Duke Energy®

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September 17, 2008

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
Requests for Additional Information for Proposed License Amendment Request to Revise the Technical Specifications for AREVA NP Mark-B-HTP Fuel and for Methodology Report DPC-NE-2015-P "Mark-B-HTP Fuel Transition Methodology"
License Amendment Request No. 2007-12

Duke Energy Carolinas, LLC (Duke) submitted a license amendment request (LAR) dated October 22, 2007, for the Oconee Nuclear Station Renewed Facility Operating License (FOL) and Technical Specifications (TS) pursuant to 10 CFR 50.90. Specifically, Duke requested NRC review and approval of methodology report DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology" and revisions to Technical Specifications 2.1.1.2 and 5.6.5.b. Associated revisions to associated Technical Specification Bases B.2.1.1 and B.3.4.1 are provided. These revisions will allow the use of the AREVA NP Mark-B-HTP fuel design at the Oconee Nuclear Station beginning with Oconee Unit 2 Cycle 24 in December 2008. The Mark-B-HTP design is currently in use at several B&W design reactors.

Duke met with the NRC on March 3, 2008 to facilitate the LAR review. In emails dated May 8, 2008 and May 28, 2008, Duke received requests for additional information (RAI). Duke submitted responses to this RAI on July 14, 2008.

On August 27, 2008, following a conference call between Duke and the NRC, additional clarification was requested to the earlier responses to questions 6, 9, and 10. This submittal supersedes Duke's earlier RAI submittal dated July 14, 2008, and includes the revisions to these questions as well as a restatement of Duke's prior responses to the remaining questions.

Attachment 1 contains information that is proprietary to Duke and AREVA NP. In accordance with 10 CFR 2.390, Duke requests that this information be withheld from public disclosure. Affidavits are included (Enclosures 2 and 3) from each organization attesting to the proprietary nature of the information in the report. The specific information that is proprietary to each

Attachment 1 to this letter contains sensitive information
Withhold From Public Disclosure Under 10 CFR 2.390(d)(1).
Upon removal of Attachment 1, this letter is uncontrolled.

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organization is identified in the report. A non-proprietary version of this report is included in Attachment 2 that is suitable for public dissemination.

As communicated earlier, Duke requests approval of the LAR by September 30, 2008 with the amendment to become effective commencing with Oconee Unit 2 Cycle 24. This response is bounded by the initial review and approval of the Plant Operations Review Committee and Nuclear Safety Review Board; therefore, additional reviews were not required. Additionally, a copy of this response 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) 885-3364.

Sincerely,

Dave Baxter, Vice President
Oconee Nuclear Site

Enclosures:
1. Notarized Affidavit of Dave Baxter
2. Notarized Affidavit of T. C. Geer
3. Notarized Affidavit of Gayle F. Elliott

Attachment:
1. Oconee Nuclear Station, Mark-B-HTP Fuel Transition Methodology, Response to NRC Request for Additional Information [Proprietary Version - Withhold from Public Disclosure]
2. Oconee Nuclear Station, Mark-B-HTP Fuel Transition Methodology, Response to NRC Request for Additional Information [Non-Proprietary Version].

bc w/enclosures and attachments:

Mr. Luis Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. Lenny Olshan, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14 H25
Washington, D. C. 20555

Mr. Andy Hutto
Senior Resident Inspector
Oconee Nuclear Site

Ms. Susan E. Jenkins, Manager,
Infectious and Radioactive Waste Management Section
2600 Bull Street
Columbia, SC 29201

ENCLOSURE 1

AFFIDAVIT OF DAVE BAXTER

AFFIDAVIT

Dave Baxter, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Carolinas, LLC, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Dave Baxter, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 17 day of September 2008

Sheila A Smith
Notary Public

My Commission Expires:

6-12-2013
Date

SEAL

ENCLOSURE 2

AFFIDAVIT OF T. C. GEER

AFFIDAVIT OF THOMAS C. GEER

1. I am Vice President of Duke Energy Corporation and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke.

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 proprietary information sought to be withheld in the submittal is that which is marked in the proprietary version of the response to the Request for Additional Information from the Nuclear Regulatory Commission concerning Revision 0 to the Duke methodology report DPC-NE-2015-P, *Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology.* This information enables Duke to:

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

   (b) Perform nuclear design calculations on Oconee reactor cores containing low enriched uranium fuel.

(Continued)

Thomas C. Geer

T. C. Geer

(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.

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

Thomas C. Geer

Thomas C. Geer

Subscribed and sworn to me: June 19, 2008

Date

Deborah S. Rome Deborah S. Rome

Notary Public

My Commission Expires: December 19, 2009

SEAL

DEBORAH S ROME
NOTARY
PUBLIC
CATAWBA COUNTY, NC

ENCLOSURE 3

AFFIDAVIT OF GAYLE F. ELLIOTT

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 Response to NRC RAI on DPC-NE-2015-P, Revision 0, "Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology," dated June 2008 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 June, 2008.

____

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

ATTACHMENT 2

DPC-NE-2015-P, REVISION 0

RESPONSES TO NRC REQUEST FOR INFORMATION

[NON-PROPRIETARY VERSION]

A Request for Additional Information (RAI) regarding the submitted methodology report DPC-NE-2015 was received via email on May 8, 2008. The email was sent by L. N. Olshan of the NRC to G. B. Swindlehurst of Duke Energy. The email contained the first eight questions listed below. A second email sent on May 28, 2008 added the 9th and 10th questions listed below.

On August 27, 2008, following a conference call between Duke and the NRC, additional clarification was requested and added to the responses to questions 6, 9, and 10.

1. References 2-1 and 2-2 do not provide evidence that Mark-B-HTP fuel design is approved to 62 GWd/MTU. Please provide the justification for the burnup limit of 62 GWd/MTU for Mark-B-HTP fuel design. Also, if Mark-B-HTP fuel was evolved from the Mark-B11 fuel, then please address the plant-specific requirements in the SE for Mark-B11 fuel design in BAW-10229P.

2. Provide technical explanation how the fuel assembly bow is accounted for in fuel melting, clad strain, DNBR, and LOCA power distribution analyses in Revisions 4-3, 4-4, and 4-5.

3. Provide justification of the nuclear uncertainty factors in Revision 4-8.

4. Provide an example calculation of the densification power spike in Revision 4-10.

5. Provide justification how the COROS02 model meets the 100 microns corrosion limit by reducing 10 microns for best estimate oxide thickness in Revision 5-2.

6. Provide the mixed core analysis with regard to seismic and/or LOCA loading as described in Appendix A to SRP 4.2.

7. Provide the revised Figure 6-1 in DPC-NE-2003P-A in Revision 6-7.

8. Provide justification that the use of BHTP CHF correlation meets the Limitations and Conditions described in the SE of BAW-10241P, Revision 1.

9. Provide technical basis or justification for the revised pin power distribution in Appendix E to DPC-NE-3000-P for the Mark-B-HTP fuel. Technical information should include analytical results and assumptions used which are bounded by the TS and within the acceptable ranges of the approved methodologies for Chapter 15 events.

10. Elaborate the conditions of using different approaches in mixed core effects to account for DNBR penalty in Section 7.5.

RAI Question # 1

References 2-1 and 2-2 do not provide evidence that Mark-B-HTP fuel design is approved to 62 GWd/MTU. Please provide the justification for the burnup limit of 62 GWd/MTU for Mark-B-HTP fuel design. Also, if Mark-B-HTP fuel was evolved from the Mark-B11 fuel, then please address the plant-specific requirements in the SE for Mark-B11 fuel design in BAW-10229P.

Response

Section 3.3 of BAW-10179P-A, Revision 7 (Reference 1), describes AREVA's standard Mark-B fuel assembly and states that as an option to enhance fretting resistance, the standard Mark-B product incorporates the HTP grid. This section also states that the NRC staff has found the standard Mark-B fuel to be acceptable to a rod average burnup of 62 GWd/mtU. The Mark-B-HTP fuel design is an evolution of the standard Mark-B fuel product, using standard 0.430 inch OD Mark-B fuel rods. Thus, the burnup limit for the Mark-B-HTP design is 62 GWd/mtU. The Mark-B-HTP fuel design includes M5® cladding, instrument, and guide tubes and BAW-10227P-A (Reference 2) provides justification for use of M5® to the approved fuel rod average burnup limit of 62 GWd/mtU.

Fuel rod mechanical analyses for the Mark-B-HTP design are performed with TACO3 (Reference 3) as described in DPC-NE-2008P-A (Reference 4). The NRC approved increasing the TACO3 burnup limit from 60 to 62 GWd/mtU in Reference 5.

The Mark-B-HTP fuel design is not an evolution of the Mark-B11 fuel design. The plant-specific requirements in the BAW-10229P (Reference 6) SE do not specifically pertain to Mark-B-HTP fuel, but the analyses specified in the SE have been performed for Mark-B-HTP fuel. Duke has completed plant specific cladding oxidation, rod internal pressure, and clad overheating analyses for Mark-B-HTP fuel using the methodology given in Reference 4. AREVA has performed ECCS related analyses for Mark-B-HTP fuel at Oconee as discussed in Chapter 8 of DPC-NE-2015.

References

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

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

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

4) Duke Power Company Fuel Mechanical Reload Analysis Methodology Using TACO3, DPC-NE-2008P-A, April 1995.

5) Letter from R. C. Jones (NRC) to J. H. Taylor (FCF), Extending Burnup Limit for TACO3, January 11, 1996.

6) Mark-B11 Fuel Assembly Design Topical Report, BAW-10229P-A, Revision 0, April 2000.

RAI Question # 2

Provide technical explanation how the fuel assembly bow is accounted for in the fuel melting, clad strain, DNBR, and LOCA power distribution analyses in Revisions 4-3, 4-4, and 4-5.

Response

Fuel assembly bow enlarges the inter-assembly water gap which increases neutron moderation which in turn increases pin power peaking. The effect of fuel assembly bow is strongest in the fuel pins near to the enlarged inter-assembly water gap and weaker in fuel pins farther away from the enlarged water gap. Duke assumed diagonal bowing to grid contact and modeled a large number of different fuel type combinations to determine the fuel assembly bow penalties.

Because the effect of fuel assembly bowing is location-specific, the fuel pins in each assembly were divided into six regions as shown in the layout map below. A pin power peaking factor was chosen for each region to bound all of the pin power increases in that region. The largest pin power peaking factors were determined to be in fuel combinations containing only Mark-B-HTP assemblies. Typical peaking factors are shown in the table below. Note that Duke may change the number of pin regions and re-define the layout of those regions in any future calculations of fuel assembly bow penalties.

Fuel Assembly Bow Pin Peaking Factor Application Region Layout Map

1	1	2	2	2	2	2	2	2	2	2	2	2	1	1
1	3	3	3	3	3	3	3	3	3	3	3	3	3	1
2	3	4	4	4		4	4	4		4	4	4	3	2
2	3	4		5	5	5	5	5	5	5		4	3	2
2	3	4	5	6	6	6	6	6	6	6	5	4	3	2
2	3		5	6		6	6	6		6	5		3	2
2	3	4	5	6	6	6	6	6	6	6	5	4	3	2
2	3	4	5	6	6	6		6	6	6	5	4	3	2
2	3	4	5	6	6	6	6	6	6	6	5	4	3	2
2	3		5	6		6	6	6		6	5		3	2
2	3	4	5	6	6	6	6	6	6	6	5	4	3	2
2	3	4		5	5	5	5	5	5	5		4	3	2
2	3	4	4	4		4	4	4		4	4	4	3	2
1	3	3	3	3	3	3	3	3	3	3	3	3	3	1
1	1	2	2	2	2	2	2	2	2	2	2	2	1	1

Region #	# of Fuel Pins in the Region	Pin Peaking Factors for all Mark-B-HTP Cores	Description of the Region
1	12	1.082	The 3 pins in each corner of the assembly
2	44	1.065	The remaining pins on the outer row
3	48	1.046	The pins on the 2nd row in from the periphery
4	32	1.021	The pins on the 3rd row in from the periphery
5	28	1.014	The pins on the 4th row in from the periphery
6	44	1.012	The remaining interior pins

For centerline fuel melt and cladding strain (CFM/CS) analyses, the fuel assembly bow peaking factor is statistically combined with other factors to form a statistically combined uncertainty factor (SCUF) for CFM/CS analysis. The SCUF equation for CFM/CS analysis is shown below.

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

where :

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

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.

The LOCA SCUF equation is identical to the CFM/CS SCUF equation described above, except that the fuel densification power spike factor is not included in the SCUF equation for LOCA analysis. Note that credit may be taken for the additional cooling available due to the enlarged water gap to reduce the fuel assembly bow peaking factors applied in the LOCA analysis.

DNB calculations are performed on a sub-channel basis as described in DPC-NE-2003P-A and DPC-NE-3000P-A. As noted earlier, the enlarged water gap results in the largest local power increase in the fuel rods directly adjacent to the water gap with other fuel rods seeing less of an effect with increasing distance from the enlarged water gap. For the peripheral fuel rods where the effect is the largest, the fuel assembly bow results in increased cooling due to the enlarged water gap. The [ ] VIPRE model is required to accurately model fuel assembly D bow and was used to determine the DNB penalty for assembly bow as described below.

The VIPRE model is modified 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 considered a wide range of statepoint conditions and axial power shapes, a single bounding fuel assembly bow pin peaking factor for DNB analyses was determined for Mark-B-HTP fuel. This peaking factor is statistically combined with other factors to form a SCUF for statistical core design (SCD) based DNB analysis.

The SCUF equation for SCD-based DNB analysis is shown below.

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

where :

Bias = assembly radial power bias
LBP = 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.

RAI Question # 3

Provide justification of the nuclear uncertainty factors in Revision 4-8.

Response

The uncertainty factors being revised were developed with a CASMO-2 based EPRI-NODE-P core model. They were superseded by uncertainty factors developed with a CASMO-3 based SIMULATE-3 core model in DPC-NE-1004A (Reference 4-3), which was approved by the NRC in November 1992. The uncertainty factors in DPC-NE-1002A were never updated to match the new factors in DPC-NE-1004A, so they are being updated now. The updated bias and nuclear uncertainty factors are all taken from the table in Section 5.1 of DPC-NE-1004A.

RAI Question # 4

Provide an example calculation of the densification power spike in Revision 4-10.

Response

The fuel densification power spike factors are provided to Duke by AREVA. The power spike factors are available for several percentages of in-reactor fuel densification and for several fuel enrichments. The representative table of Mark-B-HTP power spike factors shown below is for in-reactor fuel densifications up to 1.5 % and fuel enrichments up to 5 % U^{235}.

Linear interpolation is used within the table to determine a power spike factor to be applied at each axial level in the core model. Note that the lowest data point in the table was added by Duke to preclude extrapolation to a power spike factor less than 1.0 at the bottom of the core.

A

The fuel densification power spike factor is statistically combined with other factors to form a statistically combined uncertainty factor (SCUF) for centerline fuel melt and cladding strain (CFM/CS) analyses.

RAI Question # 5

Provide justification how the COROS02 model meets the 100 microns corrosion limit by reducing 10 microns for best estimate oxide thickness in Revision 5-2.

Response

Duke will perform corrosion analyses for Mark-B-HTP fuel using AREVA's approved methodology discussed in Reference 1. Reference 2 approved Duke Energy's use of the COROS02 model and AREVA NP's corrosion methodology for the Oconee Nuclear Station. The January 25, 1999 NRC letter (Reference 3), included in BAW-10186P-A (Reference 1), approves the use of the COROS02 model for best estimate calculation of corrosion with a limit of 100 microns. Reference 1 also includes an October 28, 1997 letter from J. H. Taylor to USNRC (Reference 4) that documents how COROS02 is used for oxide thickness calculations. Reference 4 states that "FCF (now AREVA NP) will use available high burnup data for FCF fuel designs to quantify the amount of conservatism in the model at the NRC imposed 100 micron limit. The prediction will then be adjusted by this amount and used as a best estimate predictor for oxide thickness calculation as appropriate."

AREVA NP used available high burnup oxide data for AREVA NP fuel designs to determine that the COROS02 model at the 100 micron limit is conservative by [ ] microns. COROS02 is used to predict the oxide thickness for each reload core. [ ] The best estimate oxide thickness must be less than the 100 micron limit. A

References

1) Extended Burnup Evaluation, BAW-10186P-A, Revision 2, AREVA NP, June 2003.

2) Letter, D. LaBarge (NRC) to W. R. McCollum, Jr, (Duke), "Use of Framatome Cogema Fuels Topical Report on High Burnup – Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. MA0405, MA0406, MA0407)", March 1, 1999.

3) Letter, T. H. Essig (USNRC) to J. H. Taylor (FCF), "Acceptance for Referencing of Framatome Cogema Fuels Topical Report BAW-10186P: Extended Burnup Evaluation (TAC No. MA3705)", January 25, 1999.

4) Letter, J. H. Taylor (FCF) to NRC Document Control Desk, JHT/97-39, "Application of BAW-10186P-A, Extended Burnup Evaluation", October 28, 1997.

RAI Question # 6

Provide the mixed core analysis with regard to seismic and/or LOCA loading as described in Appendix A to SRP 4.2.

Response

Mark-B-HTP fuel assemblies will be operational for the first time in Oconee Unit 2 Cycle 24. These fuel assemblies will be present in the Cycle 24 core along with the resident Mark-B11 fuel.

The horizontal faulted analysis computed the worst case loads for the BOL and EOL fuel assemblies due to LOCA and seismic events. Load cases considered included core flood and decay heat LOCA loads, and safe-shutdown earthquake (SSE) seismic loads in the X and Z horizontal directions. Both mixed core and all Mark-B-HTP configurations were evaluated. The maximum peak impact force observed occurred during the SSE-X event for the 5 row EOL Mark-B-HTP configuration.

The Mark-B11 grid properties (local grid damping and translational stiffness) are incorporated in the model. It is assumed that the grid modifications do not affect the general behavior and natural frequencies of the fuel assembly. The peak grid impact forces and the fuel assembly shears and moments for each configuration are tabulated in Table 1 on the next page. The maximum peak impact force observed is [A
]. This impact occurs during the SSE-Z event for the 5 row EOL Mark-B-HTP configuration.

The listed peak impacts, shears and moments in Table 1 are shown to be qualified for both the Mark-B-HTP and the resident Mark-B11 fuel assemblies. The fuel assemblies remain elastic due to the faulted loads and moments. Fuel rod fragmentation does not occur due to seismic loading and fuel system coolability (control rod insertion) is maintained per the requirements of NUREG-0800, Standard Review Plan, Section 4.2 Appendix A.

Table 1 Peak Impacts, Shears and Moments

A

RAI Question # 7

Provide the revised Figure 6-1 in DPC-NE-2003P-A in Revision 6-7.

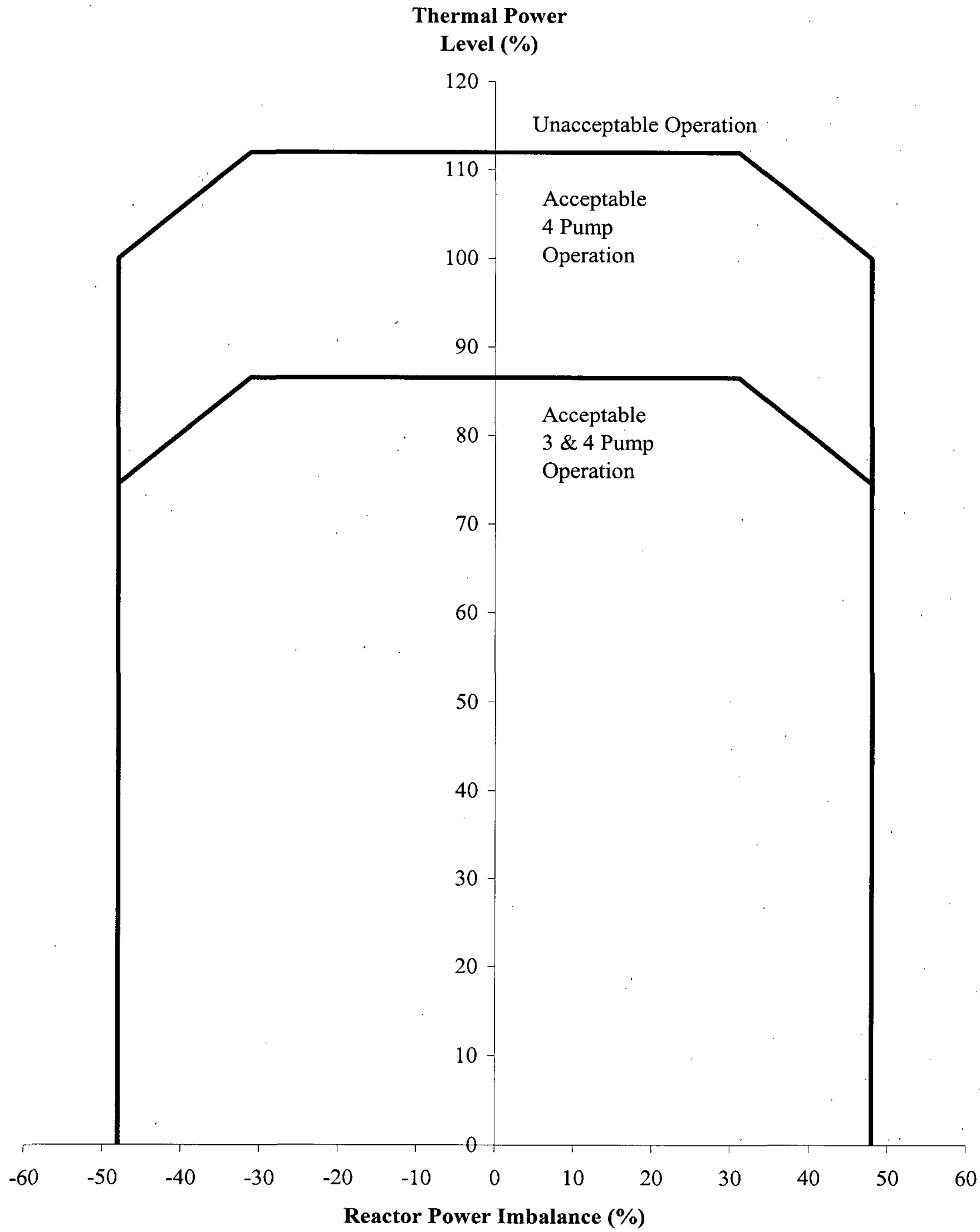
Response

The revised Figure 6-1, with the [ ] limit line removed, is presented on the following page. The data points in the figure are shown in the table below. D

Power-Imbalance Safety Limits for 4 and 3 Reactor Coolant Pump (RCP) Operation

% Imbalance	4 RCP Operation (% full power)	3 RCP Operation (% full power)
-48.0	0.0	0.0
-48.0	100.0	74.6
-31.1	112.0	86.6
+31.1	112.0	86.6
+48.0	100.0	74.6
+48.0	0.0	0.0

Figure 6-1 RPS Core Protection Safety Limits



RAI Question # 8

Provide justification that the use of BHTP CHF correlation meets the Limitations and Conditions described in the SE of BAW-10241P, Revision 1.

Response

As discussed on page 6-9 of DPC-NE-2015 (Appendix F of DPC-NE-2005-P), the complete BHTP CHF correlation database, including the additional [ ] data points used to justify extension of the range of application of the correlation (Reference 1), was analyzed with the VIPRE-01 code. The predicted CHF to measured CHF (P/M) ratios are plotted against mass velocity, pressure and thermodynamic quality in attached Figures 1, 2 and 3. Figures 1-3 show (1) that there is no bias of VIPRE-01 predicted to measured CHF values with respect to mass velocity, pressure and thermodynamic quality, and (2) the BHTP correlation in VIPRE-01 conservatively predicts CHF for the extended range of independent parameters. Figures 1-3 compare closely with the same parameter representations in Reference 1 (Figures A.3, A.6 and A.7).

The BHTP CHF correlation will be used for DNBR analyses of Mark-B-HTP fuel using VIPRE-01 and the BHTP correlation will be applied within the range of independent variables given in Table F-3 (pg. 6-14) of DPC-NE-2015. The range of variables in Table F-3 is identical to the extended range of variables given in Table 1 of Reference 1. If operating conditions require extrapolations beyond the approved pressure or quality ranges, the limitations and conditions listed in Section 4 of Reference 1 will be adhered to:

- When pressure greater than the pressure limit of 2425 psia, but less than 2600 psia, is encountered, all of the local coolant conditions are calculated at the upper pressure limit of 2425 psia using the NRC-approved VIPRE-01 thermal-hydraulic code and then used in the calculation of the BHTP CHF.

- Extrapolation below the minimum quality range is performed with no lower limit, consistent with EMF-92-153(P)(A) Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel".

No other parameter extrapolations are performed.

Reference

1) Letter, H. N. Berkow (USNRC) to R. L. Gardner (Framatome ANP), Final Safety Evaluation for Framatome ANP (FANP), Appendix A to Topical Report (TR) BAW-10241(P), Revision 1, "Extension of the BHTP CHF (Critical Heat Flux) Correlation Ranges", TAC No. MC6374, July 25, 2005.

**Figure 1. VIPRE-01 P/M CHF versus Local Mass Velocity
Original and New (Uncorrelated) BHTP Data with BHTP Correlation**

A,D

Figure 2. VIPRE-01 P/M CHF versus System Pressure
Original and New (Uncorrelated) BHTP Data with BHTP Correlation

A,D

Figure 3. VIPRE-01 P/M CHF versus Thermodynamic Quality at CHF Original and New (Uncorrelated) BHTP Data with BHTP Correlation

A,D

RAI Question # 9

Provide technical basis or justification for the revised pin power distribution in Appendix E to DPC-NE-3000-P for the Mark-B-HTP fuel. Technical information should include analytical results and assumptions used which are bounded by the TS and within the acceptable ranges of the approved methodologies for Chapter 15 events.

Response

The revised pin power distribution used as input for VIPRE hot assembly modeling for UFSAR Chapter 15 non-LOCA transients and accidents will originate from a SIMULATE-3 model for a recent core design. This model will be from the time in core life that has the maximum radial peak, and therefore typically the minimum DNBR margin. This pin power distribution will be flattened by applying a factor to each pin power that decreases the difference in pin power relative to the maximum pin power. This is done to introduce conservatism since a flattened power distribution is more conservative for DNB due to less mixing in the limiting subchannel. This revised pin power distribution will then be a conservative input to the VIPRE models. The revised distribution included in Appendix E is to demonstrate the process and the distribution that could be expected using that process. Prior to using the method to generate the revised distribution for reload analyses, analyses will be performed at various thermal hydraulic conditions and axial power shapes to ensure that the revised pin power distribution provides conservative DNBR predictions relative to actual pin power distributions obtained from SIMULATE-3. Additionally, analyses will be performed to confirm that the revised pin power distribution remains conservative for future reload cores, or a new revised pin power distribution will be developed using the same process.

This revised pin power distribution will be used in the same manner as the current vendor-supplied pin power distribution. There are no changes in how the VIPRE models are applied to predict the DNBR result for UFSAR Chapter 15 non-LOCA transients and accidents. There are no resulting changes to the technical specifications or the Core Operating Limits Report or the acceptable ranges of the approved methodologies, with the exception that the numerical results of the VIPRE analyses will be different. The DNBR acceptance criteria will remain those that have been reviewed and approved by the NRC in the Duke methodology reports.

RAI Question # 10

Elaborate the conditions of using different approaches in mixed core effects to account for DNBR penalty in Section 7.5.

Response

The last paragraph of Section 7.5 will be revised to eliminate use of the first approach as described in Section 6.4. The original and revised paragraphs are shown below.

Original Paragraph

"With regard to the modeling of the mixed core effects in VIPRE-01 during UFSAR Chapter 15 transients and accidents, two approaches will be used. The first approach is to use the mixed core penalty developed as described in Section 6.4. This penalty can then be applied as a peaking penalty or a DNBR penalty. An additional approach using more detailed modeling of mixed cores has been developed. For each of the Oconee VIPRE-01 models described in Section 2.3, in Appendices D and E, and in DPC-NE-3005-PA, the mixed core effect can be explicitly modeled by including the number and location of each fuel assembly type in each VIPRE-01 model. For example, in a VIPRE-01 model that has different fuel assemblies modeled, the core loading pattern for a mixed core can be used to specifically model the spatial relationship of each fuel assembly type. ..."

Revised Paragraph

"With regard to the modeling of the mixed core effects in VIPRE-01 during UFSAR Chapter 15 transients and accidents, the following approach will be used. For each of the Oconee VIPRE-01 models described in Section 2.3, in Appendices D and E, and in DPC-NE-3005-PA, the mixed core effect can be explicitly modeled by including the number and location of each fuel assembly type in each VIPRE-01 model. For example, in a VIPRE-01 model that has different fuel assemblies modeled, the core loading pattern for a mixed core can be used to specifically model the spatial relationship of each fuel assembly type. ..."