachment 3.

DPC-NE-1002-A - Oconee Nuclear Station Reload Design Methodology II (Revision 3b)

- Since Oconee cores will contain gadolinia as well as LBPs and only one factor is generated that applies to any LBP/gadolinia combination, the factor is renamed the burnable poison (BP) factor. The three statistical combination of uncertainties (SCUF) equations from DPC-NE-1002-A are given below.

$$SCUF_{CFM/CS} = 1.0 + Bias + \sqrt{(U_{A-T})^2 + (U_{R-L})^2 + EHC^2 + LBP^2 + FRB^2 + FAB^2 + FSPIKE^2}$$

$$SCUF_{DNB} = 1.0 + Bias + \sqrt{LBP^2 + FAB^2}$$

$SCUF_{LOCA} = SCUF_{CFM/CS}$ *without the FSPIKE term*

where LBP will be changed to BP in the revised report.

Since the effect of the LBP and gadolinia manufacturing tolerance on the pin power uncertainty can be combined as appropriate into one factor representing a burnable poison manufacturing tolerance penalty, the factor is renamed to be more generic.

- An editorial change to revise the cover page and abstract for Revision 4 and brief description thereof.

- An editorial change to revise Section 1 by adding a brief description of Revision 4 to the end of the section.

- Revise the description following the appearance of the SCUF equation in Section 2.A to reference DPC-NE-1004 for the assembly bias term, the assembly total uncertainty term, and the pin uncertainty term until such a time that Duke transitions to DPC-NE-1006, in which case those 3 terms are obtained from there. The description following the SCUF equation in Section 2.B is revised to reference DPC-NE-1004 for the bias term until transition to DPC-NE-1006 can be completed, after which, DPC-NE-1006 is the valid reference.

  Oconee core designs will be performed using CASMO-3/SIMULATE-3 methods documented in DPC-NE-1004 until they can be transitioned to CASMO-4/SIMULATE-3 methods documented in DPC-NE-1006.

Therefore, reference to both methodologies is made. This allows for an orderly transition in the core designs from CASMO-3/SIMULATE-3 based methods to CASMO-4/SIMULATE-3 based methods.

- Reference 1 is updated to reflect the anticipated NRC approval and subsequent publication date. DPC-NE-1006 is added as Reference 4 for the CASMO-4/SIMULATE-3 generated uncertainties used in the SCUF equations.

These changes are included in Attachment 4.

DPC-NE-2003-A - Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01 (Revision 2a)

- An editorial change is being made to revise the cover page and add Revision 3 history page.

- In Section 5.9, a brief mention that gadolinia affects the radial power distribution but that the reference radial power distribution remains bounding and conservative is added.

  This is a clarification that acknowledges that gadolinia does not adversely affect the reference radial power distribution used in the generation of the maximum allowable peaking (MAP) curves and that the distribution provided in Figures 4-1 through 4-7 remain valid The reference radial power distributions are intentionally flat, i.e., have a very small pin-to-pin power gradient. This maximizes the heat addition to the subchannels and limits the benefits of subchannel cross-flow. Modeling gadolinia would suppress the pin powers thereby reducing the heat addition and provide for additional cross flow benefits. Therefore, the reference distributions remain conservative and bounding.

- The non-SCD axial uncertainty, Fq" in Section 5.11, is updated to [ ] as documented in Tables 5-1 and 5-2 of DPC-NE-1006-P.

  The axial uncertainty calculated with CASMO-4/SIMULATE-3 (DPC-NE-1006-P) is different than the uncertainty calculated with CASMO-3/SIMULATE-3 (DPC-NE-1004-A). Since the CASMO-3/SIMULATE-3 uncertainty value is included, the CASMO-4/SIMULATE-3 value is added. The CASMO-3/SIMULATE-3 axial uncertainty value is used and will continue to be used for LEU fuel until such a time as DPC-NE-1006-P gains NRC approval and cores are designed using that method, after which, the CASMO-4/SIMULATE-3 axial uncertainty will be used.

- The HCF, also discussed in Section 5.11 is updated to add the HTP value with the gadolinia value.

Since the UO_2 hot channel factors are supplied, and since the gadolinia hot channel factor is different, the gadolinia hot channel factor is added to maintain a commensurate level of detail. The paragraph is also rearranged for clarity.

- DPC-NE-1006-P is added to the reference list as Reference 16 which is required for the CASMO-4/SIMULATE-3 generated uncertainties used in the non-SCD analyses. References 1 and 9 are updated to reflect the anticipated NRC approval and subsequent publication date. Reference 6 is updated to the most recent revision number and date.

These changes are included in Attachment 9.

DPC-NE-2008-A -Fuel Mechanical Reload Analysis Methodology Using TACO3 (Revision 1a)

- The title of the report is updated to include GDTACO.

  Appendix A is added to specifically address the use of GDTACO. Therefore, the methodology report will be equally applicable to TACO3 and GDTACO following NRC review and approval.

- Editorial changes are being made to revise the abstract to document the revision description and add GDTACO

- Editorial changes are being made to update the table of contents to include the new GDTACO appendix and delete the heading for Attachment A, which does not exist anymore.

- Editorial changes are being made to modify the introduction section to include discussion of GDTACO and gadolinia fuel.

- An editorial change is being made to add GDTACO to the Reference section.

- Appendix A is added for the GDTACO application description.

  Duke uses the AREVA NP TACO3 code (Reference 11) for steady-state fuel rod thermal and mechanical analysis for AREVA NP fuel designs at all Duke nuclear stations that contain AREVA NP fuel. Duke will use gadolinia as an integral burnable absorber for Oconee with the first use in HTP fuel. AREVA NP has developed the GDTACO code to model gadolinia fuel due to the different material properties. The NRC has reviewed and approved the GDTACO code and implementation methodology. GDTACO is derived from TACO3 and is applied in the same manner as TACO3 is applied to UO_2 fuel. The input file for running GDTACO is essentially a TACO3 input file with additional data to activate the gadolinia thermal models and to provide the additional required input.

Duke will be applying the GDTACO methodology to gadolinia fuel in the same manner that TACO3 is applied to UO_2 fuel as described in DPC-NE-2008.

The specific thermal models in TACO 3 that are revised in GDTACO are the following:

- Radial power profile
- Fuel thermal conductivity
- Fuel melting point
- Fuel densification and swelling

The details of each of these new models are in BAW-10184P-A (Reference 9). The remaining models in TACO 3 are unaffected by the presence of gadolinia, and are also used in GDTACO. All of the cladding models in TACO3 were retained in GDTACO since the composition of the fuel pellet does not affect the behavior of the cladding.

Since the GDTACO code has been approved by the NRC (Reference 10), and since experienced personnel at Duke will be applying GDTACO in the same manner as AREVA NP and within the NRC Safety Evaluation Report (SER) restrictions (see below) placed on the methodology, use of the GDTACO code by Duke for reload analysis of AREVA NP fuel with gadolinia is justified. The GDTACO SER lists the following two restrictions that Duke will comply with.

1. Limited to gadolinia concentrations up to 8 weight percent
2. Cycle-specific analyses must be performed for each reload

These changes are included in Attachment 10.

<u>DPC-NE-3000-A - Thermal-Hydraulic Transient Analysis Methodology (Revision 4a)</u>

- Editorial changes are being made to revise the abstract to add a description of the revision and to update the cover page.

- The radial power distribution description given in Section 2.3.3.4 of DPC-NE-3000 is modified to acknowledge that gadolinia will perturb the radial power distribution (in VIPRE-01) but that the distribution represented in Figure 2.3-4 remains conservative for generic bounding analyses.

  This is a clarification that acknowledges that gadolinia does not adversely affect the generic radial power distribution and that the distribution provided in Figure 2.3-4 remains valid. The reference radial power distribution is intentionally flat, i.e., it has a very small pin-to-pin power gradient. This maximizes the heat addition to the subchannels and limits the benefits of subchannel cross-flow. Modeling gadolinia would suppress the pin powers thereby reducing the heat addition and provide for additional cross flow benefits. Therefore, the reference distribution remains conservative and bounding.

- An editorial change is being made to update the fuel assembly description in Appendix D of DPC-NE-3000 to include discussion of gadolinia.

- The HTP gadolinia HCF is added in Appendix D of DPC-NE-3000.

  Since the UO_2 HCF is supplied, and since the gadolinia HCF is different, the gadolinia HCF is added to maintain a commensurate level of detail.

- Reference E-1 in Appendix E is updated to reflect the anticipated NRC approval and subsequent publication date. Reference E-2 in Appendix E is updated to DPC-NE-1006-P to reflect that the radial power distributions for input to VIPRE could be generated using the CASMO-4/SIMULATE-3 code package. Reference E-3 in Appendix E is updated for the latest SIMULATE-3K reference.

These changes are included in Attachment 11.

<u>DPC-NE-3005-A - UFSAR Chapter 15 Transient Analysis Methodology (Revision 3b)</u>

- Editorial changes are being made to the title page, abstract, and Table of Contents to revise the report revision number to Revision 4, briefly describe the revisions, and to update the report headings, respectively. All references in all sections are updated for the most recent versions or to reflect anticipated NRC approval and subsequent publication for those reports that are revised and submitted in the gadolinia LAR. A revision history page is added after Appendix A to describe in detail the changes made in going from Revision 3b to Revision 4.

- An editorial change is being made to add a revision description to the end of Chapter 1.1. This addition references the gadolinia LAR, which is added as Reference 1-41.

- CASMO-3 is being replaced with CASMO-4 so that gadolinia can be modeled with the same code package as the LEU fuel is modeled. Therefore, the description of CASMO-3 in Section 1.2 is replaced by CASMO-4. Reference 1-21 replaces the CASMO-3 reference with the CASMO-4 reference, Reference 1-22 replaces DPC-NE-1004-A with DPC-NE-1006-P, and Reference 1-23 replaces the SER for DPC-NE-1004-A with the future reference to the SER for DPC-NE-1006-PA.

- An editorial change is being made to revise Chapter 1.3 to add a brief description of gadolinia and its implementation in Oconee cores for completeness.

- The SIMULATE-3K reference is updated for the appropriate code version used in the rod ejection reanalysis. The affected references are 1-26, 2-5, and 14-3.

- Chapter 2.4 is updated to state that CASMO-3 is no longer used but the description is retained since it was used by the ARROTTA demonstration analyses that remain in the report.

  CASMO-3 is being replaced with CASMO-4. However, since the demonstration analyses were performed with CASMO-3, the description of CASMO-3 is retained and revised to state that it is no longer used except in the demonstration analyses presented.

- The SIMULATE-3K discussion in Chapter 2.7 is updated to describe the latest code version.

  The old version of SIMULATE-3K is obsolete and no longer supported by Studsvik. The most recent Version 2 is therefore used and described along with the new models and features that are utilized.

- The SIMULATE-3K code validation discussion in Chapter 2.7.2 is updated to describe the validation of the latest code version.

  Reference to SPANDEX and NEM are deleted (as are the References themselves, 2-19 and 2-20, respectively) since the latest version was not benchmarked to those codes in Reference 2-5. New Reference 2-27 adds an assessment that was performed comparing SIMULATE to reference solutions. SIMULATE-3K was benchmarked to the TRAC code in an earlier version of the SIMULATE-3K Theory and Model manual, but this benchmark is not described in the current SIMULATE-3K Model manual (Reference 2-5), so the earlier version was added as new Reference 2-28. Finally, the steady-state components of the SIMULATE-3K model are described as consistent with CASMO-4/SIMULATE-3P methods documented in methodology report DPC-NE-1006-P (new Reference 2-26).

- The CASMO-4 description as well as new References 2-24, 2-25, and 2-26 are added in new Chapter 2.8.

  CASMO-4 will replace CASMO-3. Since the description of CASMO-3 is retained, a new chapter is added describing CASMO-4. As a result of the new chapter, two new References, 2-24 and 2-25, are added to describe CASMO-4 and one new Reference, 2-26, is added to point to DPC-NE-1006 for the CASMO-4/SIMULATE-3 methodology.

- DPC-NE-1004-A is replaced with DPC-NE-1006-P in all reference lists except Reference 2-15, where it is retained since it is referenced by the CASMO-3 discussion that is also retained. The affected references are References 3-1, 9-6, 10-6, and 15-4.

  The CASMO-3/SIMULATE-3 methodology is being replaced with the CASMO-4/SIMULATE-3 methodology, which is required to model gadolinia. The references are updated appropriately.

- The VIPRE fuel conduction model description in Chapter 10.3.2.5 (Locked Rotor) is revised to acknowledge that gadolinia rods are modeled as if they were UO_2 rods.

  Gadolinia is not modeled explicitly in the VIPRE conduction model because the transient heat flux for the gadolinia rods would be lower than for the corresponding UO_2 rods due to the reduced thermal conductivity of the gadolinia fuel. A reduced heat flux is a MDNBR benefit.

- The initial gap conductance used in the locked rotor transient (Chapter 10.3.2.5) is determined by matching the initial VIPRE fuel temperature to a bounding core average fuel temperature, not by matching the initial VIPRE fuel temperature to TACO-3 predictions. As a result of this change, Reference 10-7 (TACO-3) is also deleted.

  The analysis uses core average fuel temperature as the target for determining the initial gap conductance. This is an error correction.

- The VIPRE fuel conduction model description in Chapter 14.2.4.1 (Rod Ejection peak pellet enthalpy) is revised to state that the gadolinia fuel properties will be used for gadolinia fuel.

  For the same rod power, it is expected that gadolinia rods will have a slightly higher peak pellet enthalpy than a corresponding UO_2 rod due to the gadolinia fuel thermal properties. Therefore, gadolinia thermal properties are modeled explicitly for the gadolinia rods.

- The Conservative Factors section of Section 14.2.4.1 is revised to add the HTP gadolinia HCF.

  Since the UO_2 HCF is supplied, and since the gadolinia HCF is different, the gadolinia HCF is added to maintain a commensurate level of detail.

- The VIPRE fuel conduction model description in Chapter 14.2.4.2 (Rod Ejection DNBR) is revised to state that the gadolinia rods are modeled as if they were UO_2 rods.

Gadolinia is not modeled explicitly in the VIPRE conduction model because the transient heat flux for the gadolinia rods would be lower than for the corresponding UO_2 rods due to the reduced thermal conductivity of the gadolinia fuel. A reduced heat flux is a MDNBR benefit. Furthermore, gadolinia fuel temperatures are higher than UO_2 fuel temperatures for the same rod power, which translates into a lower initial gap conductance following the method prescribed in chapter 14.2.4.2, which also results in a lower transient heat flux. Consequently, using the UO_2 (i.e., TACO3) fuel temperatures as a target for establishing the gap conductance is conservative.

- The flux/flow and high Reactor Coolant System (RCS) pressure trips are added to the list of reactor trip signals in the SIMULATE-3K analysis (Chapter 14.3.1).

  Credit is being taken for additional trip functions not specified in the report.

- The excore signal discussion in the Reactor Trip and Single Failure section of Section 14.3.1 is updated to say that it is "synthesized from a radially weighted combination of power densities from every assembly in the core".

  This change describes the new method for modeling the excore detector signal. The previous method only calculated an excore signal based on a few assemblies close to the detector. The newer method is more accurate since it accounts for the contribution from all assemblies in the excore power prediction.

- The following section in Chapter 14.3.1 is revised as shown.

  "<u>Scram Curve and Worth</u>
  The reactor scram assumes that the ejected rod, part length axial power shaping rods, and the remaining rod with the highest worth do not fall into the core. The remaining control rods (reduced by an appropriate uncertainty) drop into the core at a speed that satisfies the maximum rod drop time in the TS."

  The rod ejection reanalysis will not limit the scram worth to maintain the minimum shutdown margin nor the rate at which the scrammed rods fall into the core. The minimum trippable rod worth ensures that a conservatively low amount of worth is tripped into the core thereby negating the need to ensure that the minimum shutdown margin is ensured all throughout the scram. The TS rod drop times are assumed in the analysis thereby precluding the need to limit the rate during the entire time the rods are falling into the core.

- The Conservative Factors section of Section 15.3.1.2.4 is revised to add the HTP gadolinia HCF.

  Since the UO_2 HCF is supplied, and since the gadolinia HCF is different, the gadolinia HCF is added to maintain a commensurate level of detail.

These changes are included in Attachment 12.

3.0 TECHNICAL EVALUATION

3.1 UFSAR Chapter 15 non-LOCA Evaluation

An evaluation has been performed on the impact of the use of gadolinia as an integral absorber on the non-LOCA licensing basis transients and accidents as described in Chapter 15 of the Oconee Updated Final Safety Analysis Report. Gadolinia is added to fuel pellets for beginning of cycle reactivity hold down and for power peaking control. DPC-NE-1006-P (Reference 1) demonstrates the ability to predict power distributions and physics parameters of cores containing gadolinia fuel. The presence of gadolinia also affects the physical characteristics of the fuel pellets. The most significant impacts are as follows:

- Thermal conductivity – the addition of gadolinia to UO_2 lowers the thermal conductivity of the fuel pellets. This raises the initial stored energy of the fuel rods which contain gadolinia. This also retards the heat transfer from the pellet to the cladding and coolant during a transient.
- Pellet radial power profile – the presence of neutron absorbing gadolinia atoms in the fuel depresses the neutron flux in the center of the pellet, resulting in a pellet radial power profile that is more outside-peaked than conventional uranium fuel.

As a result of the reduced thermal conductivity, the fuel temperature of the gadolinia rod will be higher than a rod containing LEU assuming both rods are at the same power. While an increase in core average fuel temperature (CAFT) is not anticipated due to the reduced power of the gadolinia bearing fuel rods, it is conservatively assumed that the CAFT could increase slightly. Since the physics code cannot account for an increase in fuel temperature due to the presence of gadolinia, a separate CAFT adder is calculated to account for this possibility. The CAFT adder is then applied to the cycle specific CAFT predicted by the physics code and compared to the CAFTs assumed in the safety analyses to ensure the safety analyses remain bounding.

At beginning of life, fuel rods with gadolinia are clearly non-limiting from a thermal perspective because the thermal neutron flux and power are suppressed in the gadolinia rods. As the neutron-absorbing gadolinium isotopes are depleted, the power in the rod will approach or exceed the power of the rods which do not contain gadolinia. Neutron absorption in gadolinium isotopes produces other gadolinium isotopes, so the degradation of thermal conductivity due to gadolinia persists throughout the life of the fuel. For the purpose of this evaluation it was assumed that the rods containing gadolinia could be the highest power rod in a given assembly except during the initial exposure period.

UFSAR Chapter 15 safety analyses typically are based on system and core thermal-hydraulic analyses which assume bounding values for key physics parameters. As discussed in DPC-NE-3005-PA, these analyses are performed generically and remain applicable as long as the key physics parameters for a specific reload core are bounded by the generic analysis. For cores containing gadolinia fuel the presence of gadolinia in a limited number of pins does not significantly perturb key physics parameters, so the existing analyses will remain valid.

However, for some UFSAR Chapter 15 events the gadolinia changes to pellet thermal conductivity and pellet radial power profile have the potential to affect transient and accident analysis results. The impact on Chapter 15 events with the potential to challenge the DNBR criterion is as follows: Higher initial stored energy provides more heat to be transferred to the coolant, which would tend to reduce margin to DNB. An outside-peaked pellet radial power profile means that the initial fuel pellet energy is more available for transfer to the coolant, which would also tend to reduce margin to DNB. The degradation of thermal conductivity retards the transfer of pellet energy to the coolant and thus provides a DNB benefit. Sensitivity studies of design basis transients were performed to evaluate the overall impact of these effects. The studies indicate that the degradation of thermal conductivity, a DNB benefit, is the predominant effect. Therefore, addition of gadolinia to fuel pellets has no significant adverse impact on transient DNB.

Preventing centerline fuel melt (CFM) is another thermal acceptance criterion for design basis events. The impact on Chapter 15 events with the potential to challenge the CFM criterion is as follows. By lowering thermal conductivity in the fuel pellets, gadolinia will increase initial fuel temperatures and thereby lower the margin to CFM. As discussed in the revisions to DPC-NE-2008 (Attachment 6 of this submittal), linear heat rate limits to preclude CFM will be calculated specifically for gadolinia rods using the AREVA NP GDTACO code (Reference 9). These gadolinia-specific linear heat rate limits are then compared to predicted core powers in gadolinia rods to ensure margin to CFM during design basis events.

One UFSAR non-LOCA accident that involves detailed fuel rod modeling is the control rod ejection accident. The rod ejection licensing basis for Oconee involves three acceptance criteria: 1) fuel pellet energy deposition (i.e., cal/gm limits), 2) offsite dose (based on the number of rods that experience DNB), and 3) peak RCS pressure. The second and third criteria are not adversely affected by the presence of gadolinia. With respect to the first criterion, fuel pellet energy deposition, the effects of gadolinia described above will require separate consideration for gadolinia fuel. As noted above, the power will be suppressed in the gadolinia fuel rods until the gadolinium has been depleted. If the power response is sufficiently lower in the gadolinia fuel rods, relative to the UO_2 rods, then the differences in the gadolinia rods is of no consequence. However, if the power response is comparable to the UO_2 rods, then modeling of the fuel rod response with the VIPRE-01 code using appropriate inputs for gadolinia fuel pellets will be performed.

In summary, an evaluation has been performed of the potential impact of gadolinia fuel on the licensing basis non-LOCA transient and accident analyses. The presence of gadolinia will suppress the initial local pin powers until the gadolinium has been depleted. It is also expected that the gadolinia rods will be designed to operate at lower power levels than the UO_2 rods, although this design approach is not credited in this evaluation. The limited number of gadolinia rods will not have a significant impact on the key physics parameters that are assumed for transient and accident analyses, and that are checked for each reload design. Reanalysis will be required if the key parameters are exceeded. The evaluation has concluded that there is no adverse effect on gadolinia on those events involving the DNBR criterion. The CFM criterion will be checked explicitly for the gadolinia rods based on limits calculated with the GDTACO code. The acceptance criteria that are associated with the rod ejection accident will be shown to be met for gadolinia fuel.

3.2 UFSAR Chapter 15 LOCA Evaluation

4.0 REGULATORY SAFETY ANALYSIS

4.1 Significant Hazards Consideration

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request does not involve a significant hazards consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed revisions to the technical specifications and to Duke's NRC approved methodology reports support the use of the gadolinia in the Oconee fuel design. The methodology reports will be approved by the NRC prior to plant operation with the new fuel. The proposed safety limit ensures that fuel integrity will be maintained during normal operations and anticipated operational transients. The Core Operating Limits Report (COLR) will be developed in accordance with the approved methodology reports. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no negative impact on the source term or pathways which have been assumed in accidents previously analyzed. 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.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The new and revised fuel melt equations are not an accident / event initiator. No new initiating events or transients result from the use of the revised safety limit.

3) Involve a significant reduction in a margin of safety.

   The proposed safety limit value has been reviewed and approved by the NRC as part of the approval of the AREVA NP TACO3 and GDTACO topical reports to, in part, specifically calculate the temperature at which the fuel will melt. Duke uses TACO3 and will use GDTACO in accordance with the restrictions stipulated in the safety evaluation of both AREVA NP topical reports and those set forth in Duke's NRC approved methodology reports to ensure that the limit is not exceeded for those events in which fuel melt is not allowed. The other reactor core safety limits will continue to be met by analyzing the reload using NRC approved methods and incorporation of resultant operating limits into the COLR.

4.2 Applicable Regulatory Requirements/Criteria

The proposed change to the Technical Specifications is based on the forthcoming approval of this License Amendment Request, which includes approval of the methodology changes provided in the attachments.

The fuel melt limit equations were approved as part of the AREVA NP TACO3 and GDTACO topical report approvals.

4.3 Precedent

June 7, 1993	Exelon Generation Company, LLC - TMI 1 – License Amendment to revise the plant TS to reflect the inclusion of gadolinia-urania in the fuel rod design description, the borated water storage tank boron concentration limits, and to clarify the bases.
September 10, 1993	Exelon Generation Company, LLC - TMI 1 – License Amendment #187, Revises the plant TS to reflect the inclusion of gadolinia-urania in the fuel rod design description, the borated water storage tank boron concentration limits, and to clarify the bases.

5.0 ENVIRONMENTAL CONSIDERATION

Duke Energy Carolinas, LLC, has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.1, revising the methodology reports in support of Gadolinia does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

Revising the methodology reports in support of Gadolinia will not impact effluents released offsite. Therefore, there will be no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

Revising the methodology reports in support of Gadolinia will not have an adverse impact on occupational radiation exposure. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this action.

6.0 REFERENCES

1. Letter from D. A. Baxter (Duke) to NRC dated June 10, 2009, Docket Numbers 50-269, 50-270, and 50-287, "Proposed License Amendment Request for Methodology Report DPC-NE-1006-P, 'Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3' ", License Amendment Request No. 2009-02.
2. NFS-1001-A, Rev. 6a, Oconee Nuclear Station Reload Design Methodology, June 2009.
3. DPC-NE-1002-A, Rev. 3b, Oconee Nuclear Station Reload Design Methodology II, June 2009.
4. DPC-NE-1004-A, Rev. 1a, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, January 2009.
5. DPC-NE-2003-A, Rev. 2a, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, December 2008.
6. DPC-NE-2008-A, Rev. 1a, Fuel Mechanical Reload Analysis Methodology Using TACO3, December 2008.
7. DPC-NE-3000-A, Rev. 4a, Thermal-Hydraulic Transient Analysis Methodology, July 2009.
8. DPC-NE-3005-A, Rev. 3b, UFSAR Chapter 15 Transient Analysis Methodology, July 2009.
9. GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code, BAW-10184-A, AREVA NP, February 1995.
10. Letter, A. C. Thadani (NRC) to J. H. Taylor (AREVA NP) dated June 24, 1993 (GDTACO Safety Evaluation).
11. TACO3 Fuel Pin Thermal Analysis Computer Code, BAW-10162-A, AREVA NP, November 1989.

ATTACHMENT 1
Technical Specification and Technical Specification Bases – Mark Up

TS 2.1.1.1
TS 5.6.5.b
TSB 2.1.1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature shall be $\leq$ 4656~~42~~ - (5.8 x 10^{-3} x (Burnup, MWD/MTU)) – 709.04|chi| - 786.62$(chi)^2$ + 1087.07$(chi)^3$ °F where chi is the quantity oxygen-to-uranium ratio minus 2.0. For gadolinia fuel, the local fuel pin centerline temperature shall be $\leq$ 4656 – (6.5 x 10^{-3} x (Burnup, MWD/MTU)) °F. Operation within ~~this~~ these limits is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation, 1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to $\leq$ 2750 psig within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

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

(1) DPC-NE-1002-A, Reload Design Methodology II;

(2) NFS-1001-A, Reload Design Methodology;

(3) DPC-NE-2003-P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01;

(4) DPC-NE-1004-A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P;

(5) DPC-NE-2008-P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3 and GDTACO;

(6) BAW-10192-P-A, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000-P-A, Thermal Hydraulic Transient Analysis Methodology;

(8) DPC-NE-2005-P-A, Thermal Hydraulic Statistical Core Design Methodology;

(9) DPC-NE-3005-~~P-A~~, UFSAR Chapter 15 Transient Analysis Methodology; ~~and~~

(10) BAW-10227-P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel;

(11) BAW-10164P-A, RELAP 5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis; and

(12) DPC-NE-1006-P-A, Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3.

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

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 System (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) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2), the BWU (Ref. 4), and the BHTP (Ref. 5) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. The BHTP correlation applies to the MARK-B-HTP fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC), 1.19 (BWU) and 1.132 (BHTP).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The dependency of the fuel melt temperature on the as-built oxygen-to-uranium ratio for UO2 fuel pins is provided by the fuel vendor. For gadolinia fuel pins, there is no dependence on the oxygen-to-uranium ratio.

Operation above the boundary of the nucleate boiling regime could result in

ATTACHMENT 2

Technical Specification and Technical Specifications Bases - Reprinted Pages

Remove	Insert
TS 2.0-1	TS 2.0-1
TS 5.0-25	TS 5.0-25
TS 5.0-26	TS 5.0-26
TS 5.0-27	TS 5.0-27
TSB 2.1.1-1	TSB 2.1.1-1
TSB 2.1.1-2	TSB 2.1.1-2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature shall be $\leq 4656 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})) - 709.04|chi| - 786.62(chi)^2 + 1087.07(chi)^3$ °F where chi is the quantity oxygen-to-uranium ratio minus 2.0. For gadolinia fuel, the local fuel pin centerline temperature shall be $\leq 4656 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU}))$ °F. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation, 1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

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

(1) DPC-NE-1002-A, Reload Design Methodology II;

(2) NFS-1001-A, Reload Design Methodology;

(3) DPC-NE-2003-P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01;

(4) DPC-NE-1004-A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P;

(5) DPC-NE-2008-P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3 and GDTACO;

(6) BAW-10192-P-A, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000-P-A, Thermal Hydraulic Transient Analysis Methodology;

(8) DPC-NE-2005-P-A, Thermal Hydraulic Statistical Core Design Methodology;

(9) DPC-NE-3005-P-A, UFSAR Chapter 15 Transient Analysis Methodology;

(10) BAW-10227-P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel;

(11) BAW-10164P-A, RELAP 5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis; and

(12) DPC-NE-1006-P-A, Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3.

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

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 System (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) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.10, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active 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 active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

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

h. The effective plugging percentage for all plugging in each SG.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2), the BWU (Ref. 4), and the BHTP (Ref. 5) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. The BHTP correlation applies to the MARK-B-HTP fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC), 1.19 (BWU) and 1.132 (BHTP).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The dependency of the fuel melt temperature on the as-built oxygen-to-uranium ratio for UO_2 fuel pins is provided by the fuel vendor. For gadolinia fuel pins, there is no dependence on the oxygen-to-uranium ratio.

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam relief valves (MSRVs) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and anticipated transients. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints (Ref. 3), in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for RCS temperature, flow and pressure, and THERMAL POWER level that would result in a DNB ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

f. Flux/Flow Imbalance trip;

ATTACHMENT 3

NFS-1001-A – Oconee Nuclear Station Reload Design Methodology (Revision 6a) – Mark-Up

Oconee Nuclear Station
Reload Design Methodology

NFS-1001~~-A~~
Revision ~~6a~~7

~~June 2009~~ October 2009

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

Figure 1-1

Relationship of Reload Methodology Reports

DPC-NE-1004-A *
Nuclear Design Methodology Using
CASMO-3/SIMULATE-3P
or

DPC-NE-1006-PA
Oconee Nuclear Design Methodology
Using CASMO-4/SIMULATE-3

NFS-1001-A
Reload Design Methodology

DPC-NE-1002-A
Reload Design Methodology II

1.0 Introduction
2.0 Fuel Design
3.0 Fuel Cycle Design
Generation of preliminary and final fuel cycle design. Analysis determines feed batch size, enrichment, and burnable poison requirements. FFCD provides power distribution for Fuel Mechanical Design and Thermal-Hydraulic Design. All safety limits confirmed to be less limiting than those analyzed in Accident Analysis Review.
4.0 Fuel Mechanical Performance
5.0 Maneuvering Analysis
Confirmation of LHR to prevent CFM, steady state DNBR, initial condition DNBR from limiting design transient, initial condition linear heat rate from limiting LOCA, and shutdown margin.
6.0 Thermal-Hydraulic Design
7.0 Technical Specification / COLR
Limits confirmed in Maneuvering Analysis – Transmitted via COLR document.
8.0 Accident Analysis Review
9.0 Development of Core Physics Parameters
Generation of the Physics Test Manual (PTM) which provides startup physics testing data and cycle dependent nuclear data for the FFCD.

* **DPC-NE-1004 will eventually be replaced by DPC-NE-1006 as the core designs transition to the DPC-NE-1006 methodology**

BAW-10186-PA
Extended Burnup Topical for Mark-B Fuel

4.0 Fuel Mechanical Performance
DPC-NE-2008-PA
Fuel Mechanical Reload Analysis Methodology
Using **TACO 3 and GDTACO**

Verification of fuel pin pressure, clad strain, creep collapse, LOCA initialization, and corrosion using information (power distribution, fuel burnup, and flux data) from NFS-1001 FFCD.

Linear heat rate to melt limits generated and confirmed via NFS-1001 Maneuvering Analysis.

LOCA limits provided by vendor via cycle specific LOCA Check Document.

6.0 Thermal-Hydraulic Design
DPC-NE-2003-PA
Core Thermal-Hydraulic Methodology Using
VIPRE-01

DPC-NE-2005-PA
Thermal-Hydraulic Statistical Core Design Methodology

VIPRE-01 DNB MATP limits generated and confirmed via NFS-1001 Maneuvering Analysis.

8.0 Accident Analysis Review
DPC-NE-3000-PA
Thermal-Hydraulic Transient Analysis Methodology

DPC-NE-3005-PA
UFSAR Chapter 15 Non-LOCA Transient Analysis Methodology

Analysis performed using **VIPRE-01**, **RETRAN-02**, and **SIMULATE-3K**. Inputs (MATP, fuel temperature, reactivity parameters, etc.) confirmed to be bounding by analysis performed in **NFS-1001**.

3.0 Fuel Cycle Design

3.1 Preliminary Fuel Cycle Design

The purpose of the preliminary fuel cycle design (PFCD) is to determine the number and enrichment(s) of the fresh and possibly burned assemblies to be inserted during the next refueling. A preliminary fuel shuffling scheme is developed and check calculations on certain key parameters are performed.

The input required for the PFCD consists of general ground rules and design bases developed from cycle energy, contract, and operating requirements. The output of the PFCD is the number and enrichment(s) of the feed assemblies.

3.1.1 Overview of Nuclear Calculation System

The nuclear calculation system enables the nuclear designer to numerically model and simulate the nuclear reactor core. The current system in use for Oconee is described in Reference 2. **The system that will replace Reference 2 and be used following NRC approval is documented in Reference 17.**

3.1.2 Calculations and Results of PFCD

Once the calculation models are prepared for the cycle of interest, the nuclear designer chooses one or more feed enrichments, number of assemblies, and preliminary loading pattern for the reload core. Calculations are performed to verify cycle lifetime and power peaking. The process is iterated until the number and enrichment(s) of feed assemblies as well as a preliminary shuffle scheme has been determined which yield the desired cycle lifetime and a reasonable power distribution.

The preliminary number and enrichment(s) of the feed assemblies must typically be determined eighteen months prior to reactor shutdown for refueling to assure that an adequate quantity of separative work is available. Changes to these preliminary estimates are normally possible up to twelve months prior to reactor shutdown. It is necessary that the results of the PFCD be complete in time to support the fuel order.

3.2 Final Fuel Cycle Design

Having determined the preliminary number and enrichment(s) of the fuel assemblies during the PFCD, the final fuel cycle design (FFCD) concentrates on optimizing the placement of fresh and burned assemblies, control rod groupings, and burnable poison assemblies (if any) to result in an acceptable fuel cycle design. If not already performed during the PFCD, cladding corrosion calculations are performed to ensure licensing limits are met (References 7 and 8). The fuel cycle design is finalized based upon design criteria intended to ensure that the results of the subsequent calculations are acceptable. If unacceptable results are obtained, the fuel cycle design may be revised to obtain a design that produces acceptable results. When appropriate, the calculations performed to support the PFCD are incorporated into the FFCD.

3.2.3 Power Distribution Calculations

For Oconee, emphasis in the FFCD is on radial power distributions both on an assembly basis and on a local rod basis. Power distributions are calculated using the calculation methods described in Reference 2 **or Reference 17**. Radial pin peaking limits that will result in acceptable DNB and CFM margins are obtained from the accident analyses, thermal and thermal-hydraulic models. These margins are calculated and confirmed during the maneuvering analysis described in Section 5.0 and the accident analysis review described in Section 8.2.

3.2.4 Fuel Burnup Calculations

Current design criteria include limitations on fuel burnup. These limitations may be required as a result of calculations of internal fuel rod pressure, fuel rod growth, cladding corrosion, or licensing limitations. Fuel burnup calculations are performed using the calculation methods described in Reference 2 **or Reference 17**. Both assembly average and local fuel rod burnups may be calculated using these methods.

3.2.5 Reactivity Coefficients and Deficits

Reactivity coefficients define the reactivity insertion for small changes in reactor parameters such as moderator temperature, fuel temperature, and power level. These parameters are calculated using the methodology described in Reference 2 **or Reference 17**. These parameters are input to the safety analyses and used in modeling the reactor response during accidents and transients. Whereas reactivity coefficients represent reactivity effects over small changes in reactor parameters, reactivity deficits usually apply to reactivity inserted from larger changes typical of hot full power (HFP) to hot zero power (HZP). An example of a reactivity deficit is the power deficit from HFP to HZP. A different way of looking at the terms is that the coefficient when integrated over a given range yields the deficit, or the coefficient is the partial derivative of reactivity with respect to one specific parameter.

Typically, a nominal case is established at some reference condition. Then one parameter of interest is varied up and/or down by a fixed amount in another calculation and the resulting change in core reactivity divided by the parameter change yields the reactivity coefficient.

3.2.5.1 Doppler Coefficient

The Doppler coefficient or fuel temperature coefficient (FTC) is the change in core reactivity produced by a small change in fuel temperature. The major component of the Doppler coefficient arises from the behavior of the uranium-238 and plutonium-240 resonance absorption cross sections. As the fuel temperature increases, these resonances broaden and increase the chance that a neutron will be absorbed and thus decrease the core reactivity.

3.2.6 Boron Related Parameters

Critical boron concentrations are calculated at a variety of conditions as described in Reference 3.

3.2.7 Xenon Worth

The HFP equilibrium xenon worth may be calculated at BOC (4 EFPD) and at EOC. These values are compared to previous cycle values when a reload report is generated.

Calculations are performed for HFP equilibrium xenon conditions and for no xenon conditions. The difference in reactivities between the equilibrium and no xenon cases is the xenon worth.

3.2.8 Kinetics Parameters

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

The computer codes used to calculate these parameters are described in References 2, **3** and ~~3~~**17**. This information is needed for validation of the accident analyses and startup physics testing. The effective delayed neutron fraction (β-effective) for the new reload cycle is compared to that of the previous cycle when a reload report is generated.

5.0 MANEUVERING ANALYSIS

The purpose of a maneuvering analysis is to generate three-dimensional power distributions and imbalances for a variety of rod positions, xenon distributions, and power levels. The maneuvering analysis can be divided into four discrete phases. The first phase is the fuel cycle depletion performed to establish a nominal fuel depletion history. The second phase is the performance of various power maneuvers that conservatively characterize the effect of maldistributed xenon on the power distributions. The third phase is to perform control rod and axial power shaping rod (APSR) scans at the most severe times during the power transients. Each of these phases involves the running of multiple cases and generating three-dimensional power distributions, rod positions, and imbalances for each case. The methodology described in Reference 2 **or Reference 17** is used to generate this information. Finally, this data is processed by computer programs which calculate CFM, clad strain, DNB, and LOCA margins to be used to set COLR (see Section 7.0) limits on rod position, axial offset versus power level, and reactor protective system setpoints.

5.1 Fuel Cycle Depletion

If appropriate restart files from the cycle depletion performed during the FFCD are not available, then the fuel cycle depletion is performed as the first step of the maneuvering analysis. Typical depletion steps are 0, 4, 12, 25, 50, 100, 150 ... EFPD. The xenon, power, and exposure data from these cases are saved for use in subsequent analyses.

5.2 Power Maneuvers

Power maneuvers are performed to generate axially skewed xenon distributions for input to the rod scan cases. The power maneuvers are performed near BOC, near EOC, and at least one intermediate burnup.

The first power maneuver is initiated by manipulating the control rods to produce a positive imbalance (with associated equilibrium iodine and xenon distributions) at full power. Control rod group 7 and the APSRs are then inserted to approximately the core midplane and core power reduced accordingly. This control rod insertion generates a large negative imbalance, and in conjunction with the power reduction causes the xenon in the bottom of the core to be depleted while the initial iodine in the top of the core increases the xenon concentration. The power level and rod positions are held constant, and the xenon concentration is allowed to peak over the next several hours. At a timestep near the peak xenon concentration, the xenon distribution is saved for input to the rod scan cases.

Section 7.2.2.1 (continued from previous page in NFS-1001)
The allowable total peaking factor (MAPF) is established by the relation:

$$MAPF = \frac{CFMLHR}{LHR \times FOP}$$

where: LHR is the average full power linear heat rate in the core, and
FOP is the power level expressed as a fraction of rated power.

The maneuvering analysis (Section 5.0) establishes the maximum calculated total peaking factors for various core conditions (power levels, xenon conditions, control rod positions and burnups). These calculated maximum total peaking factors are increased by several conservative factors to obtain the worst case expected total peaking factor corresponding to each condition. The individual conservative factors are as follows:

1) Nuclear uncertainty factor as specified in Reference 2 **or Reference 17**.
2) Spacer grid effect factor of 1.026, which is only applicable when utilizing assemblies with inconel intermediate spacer grids.
3) Engineering hot channel factor of 1.014 **for UO_2 fuel and 1.0145 for gadolinia-bearing fuel**
4) Densification power spike factor which varies with axial location of the peak in the core (Reference 15).
5) Fuel assembly bow factor.
6) Fuel rod bow factor.
7) ~~Lumped b~~**B**urnable poison manufacturing tolerance factor.

The nuclear uncertainty factor accounts for the uncertainty in the calculated peak due to the limitations of the analytical models. The spacer grid effect factor accounts for the flux distortion caused by inconel spacer grids (no spacer grid effect factor is required for zircaloy spacer grids). The engineering hot channel factor accounts for the manufacturing tolerances of critical fuel rod design parameters (pellet enrichment, pellet density, pellet diameter, etc.). The densification power spike factor accounts for the local flux enhancement resulting from gaps in the fuel column induced by fuel densification. The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor that is dependent on the location of the pin within the assembly. A burnup dependent peaking penalty consistent with topical reports BAW-10147-PA (Reference 10) and BAW-10186-PA (Reference 7) is applied to account for the potential power peaking enhancement due to fuel rod bow. The ~~lumped~~ burnable poison manufacturing tolerance factor accounts for the effect of the variance in the as-built enrichment of the lumped burnable poison (LBP) pellets in LBP rods **or the gadolinia in gadolinia-bearing fuel rods**. The statistical combination of these factors is described in Reference 15.

The worst case expected maximum total peaking factors calculated in this manner for different power levels are compared to the respective allowable total peaking factors, and the CFM margin for each condition can be determined. The margin at a particular power level is given by:

$$Margin\ (\%) = \frac{(allowable\ total\ peak - worst\ case\ expected\ maximum\ total\ peak)}{allowable\ total\ peak} \times 100$$

Core conditions which correspond to non-negative margins are acceptable conditions, and core conditions which correspond to negative margins cannot be permitted. In order to preclude core conditions with negative margins, limits should be established on acceptable values of power peaking conditions for each power level, and corresponding reactor trip setpoints should be established so as to trip the reactor when conditions approach unacceptable values. Since power peaking cannot directly be measured by the RPS, power peaks are first correlated with the RPS-measurable axial offset for each power level. The outputs of the maneuvering calculations include the maximum total peaking factor in the core, its location and the corresponding core axial offset. In order to determine the axial offset limits that correspond to an acceptable margin for a particular power level, the margin for each calculated maximum total peak for that power level is plotted against the corresponding axial offset. These plots define a relationship between core offset and margin. The value of offset at the zero margin intercept defines the offset limit for that particular set of reactor conditions. Figure 7-2 provides an example of the analysis for the 100% full power (FP) case.

In practice, detailed calculations typically are performed for the 100% FP case. Limits for other power levels may be determined by conservatively extrapolating the 100% FP limits to other power levels by using the power feedback effect on peaking factors and by validating these limits by comparison with results of a limited number of maneuvering calculations at these power levels. Offset limits are typically established for power levels of 110% FP and 100% FP.

7.2.2.2 Calculation of Power-Power Imbalance Limits for DNBR Criterion

The power-power imbalance limits based on the DNBR criterion are determined by a synthesis of the results of the thermal-hydraulic analysis and the results of the maneuvering analysis.

The thermal-hydraulic analysis establishes the maximum allowable total peaking (MATP) factors as a function of core elevation for various axial flux shapes to prevent violation of the DNBR criterion. The maneuvering analysis generates the power distribution in the core (including the maximum total peaking factor and the associated axial peaking factor for each fuel assembly, typically in a quarter-core representation, and the core axial offset) for various design conditions and for various times in the cycle. For each power distribution, the calculated maximum total peaking factors of each of the assemblies is increased by the radial nuclear uncertainty factor, by a ~~lumped~~ burnable poison manufacturing tolerance factor, and by a factor to account for the effect of fuel assembly bow, and the resulting adjusted peak is compared to the allowable peaking factor for that axial peaking factor and axial peak location. The statistical combination of these factors is described in Reference 15. Application of the radial nuclear uncertainty is not necessary when the allowable peaking factor is determined using the statistical core design methodology described in Reference 6 (which accounts for the radial nuclear uncertainty in developing the allowable peaking factor). The DNBR margin is then obtained as:

$$DNBR\ Margin\ (\%) = \frac{(allowable\ total\ peak - adjusted\ maximum\ total\ peak)}{allowable\ total\ peak} \times 100$$

Section 7.4.1 (continued from previous page in NFS-1001)
The power peaking factor in the core changes with fuel burnup, axial imbalance, full length control rod position, and part length control rod (APSR) position. In addition, the peaking factor is influenced by the existence of any quadrant power tilt and non-equilibrium xenon conditions. Therefore, allowable ranges of these core operation parameters would have to be established in order for the maximum operating peaking factors at the designated axial locations to be within the allowable values. Although the fuel densification phenomenon has the potential for enhancing power peaks, no explicit allowance is required for power spikes associated with this phenomenon in the LOCA power distribution limits on the basis that the densification power spikes do not enhance the local heat flux.

The effect of a positive quadrant power tilt on the peaking factors is quantified either on a cycle-specific basis as a function of assembly location and burnup statepoints (using the methods described in Reference 2 **or Reference 17**), or by application of a conservative generic factor. The quadrant tilt power peaking factors are calculated as the percentage change in peak per percent change in quadrant tilt for each symmetric assembly. Specifically, a series of cases are executed with each unique control rod location modeled as a dropped rod. The associated increase in peaking and tilt in the opposite quadrant is tabulated for each symmetric assembly location. The largest ratio of percent change in peak per percent change in tilt is saved for each symmetric assembly location. These cycle-specific 'tilt factors' typically range from 0.8% to 1.4% increase in peaking factor (depending on the assembly location) per percent positive quadrant tilt. The conservative tilt factor may be as high as 1.5% increase in peaking factor per percent positive quadrant tilt. Technical Specifications permit reactor operation with a positive quadrant tilt as specified in the COLR. A tilt limit of 5.0% would typically amount to a 4.0% to 7.0% increase in peaking factor when using the cycle-specific tilt factors, or a 7.5% increase in peaking factor when using the conservative generic factor. Therefore, the allowable peaking factor would have to be reduced by 4.0% to 7.0%, or by 7.5%, whichever is applicable, to account for the permitted quadrant tilt condition.

The effect of non-equilibrium xenon conditions on peaking factors is quantified by the analysis of the power peaking factors occurring during various power maneuvers. Power redistribution caused by transient xenon in the power maneuver leads to peaking and offsets being explicitly accounted for in the setting of LOCA limits.

The remaining core parameters which influence the maximum operating power peaks are the full-length control rod position, part length control rod (APSR) position, axial imbalance, and core burnup. The permissible values of these quantities are to be determined such that the resulting power peaks, after accounting for any uncertainties, would be within the maximum allowable power peaks. The maneuvering analysis establishes the relationship of operating peaking factors at various axial locations with the core imbalance and control rod positions. The maneuvering analysis calculations include part length control rod scans inducing a range of values of core axial offset for different full length control rod positions. The calculations are performed for various power levels and for the full range of core burnups. The calculations yield the values of the maximum peaking factor at the different axial planes corresponding to various full-length control rod positions, various axial offsets, and for different part length rod positions, and these calculations also yield the variations of the maximum peaking factor with axial offset.

The calculated maximum peaks at each axial plane are increased by the following factors to obtain the worst case operating peaking factors. In addition, a power level uncertainty factor, as specified in References 3 and 15, is applied as a bias to the calculated maximum peaks.

1) Nuclear uncertainty factor as specified in Reference 2 **or Reference 17**.
2) Spacer grid effect factor of 1.026, which is only applicable when utilizing assemblies with Inconel intermediate spacer grids.
3) Engineering hot channel factor of 1.014 **for UO_2 fuel and 1.0145 for gadolinia-bearing fuel**
4) Fuel assembly bow factor.
5) Fuel rod bow factor.
6) ~~Lumped b~~Burnable poison manufacturing tolerance factor.

The nuclear uncertainty factor accounts for the uncertainty in the calculated peak due to the limitations of the analytical models. The spacer grid effect factor accounts for the flux distortion caused by Inconel spacer grids (no spacer grid effect factor is required for Zircaloy spacer grids). The engineering hot channel factor accounts for the manufacturing tolerances of critical fuel rod design parameters (pellet enrichment, pellet density, pellet diameter, etc.). The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor that is dependent on the location of the pin within the assembly. A burnup dependent peaking penalty consistent with the topical reports BAW-10147-PA (Reference 10) and BAW-10186-PA (Reference 7) is applied to account for the potential power peaking enhancement due to fuel rod bow. The ~~lumped~~ burnable poison manufacturing tolerance factor accounts for the effect of the variance in the as-built enrichments of the lumped burnable poison (LBP) pellets in LBP rods **or the gadolinia in gadolinia-bearing fuel rods**. The statistical combination of these terms is described in Reference 15.

To determine the allowable values of full-length and part-length (APSR) control rod positions and the axial offsets, first an operating range for the full-length control rod position is chosen and then the ranges of axial offsets and part-length control rod positions for which the worst case operating peaking factors at the designated axial planes are less than or equal to their respective allowable values are determined. If the resulting ranges of axial offset and part-length control rod position are acceptable from the standpoint of operational flexibility, the assumed full-length control rod position ranges and the calculated range of axial offset and part-length control rod position are taken as their operating limits. If, however, the resulting ranges of axial offsets and part-length control positions are unacceptable from the standpoint of operational flexibility, a more restrictive full-length control rod bank position is selected and the corresponding axial offset and part-length control rod position limits are established.

9.0 DEVELOPMENT OF CORE PHYSICS PARAMETERS

Upon completion of the reload design, a variety of physics parameters have been generated primarily for HFP and some HZP conditions. The purpose of this stage of developing core physics parameters is to provide additional calculations to supplement those already performed. These calculations are performed using the methodology described in Reference 2 **or Reference 17**. The results of these calculations are used for startup test predictions and core physics parameters throughout the cycle. Changes to the startup test procedures, plant operations, or particular core designs may change the physics parameters that are required. The following descriptions are typical of current requirements.

9.1 Startup Test Predictions

After each refueling, the reactor undergoes a startup test program aimed at verifying that the reactor core is correctly loaded, that control rods are in the correct locations and are functioning properly, and that reactor behavior is accurately predicted by the nuclear models which were used in generating the data used in the plant's safety analyses.

9.1.1 Critical Boron Concentrations and Boron Worths

Critical boron concentrations and boron worths are typically calculated at a variety of rod configurations, at HZP and HFP, as a function of boron concentration, at different xenon concentrations, and at different times in the fuel cycle. The calculation model is capable of critical boron searches and when critical boron concentrations are desired is usually run in this mode. An acceptable alternative, however, is to not search on critical boron but to correct the input boron concentration to the critical boron concentration using a calculated boron worth and the calculated reactivity.

Both HFP and HZP critical boron calculations are normally performed for startup physics tests. Soluble boron worths are usually calculated at HFP and HZP for startup physics tests. The boron worths are usually calculated by running two similar cases except that the soluble boron concentration is varied. The differential boron worth is calculated by subtracting the reactivities and dividing by the boron difference. Differential boron worths are usually quoted in %Δρ/100 ppmb or in ppmb/%Δρ (the latter term is sometimes referred to as the inverse boron worth).

Critical boron concentration is calculated as a function of cycle burnup. These predictions may be provided in tabular form.

Differential boron worth may be calculated as a function of boron concentration and also as a function of cycle burnup. These predictions may also be provided in tabular form.

10.0 REFERENCES

1. Oconee Nuclear Station, Units 1, 2, and 3, Updated Final Safety Analysis Report, Docket Nos. 50-269, -270, and -287.

2. Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004-A, Revision 1a, Duke Energy Carolinas, January 2009.

3. UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-PA, Revision ~~3a~~**4**, Duke Energy Carolinas, ~~February 2009~~***publication date***.

4. Fuel Mechanical Reload Analysis Methodology using TACO3 **and GDTACO**, DPC-NE-2008-PA, Revision ~~1a~~**2**, Duke Energy Carolinas, ~~December 2008~~***publication date***.

5. Oconee Nuclear Station Core Thermal-Hydraulic Methodology using VIPRE-01, DPC-NE-2003-PA, Revision ~~2a~~**3**, Duke Energy Carolinas, ~~December 2008~~***publication date***.

6. Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005-PA, Revision 4a, Duke Energy Carolinas, December 2008.

7. Extended Burnup Evaluation, BAW-10186-PA, Revision 1, AREVA NP, April 7, 2000.

8. Letter from D. L. LaBarge (NRC) to W. R. McCollum, Jr. (ONS), Use of Framatome Cogema Fuels Topical Report on High Burnup – Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. MA0405, MA0406, MA0407), Docket Nos. 50-269, 50-270, and 50-287; March 1, 1999.

9. Letter from M. S. Tuckman to Document Control Desk, Duke Energy Corporation's Use of FCF's Extended Burnup Evaluation Topical Report, BAW-10186-PA, August 25, 1999.

10. Fuel Rod Bowing in Babcock and Wilcox Fuel Designs, BAW-10147-PA, Revision 1, AREVA NP, June 28, 1983.

11. Letter from H. N. Berkow to M. S. Tuckman, Subject: Duke Power use of CROV Computer Code, June 19, 1995.

12. Fuel Reconstitution Analysis Methodology, DPC-NE-2007-PA, Revision 0, Duke Power Company, October 1995.

13. Letter from D. E. LaBarge to M. S. Tuckman, Subject: Oconee Nuclear Station, Units 1, 2, and 3, RE: Topical Report DPC-NE-2003, Rev. 1 (TAC Nos. MA8234, MA8235, MA8236), June 23, 2000.

14. Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, BAW-10227-PA, AREVA NP, February 11, 2000.

15. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002-A, Revision ~~3b~~**4**, Duke Energy Carolinas, ~~June 2009~~***publication date***.

16. Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4~~**5**, Duke Energy Carolinas, ~~October 2008~~***publication date***.

17. Oconee Nuclear Station Reload Design Methodology Using CASMO-4/SIMULATE-3, DPC-NE-1006-P, Revision 0, Duke Energy Carolinas, May 2009

Appendix B

Revision History

Date	Occurrence
April 23, 1979	Revision 0 of NFS-1001 submitted to the NRC
May 20, 1980	Revision 1 of NFS-1001 submitted to the NRC
October 16, 1980	First NRC RAI issued
November 13, 1980	First Duke response to the first RAI submitted
January 28, 1981	Second Duke response to the first RAI and Revision 2 of NFS-1001 submitted to the NRC
March 18, 1981	Third Duke response to the first RAI submitted
April 22, 1981	Revision 3 of NFS-1001 submitted to the NRC
June 2, 1981	Second NRC RAI issued
June 16, 1981	Duke response to the second RAI and Revision 4 of NFS-1001 submitted to the NRC
July 29, 1981	NRC SER issued
December 22, 1999	Revision 5 of NFS-1001 submitted to the NRC
May 24, 2000	NRC RAI issued
August 23, 2000	Duke response to the RAI submitted
December 8, 2000	NRC SER issued
October 22, 2007	DPC-NE-2015 submitted to the NRC
September 17, 2008	Duke response to the (emailed) RAI submitted
October 29, 2008	NRC SER issued for DPC-NE-2015
June 2009	Revision 6a of NFS-1001 approved
publication date	**Revision 7 of NFS-1001 published**

NFS-1001 was originally submitted to the Nuclear Regulatory Commission (NRC) in April 1979. It was approved by the NRC in July 1981. In between the original submittal and the approval were two NRC Requests for Additional Information (RAI), four separate Duke responses to the RAIs, and four revisions of the report, as shown in the table above. It was Duke's practice at that time to issue a new revision of the report if the RAI response modified the report in any way. The original NRC Safety Evaluation Report (SER) for NFS-1001 was dated July 29, 1981 and it approved Revision 4 of the report.

Revision 5 of NFS-1001 was submitted to the NRC in December 1999 and it was approved by the NRC with the SER dated December 8, 2000.

Revision 6 of NFS-1001 was submitted to the NRC via DPC-NE-2015 (Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology) in October 2007. This transition report was approved by the NRC with the SER dated October 29, 2008.

Revision 6a of NFS-1001 was approved in June 2009. This revision contained the following significant changes. The first three changes were approved by the NRC in DPC-NE-2015, and the remaining changes were implemented via 10CFR50.59.

1) The Mark-B-HTP fuel design was added to the list of fuel assembly designs in Section 2.1.

2) The fuel densification power spike factor, which was specified as a value of 1.08, was replaced with an axially-dependent factor.

3) A fuel assembly bow penalty factor and a lumped burnable poison manufacturing tolerance factor were applied to CFM, DNB and LOCA margins.

4) The target average core moderator temperature was corrected from approximately 580 °F to approximately 579 °F.

5) The start of the average moderator temperature plateau was changed from approximately 15% full power to an approximate range of 15-20 % full power based on the efficiency of the steam generators.

Revision 7 of NFS-1001 was published in *mmm 2010.* This revision updated the reference methodology report DPC-NE-2008 due to the inclusion of GDTACO in the title. It added DPC-NE-1006 to the list of references and approved methodologies and updated the DNB, CFM, and LOCA margins discussion to add a description of the burnable poison manufacturing tolerance factor for gadolinia as it is included in the SCUF equations in DPC-NE-1002. It also added the engineering hot channel factor for HTP with gadolinia.

ATTACHMENT 4

DPC-NE-1002-A – Oconee Nuclear Station Reload Design Methodology II (Revision 3b) – Mark-Up

Oconee Nuclear Station
Reload Design Methodology II

DPC-NE-1002~~-A~~
Revision ~~3b~~4

~~June 2009~~ October 2009

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

ABSTRACT

This report was originally written as a supplemental report to NFS-1001-A (Reference 1). The subsequent revision of NFS-1001-A, the replacement of CASMO-2 with the CASMO-3 lattice computer code, and the obsolescence of the Mark B5-Z fuel assembly design has made most of the material content of this report obsolete. The obsolete material was deleted in Revision 3a. The remaining content of this report consists mostly of the equations for the statistically combined uncertainty factors (SCUFs) for the centerline fuel melt (CFM), departure from nucleate boiling (DNB), and loss of coolant accident (LOCA) criteria. These SCUF equations were updated in Revision 3b to match the SCUF equations described in DPC-NE-2015-PA (Reference 3). **The CASMO-4/SIMULATE-3 code system is introduced in Revision 4 and is an alternative to, and eventually a replacement for, the CASMO-3/SIMULATE-3 code system for the generation of nuclear data and power distributions. The definition of the LBP term is updated in the SCUF equations to include both lumped and integral burnable absorbers.**

(continued from Chapter 1.0 of DPC-NE-1002)
The remaining content of DPC-NE-1002 following Revision 3a consisted mostly of the equations for the statistically combined uncertainty factors (SCUFs) for the centerline fuel melt (CFM) and loss of coolant accident (LOCA) criteria. The nuclear uncertainty factors used in these two SCUF equations were updated for the CASMO-3 based methodology approved in DPC-NE-1004-A (Reference 2). A burnup-dependent fuel rod bow penalty was added to both SCUF equations, and the bounding fuel densification power spike factor in the CFM SCUF equation was changed to an axially-dependent factor.

Revision 3b of DPC-NE-1002 adds a new SCUF equation for departure from nucleate boiling (DNB) criteria, and updates the SCUF equations for CFM and LOCA criteria. A fuel assembly bow factor and a lumped burnable poison manufacturing tolerance factor are added to each SCUF equation. These new and updated SCUF equations are taken from the response to RAI Question # 2 in the NRC-approved DPC-NE-2015-PA report (Reference 3).

Revision 4 of DPC-NE-1002 revises the LBP term in the SCUF equations to simply call it a burnable poison term since it will include gadolinia effects as well as LBP effects. The report is also updated to include reference to CASMO-4/SIMULATE-3 uncertainties documented in Reference 4.

2. TECHNICAL SPECIFICATIONS REVIEW AND DEVELOPMENT

Section 7 of NFS-1001-A discusses the process involving the review and development of Technical Specifications for Oconee reload designs. Except for the minor revisions discussed below, the methodology described in NFS-1001-A remains unchanged.

A. Section 7.2.2.1 of NFS-1001-A

The SCUF equation for centerline fuel melt and cladding strain (CFM/CS) analysis is shown below.

$$SCUF_{CFM/CS} = 1.0 + Bias + \sqrt{(U_{A-T})^2 + (U_{R-L})^2 + EHC^2 + \cancel{L}BP^2 + FRB^2 + FAB^2 + FSPIKE^2}$$

where :

- Bias = assembly total power bias
- U_{A-T} = assembly total uncertainty
- U_{R-L} = pin (radial-local) uncertainty
- EHC = engineering hot channel factor
- ~~L~~BP = ~~lumped~~ burnable poison manufacturing tolerance factor
- FRB = fuel rod bow factor (varies by assembly exposure)
- FAB = fuel assembly bow factor (varies by location of pin within each assembly)
- FSPIKE = fuel densification power spike factor (varies by axial location)

The first three factors (Bias, U_{A-T} and U_{R-L}) comprise a total nuclear uncertainty factor which accounts for the uncertainty in the calculated total peak ~~due to the limitations of~~**associated with** the analytical models. The values for these three factors, and the equation for the total nuclear uncertainty factor, are defined in Section 5.1 of DPC-NE-1004-A (Reference 2). **Duke will transition to the code package described in DPC-NE-1006-P (Reference 4) following NRC approval of Reference 4. Core designs based on DPC-NE-1006 obtain the values for the first three factors from Tables 5.1 and 5.2 of that report using the total nuclear uncertainty factor equations defined in Sections 5.2 and 5.3 of that report.**

The engineering hot channel factor (EHC) accounts for the manufacturing tolerances of critical fuel rod design parameters (pellet enrichment, pellet density, pellet diameter, etc.). The value for this factor is defined in Section 7.2.2.1 of NFS-1001-A (Reference 1).

The effect of ~~lumped~~ burnable poison manufacturing tolerances on the pin power distribution is accounted for by a penalty factor (~~L~~BP). The effect of fuel rod bow on the pin power distribution is accounted for by a penalty factor (FRB) that is a function of assembly exposure. The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor (FAB) that is a function of pin location within each assembly. Fuel pin regions within each assembly are defined in the application of the fuel assembly bow penalty. The values of these three factors are determined by Duke.

The fuel densification power spike factor (FSPIKE) accounts for the local flux enhancement resulting from gaps in the fuel column induced by fuel densification. This factor varies by axial location and is provided to Duke by the fuel vendor. Linear interpolation is used within the table to determine a power spike factor to be applied at each axial level in the core model. Duke adds a power spike factor of 1.0 at zero elevation to the table to preclude extrapolation to a power spike factor less than 1.0 at the bottom of the core.

A separate CFM/CS SCUF is calculated for each combination of assembly exposure, axial location, and fuel pin region. Each nodal (axial) pin power in the limiting fuel pin in each fuel pin region is multiplied by the appropriate CFM/CS SCUF before it is compared to the CFM/CS kw/ft limit.

B. Section 7.2.2.2 of NFS-1001-A

The SCUF equation for statistical core design (SCD) based DNB analysis is shown below.

$$SCUF_{DNB} = 1.0 + Bias + \sqrt{\cancel{L}BP^2 + FAB^2}$$

where :

Bias = assembly radial power bias

~~L~~BP = ~~lumped~~ burnable poison manufacturing tolerance factor

FAB = fuel assembly bow factor

The pin powers are multiplied by this SCUF before comparison to the DNB maximum allowable radial peak (MARP) limit.

The value for the assembly radial power bias (Bias) is defined in Section 5.2 of DPC-NE-1004-A (Reference 2). **Following NRC approval of Reference 4, the assembly radial power bias (Bias) is obtained from Tables 5.1 and 5.2 of DPC-NE-1006-P (Reference 4).** The effect of ~~lumped~~ burnable poison manufacturing tolerances on the pin power distribution is accounted for by a penalty factor (~~L~~BP) determined by Duke. The effect of fuel assembly bow on the pin power distribution is accounted for by a single penalty factor (FAB) determined by Duke using the method described below.

Duke modifies a VIPRE model to reflect the bowed condition gap geometry. The relative power densities (RPD) of the fuel assembly pins are augmented by the fuel assembly bow pin power peaking factors. The subchannel form loss coefficients are updated to reflect the bowed conditions. A DNB ratio is calculated for both the bowed and non-bowed conditions. The difference in DNB ratios between the two cases is converted to a radial peaking penalty to be used in reload design calculations.

Based on sensitivity studies that consider a wide range of statepoint conditions and axial power shapes, a single bounding fuel assembly bow pin peaking factor for DNB analyses is determined.

3. REFERENCES

1. Oconee Nuclear Station Reload Design Methodology, NFS-1001-A, Revision ~~5~~7, Duke Energy Carolinas, ~~January 2001~~***publication date***.

2. Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004-A, Revision 1a, Duke Energy Carolinas, January 2009.

3. Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology, DPC-NE-2015-PA, Revision 0, Duke Energy Carolinas, October 2008.

4. **Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3, DPC-NE-1006-P, Revision 0, Duke Energy Carolinas, May 2009.**

ATTACHMENT 9

DPC-NE-2003-A – Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01 (Revision 2a) – Mark-Up

Oconee Nuclear Station
Core Thermal-Hydraulic Methodology Using
VIPRE-01

DPC-NE-2003
Revision ~~2a~~3

~~December 2008~~October 2009

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

Revision 3 to DPC-NE-2003-A

Revision 3 updates the report to address the use of gadolinia in the methods and models. To model gadolinia in Oconee core designs, it is necessary to use CASMO-4/SIMULATE-3 as documented in DPC-NE-1006-P. The axial power uncertainty associated with CASMO-4/SIMULATE-3 is different than that associated with CASMO-3/SIMULATE-3. Therefore, Section 5.11 is revised for the new value. Additionally, the specific values of the various hot channel factors (HCFs) in Section 5.11 are updated. The reference radial power distribution discussion in Section 5.9 is revised to note that gadolinia will not change the bounding conservative distribution used in the generation of the MAP curves as described in this report. Finally, DPC-NE-1006-P is added to the reference list and several of the references are updated to the current versions.

5.9 REFERENCE DESIGN POWER DISTRIBUTION

The reference design power distributions are shown in Figures 4-1 through 4-7. The power distributions were designed to be conservatively high and relatively flat in the vicinity of the hot subchannel. The pin power peaking gradient within the area of the hot subchannel is approximately 1%. The pin power distribution was verified to be conservative by comparison with predicted physics power distributions. The reference design power distribution was developed using a radial-local hot pin peak, $F\Delta H^N$, of 1.714 and an assembly power of 1.6147. The $F\Delta H^N$ = 1.714 is the same reference pin peak used in the methodology discussed in Reference 1. The limiting flow coastdown transient is analyzed as discussed in Section 6.6 using the reference design power distribution. A different design power distribution may be used to add or delete margin in the transient analysis. As discussed in Section 6.5 and 6.6 maximum allowable peaking (MAP) limits are calculated to define combinations of radial and axial peaking that provide equivalent DNB protection.

For the Mark-B-HTP fuel design the reference design power distributions are shown in Figures 4-6 and 4-7. The pin power values in Rods 2 and 4 have been reduced relative to the original reference power distribution to ensure that the hot subchannel remains subchannel 1.

Utilizing an integral burnable absorber in the UO_2 fuel matrix, such as gadolinia, will perturb the pin power distribution, particularly in and adjacent to the fuel rods containing gadolinia. However, the reference design power distribution used in the generation of the MAP curves as described here remains bounding and conservative.

5.11 HOT CHANNEL FACTOR

The local heat flux factor, Fq ", and the power factor, Fq , are conservatively applied to the hot subchannel (i.e., the instrument guide tube subchannel) of the hot assembly to compensate for possible deviations of several parameters from their design values.

For non-SCD analyses, Fq" is only used in the computation of the surface heat flux of the hot pin when calculating the DNBR in the hot subchannel. Previously, Fq″ included factors/penalties accounting for the (1) effects of local variations in the pellet enrichment and weight on local (hot spot) power, (2) power spikes occurring as a result of flux depressions at spacer grids, and (3) axial nuclear uncertainty. An Fq″ factor accounting for all these effects was applied when calculating Maximum Allowable Peaking Limits (MAP) limits. Since References 7 and 8 show that local heat flux spikes have no effect on the critical heat flux results, the first two penalties are not required. Therefore, Fq″ is only used to account for axial nuclear uncertainty. For non-SCD analyses **based on CASMO-3/SIMULATE-3 methods,** the Fq″ ~~for Mark-BZ and Mark-B11 fuel~~ is [ ], Reference 6. **For analyses based on CASMO-4/SIMULATE-3 methods, Fq″ is [ ], Reference 16.** For SCD analyses, the axial nuclear uncertainty is accounted for by using the SCD limit.

The power factor **(or hot channel factors),** Fq ~~= [ ] Reference 11 (Mark-BZ) and [ ] (Mark-B11) Reference 11~~, accounts for variations in average pin power caused by differences in the absolute number of grams of U-235 per rod. The manufacturing tolerance on U-235 per fuel stack and variation on the powder lot mean enrichment are considered in determining the factor. For non-SCD analyses, Fq is applied directly to the heat generation rate of the hot pin of the hot subchannel. For SCD analyses, Fq is statistically applied to the heat generation rate of the hot pin of the hot subchannel to determine an SCD limit. The hot channel factors ~~for the Mark-B-HTP design are given in Appendix F to Reference 11~~ **are [ ] for Mark-BZ fuel (Reference 11), [ ] for Mark-B11 fuel (Reference 11), [ ] for Mark-B-HTP fuel (no gadolinia), and 1.0145 for Mark-B-HTP fuel (with gadolinia).** ~~The application of these factors is the same as described above for the other designs.~~

7.0 REFERENCES

1. Oconee Nuclear Station Reload Design Methodology, NFS-1001A, Rev. ~~5~~**7**, ~~January 2001~~***publication date.***

2. J. M. Cuta, et. Al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," EPRI-NP-2511-CCM, Rev. 4, EPRI, February 2001.

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9. Duke Energy Carolinas Oconee Nuclear Station UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-PA, Rev. ~~2, May 2005~~**4**, ***publication date.***

10. RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-7450(A), EPRI, Revision ~~5~~**6.3**, July ~~2001~~**2007**.

11. Duke Energy Carolinas Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005-PA, Rev. 4a, December 2008.

12. BAW-10229P-A, Mark-B11 Fuel Assembly Design Topical Report, AREVA NP, Lynchburg, Va., October 26, 1999.

13. BHTP DNB Correlation Applied with LYNXT, BAW-10241(P)(A), Revision 1, AREVA NP, July 2005

14. The BWU Critical Heat Flux Correlations, BAW-10199P-A, AREVA NP, August 1996 (and including Addendum 1, December 2000)

15. Reactor Vessel Model Flow Tests, BAW-10037, Revision 2, AREVA NP, November 1972

16. Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3, DPC-NE-1006-P, Rev. 0, May 2009

ATTACHMENT 10

DPC-NE-2008-A – Fuel Mechanical Reload Analysis Methodology Using TACO3 (Revision 1a) – Mark-Up

Fuel Mechanical Reload Analysis
Methodology Using TACO3 and GDTACO

DPC-NE-2008
Revision ~~1a~~2

~~December 2008~~October 2009

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

ABSTRACT

This report presents Duke Energy Carolina's (Duke's) fuel rod mechanical reload analysis methodology using TACO3 **and GDTACO.** ~~,~~ **TACO3 and GDTACO are**~~a~~ best estimate quasi steady-state fuel performance codes written by B&W Fuel Company (now AREVA NP). For each reload cycle, analyses must be performed to ensure that the generic, or specific, analyses envelope the operation of the fuel. This report describes the methodology for performing the following fuel rod analyses using TACO3**, GDTACO,** and other methods:

8. Linear Heat Rate to Melt (LHRTM)
9. Pin Pressure
10. Cladding Strain
11. Creep Collapse
12. Cladding Corrosion
13. Cladding Stress
14. Cladding Fatigue

The methodology described in this report applies to any AREVA NP fuel design used at McGuire, Catawba, or Oconee Nuclear Stations.

Revision 1a updates the methodology for cladding creep collapse and describes the methodology for cladding corrosion, stress, and fatigue. Those changes were reviewed and approved by the Nuclear Regulatory Commission in connection with methodology report DPC-NE-2015-PA, Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology. In addition, Revision 1a includes a number of minor editorial and clarification changes.

Revision 2 includes the AREVA NP GDTACO code for modeling the thermal and mechanical behavior of gadolinia fuel pellets.

Table of Contents

List of Figures

Fuel Mechanical Reload Analysis Methodology using TACO3 and GDTACO

1.0 Introduction

Fuel rod mechanical and thermal assessments must be performed for each reload fuel cycle. Generic analyses are completed for each fuel design. The generic analyses are expected to envelope the operation of future fuel cycles. For each reload, the specific fuel cycle design is compared with the generic fuel rod analyses. This report discusses the generic TACO3 **and GDTACO** analyses that are performed and the comparisons that must be made to verify that the generic analyses are applicable to each reload design. In most cases, the fuel cycle design is bounded by the generic analyses and no reanalyses are required. In some cases it is necessary to apply these methods to cycle-specific analyses. This report describes the methodology for performing the following fuel rod analyses using TACO3 (~~ref.~~**Reference** 3) **for UO_2 fuel, GTDACO (Reference 14) for gadolinia fuel**, CROV (~~ref.~~**Reference** 8), and COROS02 (~~refs.~~**References** 10 and 11).

8. Linear Heat Rate to Melt (LHRTM)
9. Pin Pressure
10. Cladding Strain
11. Cladding Creep Collapse
12. Cladding Corrosion
13. Cladding Stress
14. Cladding Fatigue

TACO3 is a best estimate quasi steady-state fuel performance code written by BWFC (now AREVA NP). TACO3 includes gap conductance, fuel densification and swelling, fuel restructuring, cladding creep and deformation, gap closure, and fission gas release models to provide best estimate predictions of fuel rod temperatures and pressures. **GDTACO is basically the TACO3 code with additional subroutines and correlations to include the necessary models and data to simulate gadolinia fuel. Application of the GDTACO code is summarized in Appendix A.** The CROV code was developed by Babcock and Wilcox (now AREVA NP) to calculate fuel rod cladding ovality changes due to thermally and irradiation induced creep. CROV is used to provide a conservative estimate of conditions at which cladding collapse would occur. The COROS02 model was developed by Framatome (now AREVA NP) to predict fuel rod cladding oxide thickness.

The methodology described in **the body of** this report applies to any AREVA NP fuel design with low enriched uranium fuel pellets and Zircaloy-4 or M5™ cladding. **Appendix A is specific to AREVA NP fuel designs containing gadolinia.** The NRC-approved generic fuel rod burnup limit for the Mark-B and Mark-BW fuel designs is 62,000 megawatt-days per metric ton uranium (MWd/mtU), based on the NRC Safety Evaluation of AREVA Topical Report BAW-10186P-A, Revision 1 (Reference 13). It is noted that allowable maximum burnups may be lower due to site-specific considerations.

4.0 References

1. Deleted

2. Deleted

3. TACO3 Fuel Pin Thermal Analysis Computer Code, BAW-10162P-A, B&W Fuel Company, November 1989

4. Fuel Rod Gas Pressure Criterion, BAW-10183P-A, B&W Fuel Company, July 1995

5. Deleted

6. Deleted

7. Deleted

8. Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse, BAW-10084P-A, Revision 3, B&W Fuel Company, August 1995

9. Letter, H. N. Berkow (NRC) to M. S. Tuckman (Duke), Subject: Duke Power use of CROV Computer Code, June 19, 1995

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11. Letter D. LaBarge (NRC) to W. R. McCollum, Jr. (Duke), Subject: Use of Framatome Cogema Fuels Topical Report on High Burnup – Oconee Nuclear Station, Units 1, 2 and 3, March 1, 1999

12. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, BAW-10179P-A, Revision 7, AREVA NP, January 2008

13. Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, BAW-10227P-A, Revision 1, AREVA NP, June 2003

14. GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code, BAW-10184P-A, AREVA NP, February 1995.

Appendix A

Application of GDTACO for Gadolinia Fuel

The GDTACO code (Reference A-1) was developed by AREVA NP to model UO_2 fuel pellets in which gadolinia (Gd_2O_3) powder is used as an integral burnable absorber. NRC approval for GDTACO is in the safety evaluation (Reference A-2). GDTACO is based on the TACO3 code (Reference A-3) that is used to model uranium dioxide fuel, and includes the following new models:

- Radial power profile
- Fuel thermal conductivity
- Fuel melting point
- Fuel densification and swelling

The details of each of these new models are in Reference A-1. The remaining models in TACO3 are unaffected by the presence of gadolinia, and are also used in GDTACO. All of the clad models in TACO3 were retained in GDTACO since the composition of the fuel pellet does not affect the behavior of the clad.

The GDTACO code is labeled as a quasi-best-estimate code since conservatism has been added to the code where benchmarking data is sparse or non-existent. Data from 4 and 8 weight percent gadolinia rods irradiated in the Oconee Nuclear Station Unit 1 reactor to a fuel assembly burnup of [ ] MWd/mtU (Reference A-4) were used to benchmark the code, along with other sources of data.

The GDTACO code will be used for gadolinia fuel for the same applications as the TACO3 code is used for UO_2 fuel, specifically:

- Fuel melting
- Fuel rod internal gas pressure
- Cladding strain
- Creep collapse initialization

Duke will apply the GDTACO code in the same manner as the TACO3 code is applied as described in the main body of this report, with the GDTACO input file activating the gadolinia-specific models and including the additional input required to designate and describe the gadolinia fuel. The differences in the application of the methodology are all related to the differences in material properties.

The NRC safety evaluation (Reference A-2) includes the following restrictions that will be complied with by Duke in the application of the GDTACO methodology:

1) Limited to gadolinia concentrations up to 8 weight percent
2) Cycle-specific analyses must be performed for each reload

Duke will comply with the second restriction by performing generic fuel rod mechanical analyses for UO_2 and gadolinia fuel rods. For each reload cycle the specific fuel cycle design is compared with the generic fuel rod analyses to verify that the generic analyses are applicable to each cycle design. In most cases, the fuel cycle design is expected to be bounded by the generic

analyses and no reanalyses are required. If required, a specific analysis is performed to verify that the fuel rod design criteria are met.

References

A-1. GDTACO – Urania Gadolinia Fuel Pin Thermal Analysis Code, BAW-10184P-A, AREVA NP, February 1995.

A-2. Letter, A. C. Thadani (NRC) to J. H. Taylor (AREVA NP), June 24, 1993 (GDTACO Safety Evaluation)

A-3. TACO3 Fuel Pin Thermal Analysis Computer Code, BAW-10162P-A, AREVA NP, October 1989

A-4. Extended Burnup Evaluation, BAW-10186P-A, AREVA NP, Rev. 2, June 2003

ATTACHMENT 11

DPC-NE-3000-A – Thermal-Hydraulic Transient Analysis Methodology (Revision 4a) – Mark-Up

Oconee Nuclear Station
McGuire Nuclear Station
Catawba Nuclear Station

THERMAL-HYDRAULIC TRANSIENT
ANALYSIS METHODOLOGY

DPC-NE-3000
Revision ~~4a~~**5**

~~March 2009~~**October 2009**

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

Abstract

This report is the Duke Power Company response to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Action." G. L. 83-11 requires that licensees performing their own safety analyses demonstrate their analytical capabilities. Comparisons of computer code results to experimental data, plant operational data, or other benchmarked analyses were identified as areas of interest. This report describes the RETRAN-02 transient thermal-hydraulic models developed for the Oconee, McGuire, and Catawba Nuclear Stations, and the VIPRE-01 core thermal-hydraulic models developed for the Oconee, McGuire, and Catawba Nuclear Stations. Comparisons of Oconee RETRAN model predictions to nine plant transients, and comparisons of McGuire/Catawba RETRAN model predictions to eight plant transients, are detailed. VIPRE model predictions are validated by comparisons to the COBRA-IIIC/MIT code for the Oconee core design. The report concludes that the analytical capability to perform non-LOCA transient thermal-hydraulic analyses has been demonstrated.

Revision 1 describes the methodology revision for the McGuire and Catawba Unit 1 replacement steam generators and other minor revisions.

Revision 2 describes the methodology revision for the AREVA NP Mk-B11 fuel assembly design for Oconee and other minor revisions.

Revision 3 describes the methodology revisions for the Oconee replacement steam generators, the RETRAN-3D MOD003.1/DKE code, the Westinghouse RFA fuel assembly design, and other minor revisions.

Revision 4a describes the methodology revisions associated with the AREVA NP Mk-B-HTP fuel design (Appendix D). This revision also includes an expanded VIPRE model for Oconee (Appendix E) and other minor revisions.

Revision 5 adds minor editorial revisions to describe gadolinia fuel in Section 2.3.3.4 and Appendix D and revises the hot channel factor discussion also provided in Appendix D.

(Section 2.3 of main body of DPC-NE-3000)

2.3.3.3 Fuel Pin Conduction Model

For most of the transient analyses, the RETRAN heat flux boundary condition is used; the fuel pin conduction model will not be used in the VIPRE-01 transient models. This means that heat is added directly from the cladding surface to the fluid as a boundary condition on the calculation, and the heat transfer solution is not required. However, for transient analyses that require detailed modeling of fuel rod temperatures the VIPRE fuel pin conduction model is used with the neutron power as the transient forcing function.

2.3.3.4 Power Distribution

Radial Power Distribution

For transients resulting in symmetrical power distributions, the 15 x 15 1/8 core assembly radial power distribution and hot assembly pin radial-local power distributions shown in Figures 2.3-3 and 2.3-4 are applied. The hot assembly has a radial peak of 1.613 (Figure 2.3-3), and contains the maximum pin radial-local peak of 1.714 (Figure 2.3-4). For transients resulting in asymmetric radial power distributions, nuclear design analyses generate radial power distributions. Radial power distribution as a function of transient time is then input to VIPRE-01. **It should be noted that the introduction of gadolinia (see Appendix D) will perturb the radial power distribution of the hot assembly due to the effects of the integral burnable poison but that the distribution given in Figure 2.3-4 remains conservative for gadolinia fuel in the bounding generic applications.**

Axial Power Distribution

For transients resulting in symmetric radial power distributions, the 1.5 chopped cosine axial power shape is typically applied (Figure 2.3-8). For transients resulting in asymmetric radial power distributions, nuclear design analyses generate axial power distributions during transients. On a case-by-case basis, either these axial power distributions or the 1.5 chopped cosine shape will be utilized as justified.

APPENDIX D

METHODOLOGY REVISIONS FOR MARK-B-HTP FUEL

This appendix contains non-LOCA thermal-hydraulic transient analysis methodology revisions related to the Mark-B-HTP fuel assembly design. This fuel design is characterized by larger diameter fuel pins, larger diameter fuel pellets, non-mixing vane grids, and a different axial pressure drop distribution relative to the current Mark-B11 fuel design. The information included in this appendix is in addition to that presented in Sections 2.1.2.1, 2.2.3.1, and 2.3 of the main body of this report.

Fuel Assembly Description

The Mark-B-HTP fuel assembly consists of spacer grids, end fittings, fuel rods, and guide tubes. The lower end HMP grid is made of Inconel Alloy 718, while the six intermediate spacer grids and the upper grid are made of M5®. The intermediate spacer grids are comprised of the HTP non-mixing vane grid type. Each fuel assembly is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube, all M5®. The fuel rod consists of dished-end, cylindrical pellets of uranium dioxide. **In addition, a small number of fuel rods in each assembly may contain fuel pellets consisting of uranium dioxide mixed with the burnable poison gadolinia. The models and input described in this report are valid for fuel containing both uranium dioxide and uranium dioxide mixed with gadolinia.** The fuel assembly and fuel rod dimensions, and other related fuel parameters used in the thermal-hydraulic analyses are given in Table D-1. A drawing of the Mark-B-HTP fuel assembly is shown in Figure D-1. These materials and design parameters are used in developing the RETRAN-3D and the VIPRE-01 models for the Mark-B-HTP fuel designs.

VIPRE Models

The various VIPRE models described in Section 2.3 of the main body of this report are used for the Mark-B-HTP thermal-hydraulic analyses with the VIPRE-01 thermal-hydraulic computer code described in Reference D-1. The models are updated to reflect the Mark-B-HTP fuel parameters listed in Table D-1.

Code Option and Input Selections

Thermal-Hydraulic Correlations:
No changes to the correlations used in the main body of the report and Appendix A.

Turbulent Mixing Correlations:
The turbulent mixing factors for the Mark-B-HTP design, [ ] for the HTP grid and [ ] for the HMP grid, were provided by AREVA NP based on scaled testing.

Pressure Losses:
The pressure losses are calculated in VIPRE-01 as described in the main body of this report using the spacer grid form loss coefficients for the Mark-B-HTP fuel assembly provided by AREVA NP.

Critical Heat Flux Correlation

The CHF correlation that is generally used for the Mark-B-HTP design is the AREVA NP BHTP correlation (Reference D-2). Below the first intermediate grid the BWU-N correlation is used (Reference D-3). The range of conditions for which the BHTP correlation is valid is:

Pressure (psia)	1385 to 2425
Mass flux (Mlbm/hr-ft2)	0.492 to 3.549
Quality	Less than 0.512

Using the VIPRE-01 code, Duke has verified that the BHTP correlation design limit is 1.132. Applying Duke's statistical design methodology, the statistical design limit is [ ] for the following range of conditions (this limit is also applicable below the first intermediate grid):

Core power (%RTP)	76.0 to 140.0
RCS flow (% of design)	60.0 to 115.0
Pressure (psia)	1600 to 2242
Core inlet temperature (F)	505.0 to 572.8

Due to some of the main steam line break cases reaching pressure values lower than the BHTP correlation applicability limits shown above, the Modified-Barnett CHF correlation (Reference D-4) is used. The Modified-Barnett CHF correlation is valid for the following range of conditions:

Pressure (psia)	150 to 725
Mass flux (Mlbm/hr-ft2)	0.03 to 1.70
Inlet subcooling (ΔBtu/lbm)	6.0 to 373.0
Heated length (inches)	32.9 to 174.8
Axial heat flux	Uniform

AREVA NP has obtained NRC review and approval for the Modified-Barnett CHF correlation for application to main steam line break in Reference D-5. The 95/95 correlation DNBR limit is 1.135 (References D-6 and D-7) for the Modified-Barnett CHF correlation. Duke will use the 1.135 DNBR limit established by AREVA NP. This methodology will be used for main steam line break analyses when the BHTP correlation cannot be used.

It is noted that there is a range of pressure values for which neither the BHTP nor the Modified-Barnett CHF correlations are valid. Duke will conservatively reduce the pressure values to the 725 psia upper range limit for the Modified-Barnett if a limiting statepoint falls between the applicable pressure ranges.

Hot Channel Factors

The hot channel power factor, F_q, is computed statistically from the average or overall variation on rod diameter, enrichment, and fuel weight per rod. It is applied to the heat generation rate in the pin; thus it will have an effect on all terms that are computed from this heat rate with the exception of the heat flux for DNB ratio computation. The value of F_q used is [ ] for Mark-B-HTP fuel **without gadolinia and 1.0145 for Mark-B-HTP fuel with gadolinia.**

<u>References</u> **(for Appendix E of DPC-NE-3000)**

E-1 UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-PA, Revision ~~3~~~~b~~**4**, ~~July 2009~~***publication date***

E-2 **Oconee** Nuclear Design Methodology Using CASMO-~~3~~**4**/SIMULATE-3P, DPC-NE-~~1004-A~~**1006-P**, Revision ~~1a~~**0**, ~~January 2009~~**May 2009**

E-3 **SIMULATE-3KModels & Methodology,~~:~~ SSP-98/13, Revision 6, Studsvik Scandpower, Inc., January 2009** ~~**Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-95/18, Studsvik of America, October 1995**~~~~SIMULATE-3 Kinetics Theory and Model Description, SOA-96-26, Studsvik of America, April 1996~~

ATTACHMENT 12

DPC-NE-3005-A – UFSAR Chapter 15 Transient Analysis Methodology (Revision 3b) – Mark-Up

Duke Energy
Oconee Nuclear Station

UFSAR Chapter 15 Transient
Analysis Methodology

DPC-NE-3005
Revision ~~3b~~**4**

~~July 2009~~**October 2009**

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

Abstract

This report describes the Duke Energy methodology for simulating the UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station. The report includes details of the computer codes and models, methods for calculating safety analysis physics parameters and setpoints, and detailed modeling assumptions for all of the non-LOCA transients and accidents. The EPRI codes RETRAN-3D, VIPRE-01, and ARROTTA/1.10, and the Studsvik of America codes CASMO-3, SIMULATE-3P, and SIMULATE-3K are used for modeling the transient system and core thermal-hydraulic response, and the steady-state and transient core neutronic behavior. The dynamic reactor response is modeled using point, one-dimensional, and three-dimensional kinetics models, depending on the modeling requirements of each transient. This methodology will be used to reanalyze the Oconee UFSAR transients and accidents in order to establish an up-to-date design basis, and to support advanced fuel assembly and core reload designs.

Revision 1 describes methodology revisions that resulted from issues identified during the NRC review of Revision 0 and other minor changes. As described in the associated NRC Safety Evaluation, the NRC-related revisions pertain to the locked rotor, steam generator tube rupture, and steam line break accidents.

Revision 2 describes the RETRAN-3D methodology for the analysis of the replacement once-through steam generators, and other minor changes and corrections. With the NRC approval of Revision 2, RETRAN-3D became the approved system transient analysis methodology for the Oconee units using replacement steam generators (which is now all Oconee units). As part of Revision 2, Appendix A was added to the report to address in detail each of the 45 RETRAN-3D Safety Evaluation limitations and to describe Duke modifications to the RETRAN-3D code.

Revision 3b incorporates changes that were reviewed and approved by the NRC under methodology report DPC-NE-2015-PA, Oconee Nuclear Station Mark-B-HTP Transition Methodology. The primary technical change was the incorporation of the BHTP critical heat flux correlation and higher detail VIPRE-01 core thermal-hydraulic models. Revision 3b also includes minor updates and clarifications to the report.

Revision 4 describes the necessary changes to the methodology to model gadolinia cores. The major revisions are to replace CASMO-3 with CASMO-4, to update SIMULATE-3K to the most recent version, and to add the HTP gadolinia hot channel factor to the appropriate chapters. Revision 4 also includes editorial descriptions of gadolinia fuel to affected sections for clarification throughout the report.

UFSAR CHAPTER 15
TRANSIENT ANALYSIS METHODOLOGY

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Revision 3b was issued in July 2009. It includes changes approved through the NRC review of DPC-NE-2015-PA, Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology (Reference 1-38). The key technical changes include the BHTP critical heat flux correlation for use with Mark-B-HTP fuel design and a more detailed VIPRE-01 core thermal-hydraulic model for situations requiring more detail. In addition to the changes from DPC-NE-2015-PA, editorial changes and clarifications are included in Revision 3b.

Revision 4 was submitted in 2009 as part of the gadolinia implementation LAR (Reference 1-41) and documents the methodology changes as a result of implementing gadolinia fuel. The main changes are to replace the CASMO-3/SIMULATE-3 code package with CASMO-4/SIMULATE-3, to update SIMULATE-3K to the most recent version, to add the HTP gadolinia hot channel factor, and to make several editorial changes to support the switch to gadolinia fuel.

1.2 Computer Codes

Chapter 2 describes the computer codes and models used in the reanalysis of UFSAR Chapter 15 for Oconee. Each computer code is described along with the model and a summary of code and model validation. A brief summary of the codes and models is as follows:

<u>RETRAN-02</u>: The system thermal-hydraulic analysis formerly used the RETRAN-02/MOD5.2 code, which is an error-corrected version of the RETRAN-02/MOD5.1 code that has been reviewed generically by the NRC (Reference 1-16) and approved for use provided plant-specific methods have also been submitted for review. The NRC approval of Revision 2 of this report authorized the use of the RETRAN-3D computer code for system thermal-hydraulic analysis of Oconee units with replacement steam generators. All of the Oconee steam generators have now been replaced, so no future use of RETRAN-02 is planned. However, RETRAN-02 and RETRAN-3D provide very similar results for most analyses, and this report includes a large number of demonstration analyses that used RETRAN-02. Accordingly, some description of RETRAN-02 is retained in the report. Up-to-date analyses of specific design basis accidents are presented in Chapter 15 of the Oconee UFSAR.

Two RETRAN modeling applications not included in DPC-NE-3000 are included in this topical report. One is the use of one-dimensional (1-D) kinetics for modeling the core response during transients. This modeling approach will be used for some transients for which the point kinetics model does not provide sufficient results. RETRAN 1-D kinetics has been approved by the NRC for use in BWR applications, but it is our understanding that this may be the first submittal for PWR

CASMO-~~3~~4: Nuclear constants are generated with the Studsvik ~~of America~~**Scandpower** code CASMO-~~3~~**4** (Reference 1-21). This code is used in Oconee reload design (Reference 1-22), and was approved by the NRC in Reference 1-23. CASMO-~~3~~**4** is used for generating data used as input to the core models listed below.

SIMULATE-3P: Nuclear parameters and core power distributions are generated with the Studsvik of America code SIMULATE-3P (Reference 1-24). This code is used in Oconee reload design (Reference 1-22), and was approved by the NRC in Reference 1-23. SIMULATE-3P is also used in the McGuire and Catawba UFSAR Chapter 15 methodology.

ARROTTA/1.10: The EPRI code ARROTTA (Reference 1-25) was formerly used for transient three-dimensional (3-D) modeling of the rod ejection accident. Duke has transitioned to the use of the SIMULATE-3K code for transient 3-D core modeling, so ARROTTA is no longer used. However, ARROTTA results are shown in comparison to SIMULATE-3K and in some of the rod ejection demonstration analyses, so a limited description of the code is retained in this report.

SIMULATE-3K: The Studsvik of America code SIMULATE-3K (Reference 1-26) is now used for transient 3D modeling of the rod ejection accident. SIMULATE-3K provides the same neutronics solution to steady-state 3-D calculations as SIMULATE-3P. Duke intends to use SIMULATE-3K as an equivalent code for any of the steady-state applications in this report that are stated as being analyzed with SIMULATE-3P. Additional features include the time-dependent equations necessary to solve transient 3-D problems. This is the first submittal of this version of the SIMULATE family of codes for NRC approval.

1.3 Analysis Methodology

Chapter 3 describes the methods to calculate the safety analysis physics parameters which are generic to UFSAR Chapter 15 analyses. These inputs are calculated with the SIMULATE-3P code.

Chapter 4 describes the methodology for calculating the Reactor Protection System (RPS) and the Engineered Safeguards Protective System (ESPS) setpoints used in UFSAR Chapter 15 non-LOCA analyses. Determining initial conditions which incorporate allowances for parameter uncertainties is also discussed.

Chapters 5 through 16 describe the method of analysis for each UFSAR Chapter 15 non-LOCA transient and accident. Six of the transients and accidents are described in detail including a

<u>Gadolinia Fuel</u>

Gadolinia is used as an integral burnable absorber in the Mark-B-HTP fuel assemblies. Gadolinia bearing fuel rods comprise no more than 24 of the 208 fuel rods in the Mark-B-HTP assembly and the gadolinia weight percent is limited to less than or equal to 8%. Gadolinia suppresses pin powers at the beginning of cycle due to the high neutron absorption properties. The thermal properties of gadolinia also limit the transient heat flux out of the gadolinia bearing fuel rods thereby resulting in improved DNB and peak primary pressure performance. Specific modeling of gadolinia is described in Chapters 5 through 16 as appropriate.

1.4 Interface with Duke Oconee Reload Design Methodology Topical Report

This report is referenced by NFS-1001A, "Duke Energy Oconee Nuclear Station Reload Design Methodology" (Reference 1-2). Chapters 1 and 8 of NFS-1001A describe how the UFSAR Chapter 15 non-LOCA transient and accident analysis methodology of DPC-NE-3005 are integrated into the reload design process.

1.5 Appendix A

Appendix A was added in Revision 2 to address the RETRAN-3D SER conditions and limitations as relates to the modeling for Oconee.

1.6 Summary

The methodology presented in this report describes a conservative approach to performing the UFSAR Chapter 15 analyses for Oconee with modern thermal-hydraulic and nuclear analysis codes. These methods will be used to revise the existing UFSAR analyses which date to the early 1970s. The transient and accident analysis results presented are typical of those that will be used to update the UFSAR. Once implemented in the UFSAR, the revised analyses will enable a complete understanding of what the licensing basis analyses assume in terms of plant systems and component responses. This process will enhance the capability of Duke to review and assess plant operations and design in order to ensure compliance with regulations, to ensure consistency with Technical Specifications, and to ensure safe operation.

1.7 References

1-1 Letter, P. C. Wagner (NRC) to W. O. Parker, Jr. (Duke), July 29, 1981 (SER for NFS-1001)

1-2 Duke Energy Oconee Nuclear Station Reload Design Methodology~~,~~ NFS-1001-A, Revision ~~5~~7, Duke Energy, ~~January 2001~~***Publication date***

1-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

1-4 Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (Generic Letter No. 83-11), NRC, February 8, 1983

1-5 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 6.1, EPRI, June 2007

1-6 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM, Revision ~~3~~**4**, EPRI, ~~August 1989~~**February 2009**

1-7 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), November 15, 1991 (SER for DPC-NE-3000 - MNS/CNS sections)

1-8 Letter, L. A. Wiens (NRC) to M. S. Tuckman (Duke), August 8, 1994 (SER for DPC-NE-3000 - ONS sections)

1-9 Letter, R. E. Martin (NRC) to M. S. Tuckman (Duke), December 27, 1995 (SER for DPC-NE-3000, Rev. 1)

1-10 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001-PA, **Rev. 0a,** Duke Energy, ~~December 2000~~**May 2009**

1-11 FSAR Chapter 15 System Transient Analysis Methodology, DPC-NE-3002-A, Revision 4, Duke Energy, May 2005

1-12 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), November 15, 1991 (SER for DPC-NE-3001, Rev. 0)

1-13 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), November 15, 1991 (SER for DPC-NE-3002, Rev. 0)

1-14 Letter, R. E. Martin (NRC) to M. S. Tuckman (Duke), December 28, 1995 (SER for DPC-NE-3002, Rev. 1)

1-15 Letter, H. N. Berkow (NRC) to M. S. Tuckman (Duke), April 26, 1996 (SER for DPC-NE-3002 regarding safety valve opening characteristics)

1-16 Letter, M. J. Virgilio (NRC) to C. R. Lehman (PP&L), April 12, 1994 (SER for RETRAN-02/MOD 5.1)

1-17 Letter, L. A. Wiens (NRC) to M. S. Tuckman (Duke), March 15, 1995 (SER for DPC-NE-3003)

1-18 Mass and Energy Release and Containment Response Methodology, DPC-NE-3003-PA, Revision 1, Duke Energy, September 2004

1-19 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

1-20 Letter, C. E. Rossi (NRC) to J. A. Blaisdell (UGRA), May 1, 1986 (VIPRE-01 SER)

1-21 CASMO-~~34:~~, A Fuel Assembly Burnup Program ~~User's Manual~~**Methodology"**, ~~Version 4.7, Revision 3, STUDSVIK/NFA-89/3, Studsvik of America, June 1993~~**Proprietary, SOA-95/2, STUDSVIK of America, Inc., USA, STUDSVIK Core Analysis AB, Sweden, September 1995**

1-22 **Oconee** Nuclear Design Methodology Using CASMO-~~3~~**4** / SIMULATE-3~~P~~, DPC-NE-~~1004A~~**1006-P**, Revision ~~1~~**0**, Duke Energy, ~~April 26, 1996~~**May 2009**

1-23 ~~Letter, A. C. Thadani (NRC) to H. B. Tucker (Duke), November 23, 1992 (SER for DPC-NE-1004)~~**Include reference to NRC SER for DPC-NE-1006 when issued.**

1-24 SIMULATE-3 Methodology: Advanced Three-Dimensional Two-Group Reactor Analysis Code, STUDSVIK/SOA-95/18, Studsvik of America, October 1995

1-25 ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI, August 1993

1-26 **SIMULATE-3K Models & Methodology, SSP-98/13, Revision 6, Studsvik Scandpower, Inc., January 2009**~~SIMULATE-3 Kinetics Theory and Model Description, SOA-96/26, Studsvik of America, April 1996~~

1-27 Letter, W. O. Parker, Jr. (Duke), to Robert W. Reid (NRC), March 21, 1981

1-28 Letter, Robert W. Reid (NRC,) to all B&W Licensees, January 14, 1981

1-30 Letter, Philip C. Wagner (NRC), to W. O. Parker, Jr. (Duke), April 21, 1982

1-30 Letter, M. S. Tuckman (Duke) to NRC Document Control Desk, August 9, 1994 (Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Revision 1)

1-31 Letter, M. S. Tuckman (Duke) to NRC Document Control Desk, December 23, 1997 (Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Revision 2)

1-33 Letter, D. E. LeBarge (NRC) to W. R. McCollum (Duke), October 14, 1998 (SER for DPC-NE-3000, Rev. 2)

1-33 Letter, M. S. Tuckman (Duke) to NRC, February 1, 1999 (Letter submitting Revision 1 to DPC-NE-3005)

1-34 Letter, D. E. LaBarge (NRC) to W. R. McCollum (Duke), October 1, 1998 (NRC SER on DPC-NE-3005, Revision 0)

1-35 Letter, M. S. Tuckman (Duke) to NRC, June 13, 2002 (Letter submitting Revision 2 to DPC-NE-3005 and Revision 3 to DPC-NE-3000)

1-36 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

1-37 Letter, L. N. Olshan (NRC) to R. A. Jones (Duke), September 24, 2003 (NRC SER on DPC-NE-3005, Revision 2 and DPC-NE-3000, Revision 3)

1-38 Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology, DPC-NE-2015-PA, Duke Energy Carolinas, November 2008

1-39 Letter, P. S. Tam (NRC) to G. R. Peterson (Duke), February 5, 1999 (NRC SER on DPC-NE-3002, Revision 2)

1-40 Letter, C. P. Patel (NRC) to G. R. Peterson (Duke), April 6, 2001 (NRC SER on DPC-NE-3002, Revision 4)

1-41 Letter documenting the gadolinia fuel LAR (the exact reference will be updated at a later date).

generates conservative minimum DNBR and local thermal-hydraulic conditions for both steady-state and transient analyses.

2.4 CASMO-3

CASMO-3 is a multigroup, two dimensional transport theory code for burnup calculations on PWR or BWR fuel assemblies. The code models a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with integral burnable absorber, lumped burnable absorber rods, clustered discrete control rods, incore instrument channels, assembly guide tubes, and intra-assembly water gaps. The program utilizes a 40 energy group cross section library based on ENDF/B-IV with some data taken from ENDF/B-V. Two energy group edits of cross sections, assembly discontinuity factors, fission product data, and pin power data are produced for input to the SIMULATE-3P and SIMULATE-3K core models. Reference 2-14 provides a detailed description of the theory and equations solved by CASMO-3. The use of CASMO-3 in this report is consistent with the previously approved methodologies of References 2-10 and 2-15.

CASMO-3 is replaced with CASMO-4 (Section 2.8) and is no longer used by Duke. However, since CASMO-3 is used to generate the cross sections input to ARROTTA and SIMULATE-3K, which are used in the rod ejection demonstration analyses, the description above is retained.

2.5 SIMULATE-3P

SIMULATE-3P is a three-dimensional, two energy group, diffusion theory core simulator program which explicitly models the baffle and reflector regions of the reactor. Homogenized cross sections and discontinuity factors developed with CASMO-3 are used on a coarse mesh nodal basis to solve the two group diffusion equations using the QPANDA neutronics model. A nodal thermal hydraulics model is incorporated to provide both fuel and moderator temperature feedback effects. Inter- and intra-assembly information from the coarse mesh solution is then utilized along with the pinwise assembly lattice data from CASMO-3 to reconstitute pin-by-pin power distributions in two and three dimensions. The program performs a macroscopic depletion of fuel with microscopic depletion of iodine, xenon, promethium, and samarium fission products. Reference 2-6 provides a detailed description of the theory and equations solved by SIMULATE-3P. The use of SIMULATE-3P in this report is consistent with the previously approved methodologies of References 2-10 and 2-15.

2.6 ARROTTA/1.10

ARROTTA is a three-dimensional, two energy group diffusion theory core simulator applicable for both static and transient kinetics simulations. Homogenized cross sections, discontinuity factors, and 6 groups of delayed neutron precursor data are generated with CASMO-3 and used on a coarse mesh nodal basis to solve the two energy group diffusion equations using the QPANDA neutronics model. The thermal-hydraulic model is comprised of both fluid dynamics and heat transfer models. Reference 2-16 provides a detailed description of the theory and equations solved by ARROTTA. The use of ARROTTA in this report is consistent with the previously approved methodology documented in Reference 2-10.

ARROTTA is no longer used for Oconee UFSAR Chapter 15 analyses. However, ARROTTA was used to validate the use of SIMULATE-3K, and some of the rod ejection demonstration analyses are based on ARROTTA work. Accordingly, discussion of ARROTTA is retained in this report.

2.7 SIMULATE-3K

2.7.1 Code Description

The SIMULATE-3K code**, Version 2** (Reference 2-5) is used to perform three-dimensional transient core neutronic modeling during postulated rod ejection accidents. SIMULATE-3K is a three-dimensional transient neutronic version of the SIMULATE-3P code. SIMULATE-3K uses the QPANDA full two-group nodal spatial model developed in SIMULATE-3P, with the addition of six delayed neutron groups. The program employs a fully-implicit time integration of the neutron flux, delayed neutron precursor, and heat conduction models. **Subcritical neutron sources may be modeled. Decay heat is based on the ANSI/ANS-5.1-1994 standard.** A calculation of the adjoint flux solution is performed to provide an accurate value of beta for the time-varying neutron flux. The control of time step size may be determined either as an automated feature of the program or by user input. Use of the automated feature allows the program to utilize larger time steps (which may be restricted to a maximum size based on user input) at times when the neutronics are changing slowly and smaller time steps when the neutronics are changing rapidly.

Additional capability is provided in the form of modeling a reactor trip. The trip may be initiated at a specific time in the transient or following a specified excore detector response. Use of the excore detector response model to initiate the trip allows the user to specify the response of individual detectors as required to initiate the trip, as well as the time delay prior to release of the control rods. **The model also allows for excore detector signal changes as the coolant density changes.** The velocity of the control rod movement is also controlled by user input.

The SIMULATE-3K thermal-hydraulic model may include a spatial heat conduction **model** and a **5-equation** hydraulic channel model. The heat conduction model solves the conduction equation on a multi-region mesh in cylindrical coordinates. Temperature-dependent values may be employed for the heat capacity, thermal conductivity, and gap conductances. **Burnup-dependent models may be employed for thermal conductivity, gap conductance, and the pellet radial power profile.** A single characteristic pin conduction calculation is performed ~~per fuel assembly~~**consistent with the neutronic node geometry**, with an optional calculation of the peak pin behavior available to monitor local maxima. A single characteristic hydraulic channel calculation is performed per fuel assembly. The model allows for direct moderator heating at the option of the user. This thermal-hydraulic model is used to determine fuel and moderator temperatures for updating the cross-section model, and may additionally be used to provide edits of fuel temperature **and enthalpy** throughout the transient.

The SIMULATE-3K program utilizes the same **CASMO-4 (Reference 2-24)** cross-section library and reads the same restart file (exposure and burnup-related information) as SIMULATE-3P. Executed in the static mode, SIMULATE-3K performs the same solution techniques, pin power reconstruction, and cross-section development as SIMULATE-3P. Additional features of SIMULATE-3K include the application of conservatisms through simple user input. Also, the inlet thermal-hydraulic conditions can be provided on a time dependent basis through user input.

2.7.2 Code Validation

Several benchmarks **against many numerical steady state and transient benchmark problems** were performed by the code vendor ~~(Studsvik of America, Inc.) during development of SIMULATE-3K~~. **An assessment of SIMULATE-3K for light water reactor reactivity initiated transients is described in Reference 2-27 and show excellent agreement between SIMULATE-3K and the reference solutions.**~~These benchmarks and results are described in the SIMULATE-3K manual (Reference 2-5).~~ The fuel conduction and thermal-hydraulics model have been benchmarked against the TRAC code (Reference 2-17) **as described in Reference 2-28**. The transient neutronics model has been benchmarked, using standard LWR problems, to reference solutions generated by QUANDRY (Reference 2-18), ~~SPANDEX (Reference 2-19), NEM (Reference 2-20),~~ and CUBBOX (Reference 2-21). Finally, a benchmark of the coupled performance of the transient neutronics and thermal-hydraulic models was provided by comparison ~~of results from a standard NEACRP rod ejection problem~~ to the PANTHER code (Reference 2-22). Steady-state components of the

SIMULATE-3K model are implemented consistent with the CASMO-~~3~~4/SIMULATE-3P methodology and performance benchmarks ~~which were approved for use on all Duke reactors in November 1992 (Reference 2-15)~~**documented in Reference 2-26.**

2.8 CASMO-4

CASMO-4 is a multi-group, two dimensional transport theory code for burnup calculations on fuel assemblies or fuel pin cells as described in References 2-24 and 2-25. The code accommodates a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array. CASMO-4 can model fuel pins, lumped and burnable absorber, lumped burnable absorbers, control rods, guide tubes, incore instruments, water gaps, and reflectors. The nuclear data library input to CASMO-4 is based mainly on data from ENDF/B-IV. It contains cross sections from more than 100 materials commonly found in light water reactors. The cross sections are collected into 70 energy groups covering neutron energies from 0 to 10 million electron volts (MeV). CASMO-4 incorporates microscopic depletion of burnable absorbers into the main calculation, uses a geometrically heterogeneous model for the entire calculation, and uses the" method of characteristics" for solving the transport equation, all of which allow for the accurate modeling of LEU and fuel containing integral or lumped burnable absorbers. Two energy group edits of cross sections, assembly discontinuity factors, fission product data, and pin power data are produced for input to the SIMULATE-3 and SIMULATE-3K core models. The use of CASMO-4 in this report is consistent with the methodology described in Reference 2-26.

2.~~8~~9 References

2-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 6.1, EPRI, June 2007

2-2 Letter, M. J. Virgilio (NRC) to C. R. Lehman (PP&L), April 12, 1994 (SER for RETRAN-02/MOD 5.1)

2-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

2-4 Letter, L. A. Wiens (NRC) to M. S. Tuckman (Duke), August 8, 1994 (SER for DPC-NE-3000 - ONS sections)

2-5 SIMULATE-3-K~~inetics Theory And Model Description~~**Models & Methodology**, ~~SOA~~**SSP-**~~96/26~~**98/13**, **Revision 6,** ~~Studsvik of America,~~ **Studsvik Scandpower, Inc.,** ~~April 1996~~**January 2009**

2-6 SIMULATE-3 Methodology: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-95/18, Studsvik of America, October 1995

2-7 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

2-9 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores; EPRI NP-2511-CCM, Revision ~~3~~**4**, EPRI, ~~August 1989~~**February 2001**

2-9 Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA), "Acceptance for Referencing of Licensing Topical Report, VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," EPRI-NP-2511-CCM, Vol. 1-5, May 1986

2-10 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001-PA, **Rev. 0a,** Duke Energy, ~~December 2000~~**May 2009**

2-11 COBRA IIIC/MIT Computer Code Manual, MIT, March 1976

2-12 COBRA IIIC: A Digital Computer Program for Steady-State and Transient Core Thermal-Hydraulic Design Methodology, BNWL-1695, PNL, March 1973

2-13 Letter from Hal B. Tucker of Duke Energy to U. S. Nuclear Regulatory Commission, Topical Report DPC-NE-3000 Response to Request for additional Information, February 20, 1990

2-14 CASMO-3: A Fuel Assembly Burnup Program User's Manual, Version 4.7, Revision 3, STUDSVIK/NFA-89/3, Studsvik of America, June 1993

2-15 Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P, DPC-NE-1004-A, Revision 1**a**, Duke Energy, ~~April 26, 1996~~**January 2009**

2-16 ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI, August 1993

2-17 TRAC-BF1/MOD1, Vol. 1, NUREG/CR-4356, J. Borkowski and N. Wade (Editors), May 1992

2-18 QUANDRY: An Analytical Nodal Method for Solving the Two-Group, Multidimensional, Static and Transient Nodal Diffusion Equations, K. S. Smith, Massachusetts Institute of Technology, 1979

2-19 ~~Development of a Variable Time-Step Transient NEM Code: SPANDEX, Trans. Am. Nucl. Soc. 68-425, B. N. Aviles, 1993~~**deleted**

2-20 ~~NEM: A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code, B. R. Bandini, Pennsylvania State University, 1990~~**deleted**

2-21 CUBBOX: Coarse-Mesh Nodal Diffusion Method for the Analysis of Space-Time Effects In Large Light Water Reactors, S. Langenbuch, W. Maurer, and W. Werner, Nucl. Sci. Eng. 63-437, 1977

2-24 NEACRP 3-D LWR Core Transient Benchmark, NEACRP-L-335, H. Finneman and A. Galati, January 1992

2-25 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

2-24 CASMO-4, A Fuel Assembly Burnup Program Methodology, Proprietary, SOA-95/2, STUDSVIK of America, Inc., USA, STUDSVIK Core Analysis AB, Sweden, September 1995

2-25 CASMO-4, A Fuel Assembly Burnup Program, User's Manual, Proprietary, SSP-01/400, Rev.3, Studsvik Scandpower, July 2003

2-26 Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3P, DPC-NE-1006-P, Revision 0, May 2009

2-27 LWR Core Reactivity Initiated Transients with SIMULATE-3: Models and Assessment, SSP-04/443, Revision 2, Studsvik Scandpower

2-28 SIMULATE-3 Kinetics Theory And Model Description, SOA-96/26, Studsvik of America, April 1996

- Control rods
- Moderator temperature

The MTC is calculated by inducing a change in moderator temperature (and therefore, density) and dividing the resulting reactivity change by the change in moderator temperature.

Doppler Temperature Coefficient

The Doppler temperature coefficient is defined as the change in core reactivity resulting from a change in fuel temperature. The least and most negative Doppler temperature coefficients are calculated for each reload core considering the core burnup and power level. The Doppler temperature coefficient is calculated by performing a set of two cases which vary the fuel temperature. The reactivity difference between the two fuel temperatures divided by the change in fuel temperature is the definition of the Doppler temperature coefficient. Doppler temperature coefficients are often quoted at various power levels by equating changes in reactor power to changes in mean fuel temperature.

Critical Boron Concentrations and Boron Worths

Critical and shutdown boron concentrations are calculated as a function of reactor power, exposure, temperature, and control rod positions as allowed by the power dependent rod insertion limits. Differential boron worths are also calculated as a function of various combinations of the above variables. The results of these calculations are compared to inputs for several accident analyses.

3.4 Reload Cycle Specific Evaluation

The important physics parameters in Table 3-1 will be evaluated each reload cycle to ensure that values assumed in the current licensing analyses bound the reload core. Accidents for which the physics parameters are not bounded would be re-evaluated to ensure acceptable accident consequences or the core would be redesigned to obtain acceptable results.

3.5 References

3-1 **Oconee** Nuclear Design Methodology Using CASMO-~~3~~4/SIMULATE-3P, DPC-NE-~~1004A~~**1006-P**, Revision ~~1~~0, Duke Energy, ~~April 26, 1996~~**May 2009**.

5.4 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram curve worth
- Maximum reactivity insertion rate

5.5 References

5-1 Deleted

5-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~**4**, EPRI, ~~October 1989~~**February 2001**

5-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***.

5-4 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

6.3 VIPRE-01 Analysis

The forcing functions necessary to perform the DNB analysis (core average heat flux, core inlet flow and temperature, core exit pressure) are obtained from the RETRAN analysis results and input to VIPRE-01. The VIPRE-01 [ ] channel model (Reference 6-4) is then used to determine the time of the minimum DNBR statepoint for the transient conditions analyzed. At these statepoint conditions a set of maximum allowable radial peak (MARP) curves is developed for determining if the DNBR limit is exceeded.

6.4 Results

The peak primary pressure reached in the limiting case is approximately 2600 psig. This is well below the acceptance criterion of 2750 psig. The results of the DNBR analysis have demonstrated that the power peaking predicted by SIMULATE-3P will remain below the DNBR limits.

6.5 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve
- Minimum and maximum reactivity insertion rates

6.6 References

6-1 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

6-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~4, EPRI, ~~August 1989~~**February 2001**

6-3 SIMULATE-3 Methodology: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-95/18, Studsvik of America, October 1995

6-4 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***.

8.7 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve

8.8 References

8-1 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

8-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

9.4 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve

9.5 References

9-1 RETRAN-02: A Program for transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 6.1, EPRI, June 2007

9-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a,~~**5** Duke Energy, ~~July 2009~~***Publication date***

9-3 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~**4**, ~~August 1989~~**February 2001**

9-4 BWC Correlation of Critical Heat Flux, BAW-10143-PA, Babcock & Wilcox, April 1985

9-5 The BWU Critical Heat Flux Correlations, BAW-10199P-A, Framatome Cogema Fuels, April 1996

9-6 DPC-NE-~~1004A~~**1006-P**, ~~Duke Energy~~**Oconee** Nuclear Design Methodology Using CASMO-~~3~~**4**/SIMULATE-3P, ~~November 1992~~**May 2009**

9-7 DPC-NE-2003-P-A, Revision ~~2a~~**3**, Core Thermal-Hydraulic Methodology Using VIPRE-01, ~~December 2008~~***Publication date***

9-8 BHTP DNB Correlation Applied with LYNXT, BAW-10241(P)(A), Revision 1, Framatome ANP, July 2005

9-9 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

10.3.2.4 Critical Heat Flux Correlation

The CHF correlation used depends on the fuel type being analyzed, as discussed in Section 10.2.2. The critical heat flux (CHF) correlation utilized for the demonstration locked rotor transient DNBR calculation results presented herein is the BWC CHF correlation (Reference 10-4). The BWC CHF correlation SDL for the locked rotor transient will be determined utilizing the minimum DNBR state point boundary conditions described in Section 10.3.2.7.

10.3.2.5 Fuel Conduction Model

The VIPRE fuel conduction model is utilized in the [ ] channel model. Sensitivity studies have shown that a [ ] results in a conservative transient DNBR value. Thus, the [ ] is used for the analysis. The initial gap ~~conductivity~~ **conductance** is determined by [ ] to ~~the predicted fuel temperature from TACO-3 (Reference 10-7) for different power levels~~**a bounding core average fuel temperature**. Since there is [ ] during the transient. **The gadolinia rods are modeled as if they were UO_2 rods as described above.**

10.3.2.6 Heat Transfer Correlations

For the DNBR calculations, only the single-phase forced convection and nucleate boiling heat transfer modes are applicable. The [ ] is used for the single-phase forced convection mode. The [ ] is used for the nucleate boiling region. The critical heat flux correlation used to define the peak of the boiling curve is the same as that used to predict the DNBR.

- Doppler temperature coefficient
- Minimum scram worth curve

10.5 References

10-1 RETRAN-02: A Program for transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 6.1, EPRI, June 2007

10-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date.***

10-3 VIPRE-01 : A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~**4**, ~~August 1989~~**February 2001**

10-4 BWC Correlation of Critical Heat Flux, BAW-10143-PA, Babcock & Wilcox, April 1985

10-5 The BWU Critical Heat Flux Correlations, BAW-10199P-A, Framatome Cogema Fuels, April 1996

10-6 **Oconee** Nuclear Design Methodology Using CASMO-~~3~~**4**/SIMULATE-3P, DPC-NE-~~1004A~~**1006-P**, Revision ~~1~~**0**, Duke Energy, ~~April 26, 1996~~**May 2009**

10-7 ~~TACO-3, Fuel Pin Thermal Analysis Code, BAW-10162P-A, BWFC, November 1989~~**Deleted**

10-8 BHTP DNB Correlation Applied with LYNXT, BAW-10241(P)(A), Revision 1, Framatome ANP, July 2005

10-9 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

11.3 References

11-1 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

11-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~**4**, EPRI, ~~August 1989~~**February 2001**

11-3 SIMULATE-3 Methodology: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-95/18, Studsvik of America, October 1995

11-4 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

11-5 O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," NUREG/CR-0200, Volume 1, Section S2, March 2000

11-6 O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-0200, Volume 2, Section F7, March 2000

11-7 Judith F. Briesmeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code," Los Alamos National Laboratory Report, LA-13709-M, March 2000

12.5 References

12-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-1850-CCM, Revision 6.1, EPRI, June 2007

12-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

12-3 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450 (A), Volumes 1-4, Revision 6.3, EPRI, July 2007

13.6 Results

The thermal-hydraulic response resulting from this event is provided as input to a separate analysis which determines the fission product release to the environment.

13.7 Reload Cycle-Specific Evaluation

The reload physics parameter that must be checked is a minimum boron worth.

13.8 References

13-1 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

14.2.4.1 Peak Pellet Enthalpy

To show that the peak fuel enthalpy acceptance criterion is met, a [ ] channel VIPRE model with fuel conduction is used to calculate the maximum hot spot fuel temperature during the transient. Given the SIMULATE-3K predicted [,] VIPRE calculates the transient maximum hot spot average fuel temperature. This fuel temperature is then used in the calculation of the maximum radial average fuel enthalpy. Details regarding the [ ] channel VIPRE model and initial and boundary conditions follow.

Model Description

A [ ] channel model with fuel conduction is constructed to simulate the peak fuel pin in the hot assembly during the transient. Sensitivity studies have shown that a [ ] channel model and the [ ] channel model yield nearly identical fuel temperature results.

Power Distributions

During the transient, the hot assembly axial power distributions change mainly due to the ejected control rod and to the insertion of control rods as the reactor trips. [ ]

Fuel Conduction Model

The VIPRE fuel conduction model is used. A [ ] results in conservative fuel temperatures and is used in the fuel temperature calculations. During the transient, core average power will increase as will the hot fuel assembly power. Above a certain power level, [ ] **Furthermore, the same thermal properties that result in a reduced heat flux for gadolinia rods also result in higher fuel temperatures. Therefore, gadolinia properties are explicitly modeled for the gadolinia fuel rods.**

Heat Transfer Correlations

Heat transfer correlations used for the four major segments of the boiling curve are as shown below.

Single-phase forced convection: [ ]
Saturated nucleate boiling regime: [ ]
Transition boiling regime: [ ]
Film boiling regime: [ ]

The critical heat flux correlations used to define the peak of the boiling curve and to predict the DNBR are the BWU CHF correlation (Reference 14-10) for Mark-B11 fuel, and the BHTP CHF correlation (Reference 14-11) and the BWU-N CHF correlation (Reference 14-10) for the Mark-B-HTP fuel.

Flow Correlations

For the rod ejection analysis, the subcooled void, the bulk void, and the two-phase friction multiplier are modeled by using the [ ] correlations, respectively.

Conservative Factors

An engineering hot channel factor (Fq) of [ ] is applied to the hot rod **if the hot rod is UO_2 or 1.0145 if the hot rod contains gadolinia**. This factor accounts for the increase in rod power due to manufacturing tolerances such as differences in the number of U-235 grams per rod, loading tolerances of U-235 per stack, and variations on the powder lot mean enrichment.

The hot subchannel flow area is reduced to account for variations in the as-built dimensions of the subchannel. The following flow area reductions are assumed.

Unit channel: [ ] flow area reduction
Thimble channel: [ ] flow area reduction

A core inlet flow maldistribution penalty is assumed depending upon the pump configuration. This inlet flow reduction is applied to the hot assembly and the [ ] channel model and is as follows:

4 reactor coolant pumps: 5% inlet flow reduction
3 reactor coolant pumps: [ ] inlet flow reduction
2 reactor coolant pumps: [ ] inlet flow reduction

(continued from Section 14.2.4.2 in DPC-NE-3005, Fuel Conduction Model)

]The gadolinia rods are modeled as if they were UO_2 rods as described above.

Heat Transfer Correlations

For the DNBR calculations, only the single-phase forced convection and nucleate boiling heat transfer modes are applicable. The [ ] is used for the single-phase forced convection mode. The [ ] is used for the nucleate boiling regime. The critical heat flux correlation used to define the peak of the boiling curve and to predict the DNBR are the BWU CHF correlation (Reference 14-10) for Mark-B11 fuel, and the BHTP CHF correlation (Reference 14-11) and the BWU-N correlation (Reference 14-10) for the Mark-B-HTP fuel.

Flow Correlations

The [

] for the two-phase friction multiplier.

Other Thermal-Hydraulic Correlations

In addition to those correlations discussed in Section 14.2.4.1, turbulent mixing is also calculated by VIPRE for the flow and energy solutions in the DNBR calculations. The single phase mixing correlation used is shown below and is used for both single and two-phase mixing.

$$w' = AS\overline{G}$$

where: A = turbulent mixing coefficient for fuel type (see Reference 14-12)

S = gap width, feet

$\overline{G}$ = average mass velocity in the channels connected by gap K, lbm/sec-ft^2

(continued from Section 14.2.4.3 in DPC-NE-3005)

Fuel Conduction Model

A [ ] is used to provide for a conservative calculation of the channel local fluid conditions. [

]

Heat Transfer Correlations

For conservatism, only the first two segments of the boiling curve are used in the coolant expansion rate calculation. This forces VIPRE to use the [ ] for post-DNB conditions. This results in a conservatively large heat flux post-DNB and thus a conservatively high coolant expansion rate.

Single-phase forced convection: [ ]
Saturated nucleate boiling regime: [ ]

The critical heat flux correlations used to define the peak of the boiling curve are the BWU CHF correlation (Reference 14-10) for Mark-B11 fuel, and the BHTP CHF correlation (Reference 14-11) and the BWU-N correlation (Reference 14-10) for Mark-B-HTP fuel.

Flow Correlations

For the coolant expansion rate calculations, the subcooled void, the bulk void, and the two-phase friction multiplier are modeled by using the [

]

Other Thermal-Hydraulic Correlations

[.]

14.3 Nuclear Analysis

The response of the reactor core to the rapid reactivity insertion from a control rod ejection is simulated with SIMULATE-3K. Model geometry is typically four nodes per fuel assembly in the radial direction. The radial and axial nodalization depends on the fuel assembly design, such as whether or not axial blanket fuel is being modeled. For the analysis presented, a typical axial nodalization of 23 equal length fuel nodes in the axial direction is used. Beyond the fuel nodes, SIMULATE-3K explicitly calculates neutron leakage from the core by use of reflector nodes in the radial direction beyond the fuel region and in the axial direction above and below the fuel column stack.

SIMULATE-3K is used to calculate the core power level and pinwise power distribution versus time during the rod ejection transient. This information is used by VIPRE to determine the fuel enthalpy, the percentage of the fuel pins exceeding the DNB limit, and the coolant expansion rate.

14.3.1 SIMULATE-3K Analysis

The control rod ejection transient is analyzed for the following six initial conditions.

* BOC at maximum allowable core power level with 4 RCPs operating
* BOC at maximum allowable core power level with 3 RCPs operating
* BOC at HZP (two RCPs in the analysis presented, three in the future)

* EOC at maximum allowable core power level with 4 RCPs operating
* EOC at maximum allowable core power level with 3 RCPs operating
* EOC at HZP (two RCPs in the analysis presented, three in the future)

Because of the modifications to the SIMULATE-3K model which are described below, the analyses performed are expected to bound ~~all~~ future Oconee reload core designs.

Ejected Rod Location and Velocity

The core location of the ejected control rod is chosen specifically for each analysis to produce the most conservative results. Figure 14-1 shows the core configuration and location of the ejected control rod for the various analyses. For both HZP transients the control rod in core location L-10 is ejected from a fully inserted position in 0.15 seconds. The time required for rod ejection is consistent with the original FSAR analysis. It assumes that the pressure barrier has failed in such a way that it no longer offers any restriction to the ejection and that there is no viscous drag force to shut down the reactor. [

.]

Reactor Trip and Single Failure

The reactor trip signal is generated for all six transients when three of four excore detectors exceed the high flux **or flux/flow** trip setpoint used during cycle operation. This conservative modeling assumes that the detector which would indicate the highest flux level has failed and that a two out of the remaining three logic is required to generate a trip signal. The excore signals are synthesized from a ~~conservative~~ **radially weighted** combination of power densities ~~of several assemblies in the proximity of each excore detector~~**from every assembly in the core. Additionally, the high RCS pressure trip is credited.** A conservative trip delay time is assumed **for each of the trip signals.**

Scram Curve and Worth

The reactor scram assumes that the ejected rod, part length axial power shaping rods, and the remaining rod with the highest worth do not fall into the core. **The remaining control rods (reduced by an appropriate uncertainty) drop into the core at a speed that satisfies the maximum rod drop time in the technical specifications.**~~The remaining scrammed rod worth is reduced to ensure that only the minimum net shutdown margin is achieved. Additional conservatism is applied by limiting the rate at which the scrammed rods are allowed to fall into the core such that the reactivity inserted is bounded by the limiting curves.~~

The total effect of all these conservatisms is to create a SIMULATE-3K rod ejection model that will provide results which bound ~~all~~ future reload cycles. Table 14-1 identifies the conservative core parameters used in each analysis.

14.3.2 Deleted

14-2 ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI, August 1993

14-3 **SIMULATE-3K Models & Methodology, SSP-98/13, Revision 6, Studsvik Scandpower, Inc., January 2009**~~SIMULATE-3 Kinetics Theory and Model Description, SOA-96/26, Studsvik of America, April 1996~~

14-4 SIMULATE-3 Methodology: Advanced Three-Dimensional Two-Group Reactor Analysis Code, STUDSVIK/SOA-95/18, Studsvik of America, October 1995

14-5 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM, Revision 4, EPRI, February 2001

14-6 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 6.1, EPRI, June 2007

14-7 BWC Correlation of Critical Heat Flux, BAW-10143P-A, Babcock & Wilcox, April 1985

14-8 Donald L. Hagrmann, et al., MATPRO-Version 11 (Revision 2): A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, NUREG/CR-0479, August 1981

14-9 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

14-10 The BWU Critical Heat Flux Correlations, BAW-10199P-A, Framatome Cogema Fuels, April 1996 (and including Addendum 1, December 2000)

14-11 BHTP DNB Correlation Applied with LYNXT, BAW-10241(P)(A), Revision 1, Framatome ANP, July 2005

14-12 Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005-PA, Revision 4a, December 2008

14-13 RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

15.3.1.2.4 Conservative Factors

Conservative factors described in Reference 15.2 are applied to the [ ] channel VIPRE-01 model. These conservative factors are the hot channel area reduction factors (2% for the hot unit subchannel and 3% for the hot instrumentation subchannel), the engineering hot channel factor (F_q) of 1.013 **for non-gadolinia-bearing rods or 1.0145 for rods containing gadolinia**, and the core inlet flow maldistribution factor. Based on the vessel model flow test and Oconee core pressure drop measurement, the core inlet flow maldistribution is conservatively modeled as a reduction in the hot assembly flow. The hot assembly flow reduction factor for the VIPRE-01 DNB analysis is [ ].

15.3.1.2.5 Critical Heat Flux Correlation

The W-3S CHF correlation is used for the with offsite power steam line break DNBR analysis for Mk-B10 fuel. The historical range of applicability for the W-3S correlation is (Reference 15-3):

Pressure (psia)	1000 to 2300
Mass flux (10^6 lbm/hr-ft^2)	1.0 to 5.0
Quality (equilibrium)	-0.15 to 0.15

The W-3S CHF correlation has been approved by the NRC for analysis with system pressures as low as 500 psia and mass flux as low as 0.5 x 10^6 lbm/hr-ft^2 (References 15-9 and 15-10).

The BWU-Z CHF correlation with the 0.98 Mk-B11 multiplier is used for the with offsite power steam line break DNBR analysis for Mk-B11 fuel. The minimum DNBR design limits are as follows for the following parameter ranges of applicability for this correlation and design limit:

		Design DNBR
Pressure (psia)	315 to 700	1.59
	700 to 1000	1.20
	1000 to 2465	1.19
Mass flux (10^6 lbm/hr-ft^2)	0.36 to 3.55	
Quality	less than 0.74	

15.5 References

15-1 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 6.1, EPRI, June 2007

15-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a~~**5**, Duke Energy, ~~July 2009~~***Publication date***

15-3 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~**4**, EPRI, ~~August 1989~~**February 2001**

15-4 **Oconee** Nuclear Design Methodology Using CASMO-~~3~~**4**/SIMULATE-3P, DPC-NE-~~1004A~~**1006-P**, Revision ~~1~~**0**, Duke Energy, ~~April 26, 1996~~**May 2009**

15-5 Mass and Energy Release and Containment Response Methodology, DPC-NE-3003-PA, Revision 1, Duke Energy, September 2004

15-6 BWC Correlation of Critical Heat Flux, BAW-10143P-A, April 1985

15-7 BAW-10199-PA, The BWU Critical Heat Flux Correlations, Addendum 1, ~~September 1996~~**April 2000**

15-8 Letter, D. E. LaBarge (NRC) to W. R. McCollum (Duke), SER on topical report DPC-NE-3000-PA, Revision 2, October 14, 1998

15-9 Letter, A. S. Thadani (NRC) to W. J. Johnson (Westinghouse), SER on WCAP-9226-P, Reactor Core Response to Excessive Secondary Steam Releases", January 31, 1989

15-10 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), SER on topical report DPC-NE-3001, November 15, 1991

15-11 RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-7450(A), Volumes 1-4, Revision 6.3, EPRI, July 2007

16.5 References

16-1 RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-7450(A), Revision 6.3, EPRI, July 2007

16-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision ~~3~~**4**, EPRI, ~~August 1989~~**February 2001**

16-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision ~~4a,~~**5** Duke Energy, ~~July 2009~~***Publication date***

16-4 O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," NUREG/CR-0200, Volume 1, Section S2, March 2000

16-5 O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-0200, Volume 2, Section F7, March 2000

16-6 Judith F. Briesmeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code," Los Alamos National Laboratory Report, LA-13709-M, March 2000

List of Changes to DPC-NE-3005-PA Revision 3b

Revision 4 was submitted to the NRC to update the report for gadolinia implementation. The following list documents the changes to the report that were submitted.

1. **The cover page was revised for the submittal date and the revision number updated to Rev. 4**

2. **The abstract was revised to summarize the changes included in revision 4.**

3. **The table of contents was updated to add the new CASMO-4 section.**

4. **Chapter 1.1 was revised to add a brief description of revision 4 changes.**

5. **The CASMO-3 description in Chapter 1.2 was replaced with the CASMO-4 description.**

6. **A subsection describing gadolinia fuel was added at the end of Chapter 1.3**

7. **References 1-2, 1-3, 1-6, 1-10, and 1-26 are updated for the latest revision numbers and dates.**

8. **Reference 1-21 is updated to replace the CASMO-3 reference with the appropriate CASMO-4 reference**

9 **Reference 1-22 is updated to replace the DPC-NE-1004-A reference with DPC-NE-1006-P.**

10. **Reference 1-23 is updated to replace the reference for the DPC-NE-1004 SER with the yet to be issued SER for DPC-NE-1006-P.**

11. **Reference 1-41 is added for the yet to written gadolinia LAR submittal letter.**

12. **The CASMO-3 discussion is Chapter 2.4 is revised to state that CASMO-3 will no longer be used but since it was used in some of the demonstration analyses, the description is retained.**

13. **The SIMULATE-3K description in Chapter 2.7.1 is updated to describe the models in the newest (as of Revision 4) version of the code.**

14. **The SIMULATE-3K discussion in Chapter 2.7.2 is updated to describe the validation of the newest (as of Revision 4) version of the code.**

15. **Chapter 2.8 is added for the CASMO-4 description. Former Chapter 2.8 is renumbered to 2.9.**

16. **References 2-3, 2-5, 2-8, 2-10, and 2-15 are updated for the latest revision numbers and dates.**

17. **References 2-19 and 2-20 are deleted as part of the revision to Chapter 2.7.2.**

18. **References 2-24 and 2-25 are added for CASMO-4.**

19. **Reference 2-26 is added for DPC-NE-1006-P, 2-27 is added for the SIMULATE-3K assessment of the transient models, and 2-28 is added for an older version of the SIMULATE-3K theory manual that contains the TRAC benchmarking.**

20. **References 5-2, 5-3, 6-2, 6-4, 8-2, 9-2, 9-3, and 9-7 are updated for the latest revision numbers and dates.**

21. Chapter 10.3.2.5 is revised to state that gadolinia rods are modeled as if they are UO_2 rods and that the initial gap conductance is determined by matching the initial VIPRE temperature to a bounding core average fuel temperature. As a result of using the bounding core average fuel temperature, Reference 10-7 (TACO-3) is deleted.

22. References 10-2 and 10-3 are updated for the latest revision numbers and dates.

23. References 3-1, 9-6, and 10-6 are updated to replace the DPC-NE-1004-A reference with DPC-NE-1006-P.

24. References 11-2, 11-4, 12-2, and 13-1 are updated for the latest revision numbers and dates.

25. The fuel conduction model discussion in Chapter 14.2.4.1 is revised to state that the initial gap conductance is determined that will yield a boundingly high fuel temperature and that gadolinia properties are used for gadolinia rods and the justification provided.

26. The gadolinia engineering hot channel factor is added in the Conservative Factors subsection of Chapter 14.2.4.1.

27. The Fuel Conduction Model subsection of Chapter 14.2.4.2 is revised to change "conductivity" to "conductance" and to state that gadolinia rods are modeled as if they are UO_2 rods.

28. "conductivity" is changed to "conductance" in the Fuel Conduction Model subsection of Chapter 14.2.4.3.

29. Chapter 14.3.1, last sentence, "bound all future Oconee reload core designs" is changed to "bound future Oconee reload core designs". The word "all" is also stricken from the 2nd to last sentence in the last paragraph of the Scram Curve and Worth subsection.

30. The Reactor Trip and Single Failure subsection of 14.3.1 is revised as follows:

    "The reactor trip signal is generated for all six transients when three of four excore detectors exceed the high flux or flux/flow trip setpoint used during cycle operation. This conservative modeling assumes that the detector which would indicate the highest flux level has failed and that a two out of the remaining three logic is required to generate a trip signal. The excore signals are synthesized from a ~~conservative~~ radially weighted combination of power densities ~~of several assemblies in the proximity of each excore detector~~from every assembly in the core. Additionally, the high RCS pressure trip is credited. A conservative trip delay time is assumed for each of the trip signals."

31. The Scram Curve and Worth subsection of 14.3.1 is revised as follows:

    "The reactor scram assumes that the ejected rod, part length axial power shaping rods, and the remaining rod with the highest worth do not fall into the core. The remaining control rods (reduced by an appropriate uncertainty) drop into the core at a speed that satisfies the maximum rod drop time in the technical specifications. ~~The remaining scrammed rod worth is reduced to ensure that only the minimum net shutdown margin is achieved. Additional conservatism is applied by limiting the rate at which the scrammed rods are allowed to fall into the core such that the reactivity inserted is bounded by the limiting curves."~~

32. References 14-3 and 14-9 are updated for the latest revision numbers and dates.

33. The gadolinia engineering hot channel factor is added in Chapter 15.3.1.2.4.

34. References 15-2, 15-3, 16-2, and 16-3 are updated for the latest revision numbers and dates.

35. Reference 15-4 is updated to replace the DPC-NE-1004-A reference with DPC-NE-1006-P.