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Ken S. Canady
Vice President
Nuclear Engineering

September 12, 2002

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

Subject: Catawba Nuclear Station Units 1 and 2; Docket Nos. 50-413, 50-414
McGuire Nuclear Station Units 1 and 2; Docket Nos. 50-369, 50-370
Response to Request for Additional Information - Topical Report DPC-NE-1005P, Revision 0, *Nuclear Design Methodology UsingCASMO-4/ SIMULATE-3 MOX* (Proprietary)

Reference: NRC Letter dated July 29, 2002, Request for Additional Information Re: Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX (TAC Nos. MB2578, MB2579, MB2726 and MB2729)

Attached please find Duke Energy's response to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) transmitted by the reference letter.

This submittal contains information that is proprietary to Duke Energy, Studsvik Scandpower, Incorporated, the Electric Power Research Center, and Framatome ANP. The specific information that is proprietary to each organization is identified in Attachment 1. In accordance with 10 CFR 2.790, Duke requests that this information be withheld from public disclosure. Affidavits are included from each of the organizations that attest to the proprietary nature of the information in this submittal. Attachment 2 is a redacted version of the response to the RAI with proprietary information removed. Also enclosed are two copies of each of the proprietary documents requested in Question 20.

Please note that Duke has not yet obtained a proprietary affidavit from Electricité de France covering (i) information in the Question 9 response and (ii) two of the references requested in Question 20. As a result, the response to Question 9 has not been included in this submittal, and the response to Question 20 is not complete. Duke anticipates receiving the affidavit and providing the remainder of the information shortly.

DPC-NE-1005P was submitted to NRC for review on August 3, 2001. The NRC Staff has informally indicated that the target date for issuing the Safety Evaluation Report (SER) on DPC-NE-1005P is January, 2003. Duke intends to transition its reload design process to the DPC-NE-1005P methodology once the topical report has been approved by the NRC. Please confirm that the January 2003 SER schedule is still valid, or contact us to discuss a revised schedule.

A001

PROPRIETARY
Material Attached

Inquiries on this matter should be directed to G. A. Copp at (704) 373-5620.

Very truly yours,

K. S. Canady

Attachments and Enclosures

xc with Attachment 1:

L. A. Reyes, Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

C. P. Patel, NRC Senior Project Manager
U. S. Nuclear Regulatory Commission
Mail Stop O-8 G9
Washington, DC 20555-0001

S. M. Shaeffer
NRC Sr. Resident Inspector
McGuire Nuclear Station

D. J. Roberts
NRC Sr. Resident Inspector
Catawba Nuclear Station

xc with 5 copies of Attachments
and 2 copies of proprietary documents:

R. E. Martin, NRC Senior Project Manager
U. S. Nuclear Regulatory Commission
Mail Stop O-8 G9
Washington, DC 20555-0001

bxc with Attachment 1:

P. M. Abraham – EC08I
D. E. Bortz – EC08G
R. R. St Clair – EC08G
S. B. Thomas – EC08G
L. F. Vaughn – PB05E
J. L. Eller – EC09A
S. P. Nesbit – EC09A
K.S. Canady – EC08
C. J. Thomas - MG01RC
M.T. Cash – EC05O
G. D. Gilbert – CN01RC
NRIA File/ELL – EC05O
McGuire Master File – MG01DM
Catawba Master File 801.01 – CN04DM
Catawba RGC Date File (J.M Ferguson – CN01SA)
MOX File – 1607.3203

bxc w Attachment 2:

R. H. Clark – Duke Cogema Stone & Webster
A. W. Cottingham – Winston & Strawn
G. A. Meyer – Framatome Advanced Nuclear Power
Michel Ponticq – Electricité de France
K. S. Smith – Studsvik Scandpower, Inc.
P. T. Rhoads – Department of Energy
D. J. Spellman – Oak Ridge National Laboratory

AFFIDAVIT OF K. S. CANADY

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 2.790 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 2.790, 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 2.790, 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 this submittal is that which is marked in the proprietary version of the response to the Request for Additional Information from the Nuclear Regulatory Commission dated July 29, 2002 concerning Duke topical report DPC-NE-1005, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX.* This information enables Duke to:

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

K. S. Canady

(Continued)

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

(c) Perform nuclear design calculations on future planned McGuire and Catawba reactor cores containing a mixture of low enriched uranium and mixed oxide fuels.

(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.

K. S. Canady, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

K. S. Canady

Sworn to and subscribed before me this 12TH day of September, 2002.
Witness my hand and official seal.

Mary P. Nelms
Notary Public

My commission expires: JAN 22, 2006

SEAL

AFFIDAVIT OF KORD SMITH

1. My name is Kord Smith. I am Vice President of Studsvik Scandpower, Inc. (SSP) and as such have the responsibility for reviewing information sought to be withheld from public disclosure and am authorized on the part of SSP to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy Corporation's application for withholding, which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, 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 SSP and has been held in confidence by SSP and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by SSP.

   (iii) The information is to be transmitted to the NRC in confidence under the provisions of 10CFR 2.790, and 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 consists of documentation for the computer codes CASMO-4, CMS-LINK, and SIMULATE-3 MOX and responses to NRC questions concerning said computer codes contained in Duke's response to NRC Request for Additional Information dated July 29, 2002. The proprietary information sought to be withheld from public disclosure has substantial commercial value to SSP because the information:

      (a) Is not available to other parties and would require substantial cost to develop independently,

(Continued)

Kord S. Smith

Kord Smith

(b) Has been sought by and provided to other parties in return for monetary payment,

(c) Is not readily available to others and therefore has substantial value to SSP.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to SSP, because;

(a) SSP markets and sells the computer codes to nuclear utilities for the purpose of supporting the operation and licensing of nuclear power plants,

(b) The subject information could only be duplicated by competitors at similar expense to that incurred by SSP.

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

Kord Smith, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

Kord Smith

Subscribed and sworn to before me on this 6TH day of SEPTEMBER, 2002
Witness my hand and official seal.

Notary Public

My Commission Expires: 3-15-08

SHAWNA LEE SMITH NOTARY PUBLIC STATE OF IOWA

SEAL

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is James F. Mallay. I am Director, Regulatory Affairs, for Framatome ANP ("FRA-ANP"), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the information contained in a series of eight reports developed by Framatome ANP related to the EPICURE and ERASME critical experiments and provided by Duke Power to the NRC in support of its topical report DPC-NE-1005 P. A listing of these documents is attached hereto and these reports are referred to herein as "Documents." Information contained in these Documents has been classified by FRA-ANP as proprietary in accordance with the policies established by FRA-ANP for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by FRA-ANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in the Documents be withheld from public disclosure.

6. The following criteria are customarily applied by FRA-ANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FRA-ANP'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 FRA-ANP.

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

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

7. In accordance with FRA-ANP's policies governing the protection and control of information, proprietary information contained in these Documents have been made available, on a limited basis, to others outside FRA-ANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FRA-ANP 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 3rd
day of September, 2002.

Ella F. Carr-Payne
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/05

ELLA F. CARR-PAYNE
Notary Public
Commonwealth of Virginia
My Commission Exps. Aug. 31, 2005

Framatome Proprietary References not Reviewed by NRC

The following is the list of proprietary Framatome references from Duke's Topical Report DPC-NE-1005P, "Nuclear Design Methdology Using CASMO-4/SIMULATE-3 MOX." .

1. EPICURE Experiments, EPD-DC-293, Revision 0 (Proprietary), FRAMATOME, June 8, 1999.

2. "Experience EPICURE UMZONE Distribution Fine de Puissance en Presence d'une Grappe de 24 Crayons Absorbants B4C dans l'Assemblage MOX Zone Central et Effet d'Ombre 1, 9, 24 absorbants B4C," NT-SPRC-LPEx-93/124, Revision A (Proprietary), FRAMATOME, August 2, 1994.

3. "Experience EPICURE UMZONE Distribution Fine de Puissance en Presence d'une Grappe de 24 Crayons Absorbants AIC dans l'Assemblage MOX Zone Central," NT-SPRC-LPEx-92/78, Revision A (Proprietary), FRAMATOME, February 19, 1993.

4. "EPICURE Results of the Material Buckling Measurements in the MH1.2-93 Configuration," Framatome letter EPD/99.1183, Revision A, {Appendix A} from S. Tarle (FRAMATOME) to FRAMATOME COGEMA Fuels (Attention: George Fairburn, et. al) (Proprietary), November 3, 1999.

5. "Rapport d'Experience Programme EPICURE: Configuration UM 17x17/7% Mesures de la Distribution Fine de Puissance et des Rapports d'Activite d'une Chambre a Fission dans les Assemblages MOX et UO2 Adjacents," NT-SPRC-LPEx-95-025, Revision 0 (Proprietary), FRAMATOME, February 23, 1995.

6. "Programme EPICURE - Configuration UM17x17/11% Rapport d'Experience," NT-SPRC-LPEx-95-021, Revision 0 (Proprietary), FRAMATOME, February 23, 1995.

7. "Experience ERASME/L Description Geometrique et Bilan Matiere," SEN/LPRE n° 87-289 (Proprietary), FRAMATOME, June 1987.

8. "Resultats des Mesures D'effets en Reactivite et de Distributions de Puissance sur des Configurations avec une Grappe de 9 Crayons B4C Naturel dans le Cadre de L'experience ERASME/L," NTC-SPRC-LPEx-90/102 (Proprietary), FRAMATOME, June 14, 1990.

North Carolina State University is a land-grant university and a constituent institution of The University of North Carolina

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AFFIDAVIT OF PAUL TURINSKY

1. My name is Paul Turinsky. I am the Head of the Department of Nuclear Engineering at North Carolina State University. I also serve as Director of the Electric Power Research Center (EPRC) with primary responsibility for management and oversight. Duke Power is an EPRC member. I am also the Principal Investigator of the EPRC report titled "Evaluation of the Effects of Mixed LEU-MOX Core on Dynamic Rod Worth Measurement," a report that is sought to be withheld from public disclosure in connection with a Duke Power nuclear licensing action. I am authorized on the part of the EPRC to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Power's application for withholding, which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, 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 the EPRC and has been held in confidence by EPRC and its members.

   (ii) The information is of a type that would customarily be held in confidence by the EPRC. The information consists of the report "Evaluation of the Effects of Mixed LEU-MOX Core on Dynamic Rod Worth Measurement," describing work done under a Duke Power enhancement grant by the EPRC.

   (iii) The information is to be transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, and is to be received in confidence by the NRC.

   (iv) The information sought to be protected is currently not available in public to the best of our knowledge and belief.

(Continued)

Paul Turinsky

EPRC

(v) The proprietary information sought to be withheld is that which is contained in the report "Evaluation of the Effects of Mixed LEU-MOX Core on Dynamic Rod Worth Measurement." The proprietary information sought to be withheld from public disclosure has substantial commercial value to the EPRC because the information:

(a) Is not available to other parties and would require substantial cost and effort to develop independently,

(b) Describes a method of analysis and sensitivity studies that justify a method of measuring control rod worth in the presence of neutron energy spectral variations, which has potential value to other parties.

5. Public disclosure of this information is likely to cause harm to EPRC because it would allow other nuclear companies to benefit from the results of the EPRC methodology without requiring commensurate expense or allowing EPRC to recoup a portion of the expenditures or benefit from the sale of the information.

Paul Turinsky, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

Paul Turinsky

Subscribed and sworn to before me on this 5th day of September, 2002
Witness my hand and official seal.

Notary Public

My Commission Expires:

12/21/2003

SEAL

Attachment 2
Topical Report DPC-NE-1005P, Revision 0
Response to NRC Request for Additional Information Dated July 29, 2002

1. Please provide, in a side-by-side format, all of the changes made to CASMO-4 and SIMULATE-3 to accommodate the presence of mixed-oxide (MOX) fuel.

   Response:

   As a result of efforts by Studsvik Scandpower, Inc.(SSP) to enhance the accuracy of neutronics calculations for MOX-fueled cores, numerous changes have been made to the default parameters and models in the SSP codes.

   These changes are broken down here by code:

   [-- Remainder of response is proprietary --]s

2. In section 2.1, page 2-2, second paragraph from the end, it is stated that for a MOX fuel lattice, CASMO-4 automatically adjusts the detail of appropriate internal calculations to accommodate the variation of the plutonium cross-sections.

   1.01 Please provide additional details as to how this is accomplished.

   1.02 Also, it is stated in the same paragraph that CASMO-4 also edits several additional coefficients which are----. Which coefficients are referenced?

   Response:

   For response to Question 1.01, see responses detailed in items "a" and "b" of Question 1.

   For response to Question 1.02, see responses detailed in items "d" through "g" of Question 1.

3. The second paragraph on page 2-5 of the topical report states that several modifications were made to SIMULATE-3 to more accurately model the local flux gradients at the MOX-low enriched uranium (MOX-LEU) fuel interfaces. The same paragraph also briefly discusses other changes made to the SIMULATE-3 model to accommodate the presence of MOX fuel. Please provide a more detailed technical qualitative description (that is, the physics behind this claim) in support of the changes made to SIMULATE-3 to handle the presence of MOX fuel.

   Response:

   See the responses detailed in items "p" and "u" of Question 1, and the detailed discussion in Section 3 (pages 9-11) of SSP-00/420, "SIMULATE-3 MOX Enhancements and Verification Tests" (Reference 12), which is included in this submittal.

4. The last paragraph in section 2.3 addresses the issue of mixed cores, and indicates that the mixed core methodology applicable to LEU cores are also applicable to cores loaded with MOX and LEU. Please provide qualitative and quantitative technical justifications to support this claim.

Response:

This question concerns two sentences at the end of Section 2.3 of the topical report which provide a description of SIMULATE-3 MOX general model characteristics. The two sentences read as follows:

> "The modifications made to accommodate mixed cores of MOX and LEU fuel assemblies are also applicable to cores containing only LEU fuel. The new models yield results consistent with the results of the conventional methods in LEU cores."

Duke's SIMULATE-3 MOX core models divide each assembly radially into four equal size nodal volumes. Thus half of the nodal interfaces in the radial direction are within the assembly and half are at the exterior face of the assembly. As noted earlier, in mixed cores the nodal interfaces between MOX and LEU fuel assemblies are characterized by relatively steep flux gradients. Conversely, relatively benign flux gradients are present at the nodal interfaces between assemblies of the same type and at nodal interfaces within assemblies. Acceptable accuracy modeling mixed cores of MOX and LEU fuel indicates that SIMULATE-3 MOX adequately addresses both steep and benign flux gradients. Qualitatively, it is reasonable to expect that a code that models the mixed core problem well (with both steep and benign flux gradients) would also model the all-LEU core (with benign flux gradients only) in an acceptable manner.

This qualitative expectation is borne out by the quantitative results provided in the topical report. Tables 3-11 and 3-12 of the topical report summarize comparison results for McGuire and Catawba cores made up of only LEU fuel and St. Laurent cores made up of a mixture of MOX and LEU fuel. The benchmark results indicate comparable accuracy for both mixed cores and all-LEU cores.

The McGuire and Catawba core benchmarks presented in the topical report used the CASMO-4 and SIMULATE-3 MOX codes with the mixed core modifications in SIMULATE-3 MOX. The response to Question 7 summarizes the results of benchmarks of those same McGuire and Catawba cores with the currently approved methodology (CASMO-3 and SIMULATE-3). The currently approved methodology does not incorporate the mixed core modifications that are present in SIMULATE-3 MOX. Nevertheless, the topical report benchmarks and the currently approved methodology show comparable accuracy. This supports the topical report statement that the SIMULATE-3 MOX modifications are also applicable to cores containing all LEU fuel,

and that the results are consistent with conventional LEU core methods.

5. On page 2-7, it is stated that scaler multipliers may be applied to important parameters. How are the multipliers determined and who decides to apply them at the appropriate time?

Response:

The intent of this statement is to convey the general capability of SIMULATE-3K MOX. One of those capabilities is the ability to include conservatism in analyses by the use of scalar multipliers on selected parameters. Typically, scalar multipliers would be used for safety analysis applications in which bounding/conservative values of parameters are desired, rather than best estimate values. SIMULATE-3K MOX is only used in DPC-NE-1005P to support dynamic rod worth measurement as discussed in Section 6 of the report. The measurement of control rod worth requires a best estimate analysis; consequently, scalar multipliers were not used in any of the analyses described in this report.

6. On page 3-2, the last sentence of the second paragraph indicates that SIMULATE-3 MOX was compared to prior Duke methodologies. Were the prior Duke methodologies applied to the same type LEU fuel as is referred to in the methodologies described in DPC-NE-1005P, Revision 0?

Response:

Yes. All of the McGuire and Catawba cores that were benchmarked for this topical report were designed and analyzed using the currently approved CASMO-3 and SIMULATE-3 methodology.

7. On page 3-3, the second and third paragraphs also make reference to prior Duke methodologies. Therefore, question six above is also applicable to these paragraphs. Please explain. Additionally, for both paragraphs, the accuracy of the SIMULATE-3 MOX code is compared to predictions, so please quantify the accuracy of the results using: (a) the previous method and, (b) the SIMULATE-3 MOX method.

Response:

Concerning the use of prior Duke methodologies, the response to Question 6 is also applicable here. Table 1 provides a comparison between the proposed methodology (CASMO-4 / SIMULATE-3 MOX) and the currently approved methodology (CASMO-3 / SIMULATE-3) for the cores evaluated in the topical report. Figure 1 and Figure 2 show a comparison of hot full power (HFP) boron concentration over core life using both methodologies. As shown in the table and figures, the two methodologies predict the

boron concentrations, control rod worths, and isothermal temperature coefficients with comparable accuracy. As in the topical report, relative deviations are defined as measured values minus predicted values divided by the measured values.

Table 1

Summary Comparison of Benchmark Results

Parameter	Average Deviation	Standard Deviation
McGuire and Catawba with CASMO-4 / SIMULATE-3 MOX		
BOC HZP Soluble Boron (PPMB)		
HFP Soluble Boron (PPMB)		
BOC HZP Control Rod Bank Worth (%)		
BOC HZP ITC (pcm / F)		
St Laurent with CASMO-4 / SIMULATE-3 MOX		
BOC HZP Soluble Boron (PPMB)		
HFP Soluble Boron (PPMB)		
BOC HZP Control Rod Bank Worth (%)		
BOC HZP ITC (pcm / F)		
McGuire and Catawba with CASMO-3 / SIMULATE-3		
BOC HZP Soluble Boron (PPMB)		
HFP Soluble Boron (PPMB)		
BOC HZP Control Rod Bank Worth (%)		
BOC HZP ITC (pcm / F)		

D

Figure 1
CASMO-4 / SIMULATE-3 MOX HFP Boron Comparison

D

Figure 2
CASMO-3 / SIMULATE – 3 HFP Boron Comparison

D

8. In the first paragraph of section 3.2.5, the last sentence states that the fission chambers are very similar. What are the differences between them?

Response:

The response to this question contains proprietary information for which Duke has not yet received a supporting affidavit. Duke will respond to this question as soon as the affidavit is received to support withholding the information.

9. In the middle of the second paragraph from the bottom of page 3-9, it is stated that a small bias was applied to a measured signal. How small is this bias and how was the bias determined?

Response:

For the benchmark analyses in this report, a bias was applied to measured signals in MOX fuel locations, which reduced the signal by [ ]$_D$.

The fission chamber signal is almost entirely due to thermal neutron fissions in the highly enriched ^{235}U coating of the chamber. This neutron signal component is proportional to the neutron flux in the fuel assembly. The fission chamber signal also contains a small component due to ionizations caused by gamma rays. The gamma rays come primarily from fissions, so the gamma signal component is proportional to the fission rate (power) in the fuel assembly.

The thermal neutron absorption cross section is higher in ^{239}Pu than in ^{235}U. As a result, for the same power level, a MOX fuel assembly has a lower thermal neutron flux than a LEU fuel assembly. This results in a lower fission chamber signal from thermal neutrons in a MOX fuel assembly, as noted in Section 3.2.5.

However, the gamma flux in a MOX fuel assembly is similar in magnitude to the gamma flux in an LEU fuel assembly of the same power level. This is because the gamma fluxes in both MOX fuel and LEU fuel are proportional to the fission rate, which is similar in the two fuel types for the same power level.

Therefore, in the case of side-by-side MOX and LEU fuel assemblies at the same power level, the neutron signal component of the total MOX fuel fission chamber signal will be lower, because of the lower thermal neutron flux. The gamma signal component will be approximately the same for MOX fuel and LEU fuel. Accordingly, the ratio of gamma signal component to neutron signal component is higher for MOX fuel – i.e., the relative contribution of the gamma signal component to the total signal, although still small, is greater.

The process of developing the measured power distribution requires that all detector signals be normalized to total core power. If all signals have the same ratio of gamma signal component to neutron signal component, then the normalization process ensures that the gamma signal component does not affect the relative power measurement in each location. This is the case for all-LEU fuel cores, because the relative importance of the gamma signal component to the total signal is the same throughout the core. The same would be true of an all-MOX fuel core.

In a mixed core of MOX and LEU fuel assemblies, the relative importance of the gamma signal is slightly higher in MOX fuel. Absent any bias, the normalization process would result in higher relative powers in MOX fuel locations.

Duke had discussions with representatives of several foreign organizations that have contemporary experience modeling partial MOX fuel cores in reactors with Westinghouse-type incore instrumentation systems. These discussions confirmed that the standard practice is to apply a negative bias to the MOX fuel signals prior to normalization.

Duke used detailed analyses of the incore fission chambers in MOX and LEU fuel assembly lattices to establish the magnitude of the bias. The MOX fuel bias was chosen to restore the same ratio of gamma signal component to the total signal in MOX fuel as in the LEU fuel, as described below.

Coupled neutron/gamma MCNP models of an incore instrument in a MOX fuel lattice and in a LEU fuel lattice were used to determine the ratio of detector gamma signal component to the total signal. The analysis indicated that for a given detector signal in a MOX fuel assembly, a [ ]$_D$ reduction in the total signal would yield the same relative contribution from gamma ionization as was predicted in LEU fuel. This bias would enable the normalized core power distribution to be calculated in a consistent manner with both MOX and LEU fuel.

The bias was then validated against the St. Laurent B1 benchmark data. St. Laurent power distribution analyses were performed with and without the [ ]$_D$ bias. The impact on the St. Laurent observed nuclear reliability factors (ONRFs) is illustrated in the following table (note: the same St. Laurent ONRFs with the [ ]$_D$ MOX bias are also reported in Table 3-12 of DPC-NE-1005). These results indicate that the MOX fuel bias has a minor overall beneficial impact on the calculated power distribution uncertainty factors in MOX and LEU fuel.

	[ ]D MOX bias		No MOX Bias	
	LEU	MOX	LEU	MOX
FΔh	[	]D	[	]D
Fq	[	]D	[	]D
Fz	[	]D	[	]D

It should be noted that the St. Laurent ONRFs are not used directly in the calculation of the proposed uncertainty factors in Section 5. The St. Laurent ONRFs support the conclusion that power distribution uncertainties for MOX and LEU fuel locations are similar. The final uncertainties conservatively utilize McGuire and Catawba ONRFs for both LEU and MOX fuel.

It should also be noted that MOX fuel and LEU fuel power predictions from SIMULATE-3 MOX will be used directly (without any bias) in the reload design process to ensure that core designs meet peaking limits. The bias is applied only in the processing of measured incore power distributions from the incore detectors.

The bias to measured incore signals may be adjusted as additional data is obtained from the MOX fuel lead assembly program, and from mixed cores of LEU and MOX fuel at McGuire and Catawba.

10. Also, in the second paragraph from the bottom of page 3-9, it is stated that conversion factors were applied. What conversion factors? How are these conversion factors calculated and when are they applied?

Response:

Conversion factors are factors that translate the measured incore detector signals into a measured relative power distribution (they are referred to as "INCORE constants" in the original Westinghouse methodologies). The electrical signals collected by incore fission chambers are proportional to the thermal neutron flux in the instrument tube at the center of the fuel assemblies. However, the desired parameter is not the flux at the center of the fuel assembly, but the average fuel assembly power. Conversion factors are required to translate the measured parameter (thermal flux) to the desired parameter (assembly power).

These conversion factors are calculated using data generated by the core simulator code – in this case, SIMULATE-3 MOX. Axially-dependent conversion factors are determined for each assembly in the core. The conversion factors are derived from cycle specific core models for various burnups with control rods present or absent.

and 5 of the report "*Evaluation of the Effects of Mixed LEU-MOX Core on Dynamic Rod Worth Measurement*", North Carolina State University, February 2001. The simulations discussed in this report quantify the impact of mixed LEU-MOX cores and demonstrate that the existing DRWM methodology can be used to accurately measure control bank worths in partial MOX fuel cores. A copy of the report is included with this submittal.

14. The two paragraphs on page 6-3 also indicate that the presence of MOX does not impact the excore detector signal. Yet no data is provided to support this claim. Please provide quantitative technical justification (results) to support this assertion.

Response:

Section 4 of the North Carolina State University (NCSU) Dynamic Rod Worth Measurement (DRWM) report examines the impact on the excore detector signal from a slightly harder neutron spectrum and a decrease in the core average delayed neutron fraction produced by MOX fuel. The simulation results in the NCSU report demonstrate that the presence of MOX fuel does not significantly impact the excore detector signal.

15. Section 6.3 addresses the issue of model sensitivity of the dynamic rod worth measurement to the inaccuracies in the computer models. Please provide sensitivity study results for staff review.

Response:

Section 7 of the NCSU DRWM report provides results of sensitivity studies of the deduced bank worth error due to errors in the core simulator model.

16. The third paragraph on page 2-4, states that SIMULATE-3 MOX supplements the polynomial expansion method with additional terms derived from purely analytic nodal solution methods. Please provide additional details on how this is accomplished.

Response:

For details of the analytic terms in the nodal solution model, see the responses detailed in item "j" of Question 1, and the detailed discussion in Section 2 (pages 2-3) of SSP-00/420, "SIMULATE-3 MOX Enhancements and Verification Tests" (Reference 12).

17. In several places in the document a statement is made that the new models yield results consistent with the results of the conventional methods in LEU cores. For every occasion where this statement is made demonstrate that this statement is true. Provide graphics and commentary for each occasion where the statement is made.

Response:

In the response to Question 7, Table 1 compares the fidelity of the new models to that of the currently approved methodology, which uses CASMO-3 / SIMULATE-3. Comparisons of average and standard deviations are provided for BOC HZP soluble boron concentration, HFP soluble boron concentration, BOC HZP control rod worth, and BOC HZP isothermal temperature coefficient.

Table 3 below compares two sets of power distribution uncertainty factors (referred to in the topical report as Observed Nuclear Reliability Factors or ONRFs) calculated with CASMO-3/SIMULATE-3 (Sets A & B) to a set of ONRFs calculated with CASMO-4 / SIMULATE-3 MOX (Set C) from the topical report (Table 3-12). The ONRFs in Sets A and B are from previous benchmark calculations on McGuire and Catawba cores using conventional or previous methodologies. Comparison of the ONRFs in Sets A and B, with the corresponding ONRFs in Set C shows that the results obtained from the new models are consistent with those from previous methods.

Table 3
ONRF Comparisons

	Set A	Set B	Set C
Parameter			
$F\Delta h$	1.017	1.020	[ ]D
Fq	1.057	1.037	[ ]D
Fz	1.053	1.031	[ ]D

Set A – DPC-NE-1004P-A Rev 0 - Mk BW fuel, 12 axial levels, no axial blankets
Set B – DPC-NE-1004P-A Rev 1 - Mk BW fuel, 24 axial levels, no axial blankets
Set C – DPC-NE-1005P Rev 0 - Mk BW fuel and Westinghouse RFA fuel
(Mk BW fuel - 24 axial levels, axial blankets)
(Westinghouse RFA fuel - 24 axial levels, axial blankets)

18. In the first paragraph on page 2-5, the document discusses the spatial homogenization error that SIMULATE-3 MOX reduces by recalculating. Please provide a detailed discussion of how this recalculation is accomplished and why it is conservative.

Response:

For details of the spatial homogenization model, see the responses detailed in item "l" of Question 1, and the detailed discussion in Section 2 (pages 3-4) of SSP-00/420, "SIMULATE-3 MOX Enhancements and Verification Tests" (Reference 12).

With respect to the question of why the re-homogenization correction is conservative, it

should be noted that the re-homogenization model (and all other models) for MOX fuel assemblies is not designed to be conservative but rather is designed to be as accurate as possible.

19. In the first paragraph on page 4-1 of Reference 23, it is stated that the fuel assembly design is similar to the design proposed for use by Duke. Please provide details including quantifying how similar the designs are, both from a mechanical and neutronic standpoint.

Response:

Reference 23, "CASMO-4 Benchmark Against Critical Experiments," SOA-94/12, includes a side-by-side comparison of a CASMO-4 model and an MCNP model of a MOX fuel assembly. The MOX fuel assembly modeled was similar to the design proposed by Duke in that it was based on a 17 x 17 Westinghouse PWR fuel assembly with MOX fuel pins near the center of the fuel assembly at a higher plutonium concentration than pins on the outside of the assembly. The intended point in referencing this document was to show that in a typical MOX fuel assembly the fission rate (power) calculated by CASMO-4 and by MCNP are in good agreement, within [0.8% RMS]$_S$.

20. Please provide two copies of all proprietary, non-NRC reviewed references. Please note that proprietary information must be accompanied by an affidavit that identifies the document or part to be withheld and that meets the other requirements of the Commission's regulations in 10 CFR 2.790, "Public inspections, exemptions, requests for withholding."

Response:

The submittal package for this RAI includes two copies of the following proprietary references from the topical report. Note: the numbering of each document corresponds to the reference number in the topical report. As noted in the transmittal letter, References 19 and 20 are not provided with this package because the EDF proprietary affidavit has not yet been received by Duke Power. Those references will be provided as soon as the affidavit is available.

8) Dave Knott, Bengt H. Forssen, Malte Edenius, "CASMO-4, A Fuel Assembly Burnup Program Methodology," Proprietary, SOA-95/2, STUDSVIK of America, Inc., USA, STUDSVIK Core Analysis AB, Sweden, September 1995.

9) Malte Edenius, Kim Ekberg, Bengt H. Forssen, Dave Knott, "CASMO-4, A Fuel Assembly Burnup Program, User's Manual," Proprietary, SOA-95/1, STUDSVIK of America, Inc., USA, STUDSVIK Core Analysis AB, Sweden, September 1995.

10) Tamer Bahadir, Jerry A. Umbarger, Malte Edenius, "CMS-LINK_DUKE User's Manual," Proprietary, SSP-99/403, Revision 0.

11) Arthur S. DiGiovine, Joel D. Rhodes, III, Jerry A. Umbarger, "SIMULATE-3, Advanced Three-Dimensional Two-Group Reactor Analysis Code, User's Manual," Proprietary, SOA-95/15, STUDSVIK of America, Inc., USA, October 1995.

12) Kord S. Smith, Joel D. Rhodes, Scott Palmtag, "SIMULATE-3 MOX Enhancements and Verification Tests," Proprietary, SSP-00/420, STUDSVIK SCANDPOWER, Inc., June 2000.

13) Kord S. Smith, David J. Kropaczek, Jerry A. Umbarger, "SIMULATE-3 Kinetics Input Specification," Proprietary, SOA-98/12, Revision 0, STUDSVIK SCANDPOWER, Inc., July 1998.

14) Kord S. Smith, David J. Kropaczek, Jeffrey A. Borkowski, Jerry A. Umbarger, "SIMULATE-3 Kinetics Models and Methodology," Proprietary, SOA-98/13, Revision 0, STUDSVIK SCANDPOWER, Inc., July 1998.

21) David G. Knott, Malte Edenius, "CASMO-4 Benchmark Against Critical Experiments", Proprietary, SOA-94/13, Studsvik of America, Inc., USA, 1994.

23) David G. Knott, Malte Edenius, "CASMO-4 Benchmark Against MCNP", Proprietary, SOA-94/12, Studsvik of America, Inc., USA, 1994.

27) EPICURE Experiments, EPD-DC-293, Revision 0 (Proprietary), FRAMATOME, June 8, 1999.

28) "Experience EPICURE UMZONE Distribution Fine de Puissance en Presence d'une Grappe de 24 Crayons Absorbants B4C dans l'Assemblage MOX Zone Central et Effet d'Ombre 1, 9, 24 absorbants B4C," NT-SPRC-LPEx-93/124, Revision A (Proprietary), FRAMATOME, August 2, 1994.

29) "Experience EPICURE UMZONE Distribution Fine de Puissance en Presence d'une Grappe de 24 Crayons Absorbants AIC dans l'Assemblage MOX Zone Central," NT-SPRC-LPEx-92/78, Revision A (Proprietary), FRAMATOME, February 19, 1993.

30) "EPICURE Results of the Material Buckling Measurements in the MH1.2-93 Configuration," Framatome letter EPD/99.1183, Revision A, {Appendix A} from S. Tarle (FRAMATOME) to FRAMATOME COGEMA Fuels (Attention: George Fairburn, et. al) (Proprietary), November 3, 1999.

31) "Rapport d'Experience Programme EPICURE: Configuration UM 17x17/7% Mesures de la Distribution Fine de Puissance et des Rapports d'Activite d'une Chambre a Fission dans les Assemblages MOX et UO2 Adjacents," NT-SPRC-LPEx-95-025, Revision 0 (Proprietary), FRAMATOME, February 23, 1995.

32) "Programme EPICURE - Configuration UM17x17/11% Rapport d'Experience," NT-SPRC-LPEx-95-021, Revision 0 (Proprietary), FRAMATOME, February 23, 1995.

33) "Experience ERASME/L Description Geometrique et Bilan Matiere," SEN/LPRE n° 87-289 (Proprietary), FRAMATOME, June 1987.

34) "Resultats des Mesures D'effets en Reactivite et de Distributions de Puissance sur des Configurations avec une Grappe de 9 Crayons B4C Naturel dans le Cadre de L'experience ERASME/L," NTC-SPRC-LPEx-90/102 (Proprietary), FRAMATOME, June 14, 1990.

----------End of List----------

21. The second paragraph on page 4-7 discusses the EPICURE experiments. It is mentioned that the experiments used a fuel pin layout that is comparable to the Duke MOX fuel assembly layout. Please provide additional details to support this statement.

Response:

The statement refers specifically to the UMZONE No BP, UMZONE B_4C, and UMZONE AIC experiments, which are illustrated in Figures 4-18, 4-19 and 4-20. As depicted in Figure 4-17, the MOX region is a 17 x 17 layout with 24 guide tubes and one instrument tube in the central region of the EPICURE core, a configuration virtually identical to that of a McGuire/Catawba 17 x 17 fuel assembly. The EPICURE experiments have a pin pitch of 1.26 cm which is nearly identical to that of the planned MOX fuel design (1.265 cm). The MOX fuel region consists of three concentrations of MOX fuel with the lowest concentration on the outside of the lattice and the highest MOX concentration in the central part of the lattice, which is the same configuration as that in the planned MOX fuel assembly design. Also, the MOX fuel lattice in these experiments is surrounded by a buffer region of LEU fuel with a Westinghouse 17 x 17 type pin layout, which is typical of the condition that would exist in a mixed MOX/LEU fuel core. Table 4-1 provides additional comparison information for the fuel assembly

design.

22. Please provide all documentation and the code for CASMO-4 and SIMULATE-3. This entails all code documentation, including user guides, model and methods description, verification and validation, and the source codes as well as executables of the codes.

Response:

Duke is providing the requested CASMO-4 and SIMULATE-3 documentation as a part of the response to Question 20. Duke and the computer code owner, Studsvik Scandpower, Inc., have identified several issues associated with the request for the source and executable codes. These issues have been discussed with the NRC staff. Duke intends to continue to working with the NRC staff to identify an arrangement that will enable the NRC staff to perform its review, while at the same time addressing the Duke and Studsvik concerns.

23. Please provide a discussion of the differences between weapons-grade and reactor-grade MOX fuel. Provide a specific basis for why the data for reactor-grade MOX fuel is adequate for weapons-grade MOX fuel and quantify the differences between the fuel types.

Response:

The potential impacts of differences between MOX fuel derived from weapons grade plutonium and MOX fuel derived from reactor grade plutonium are addressed in Section 3 of the Framatome ANP MOX Fuel Design Topical Report (BAW-10238) that was submitted to the NRC for review in April 2002. The portion of the discussion that is relevant to neutronic performance is repeated below.

Beginning of BAW-10238 information

The characteristics and behavior of MOX fuel derived from weapons grade (WG) plutonium is bounded by the experience base with MOX fuel derived from reactor grade (RG) plutonium. The MOX fuel is characterized in terms of plutonium isotopics as RG or WG. Typical plutonium isotopic concentrations for WG and RG plutonium are compared in Table 3.1. It can be seen that the WG material has a much higher percentage of fissile material (^{239}Pu and ^{241}Pu) compared to the RG material, thus allowing lower plutonium concentrations with WG material to achieve the same total energy extraction. The fuel characteristics, as a function of burnup, of the MOX fuel derived from WG plutonium are bounded by the range of fuel characteristics of LEU fuel and of MOX fuel derived from RG plutonium. This is due to the lower concentration of ^{239}Pu in the MOX fuel derived from WG plutonium relative to the MOX fuel derived from RG plutonium.

RG plutonium is produced from reprocessed spent LWR uranium-based fuel that has been irradiated to commercial burnups, typically in the range of 30,000 to 50,000 MWd/MTU. The plutonium isotopes produced at these burnups, and extracted following irradiation, include significant percentages of ^{240}Pu, ^{241}Pu, and ^{242}Pu. The WG plutonium is created from irradiating ^{238}U to very low burnups and separating the plutonium before substantial percentages of the heavier plutonium isotopes build up. Whereas the RG material typically has 24% ^{240}Pu, the WG material is limited to less than 7% ^{240}Pu. These differences in isotopics are readily addressed through the appropriate analytical model. See Table 3.2 for typical plutonium isotopic composition of WG and RG material.

The use of WG plutonium significantly reduces the PuO_2 content of MOX fuel relative to RG material. The WG material is about 95% fissile, whereas the RG material contains significant amounts of absorber isotopes (^{240}Pu and ^{242}Pu). Thus, MOX fuel from RG material can require plutonium contents as high as 8% to 9%.

In LWRs, LEU fuel, RG MOX fuel, and WG MOX fuel all produce power as a result of nuclear fissions induced by a neutron field. For all three fuel types, the fissions occur primarily due to capture of thermal neutrons by uranium and/or plutonium. Both conventional LEU fuel and WG MOX fuel can be thought of as clean fuels. When initially loaded, both fuels produce power primarily from the fission of one isotope (^{235}U for LEU fuel, ^{239}Pu for WG MOX fuel). Both fuels have relatively small amounts of heavy parasitic isotopes in their composition. In contrast, RG MOX fuel contains important quantities of poisoning isotopes that complicate calculations. Due to the presence of the parasitic fertile plutonium isotopes, a RG MOX fuel assembly will require significantly more plutonium than a WG MOX fuel assembly with the same reactivity.

Table 3.2 and Table 3.3 show representative characteristics of unirradiated LEU, WG MOX, and RG MOX fuel assemblies with the same fuel mechanical design. The initial uranium enrichments and plutonium concentrations were chosen to produce an equivalent reactivity at approximately 20,000 MWd/t burnup. The tables show that all three fuel types are predominantly uranium. The plutonium mass (for both total and individual isotopes) of the WG MOX fuel assembly falls between that of the LEU fuel assembly and that of the RG MOX fuel assembly.

As nuclear fuel is used, the elemental and isotopic constituents of the fuel change. For LEU fuel, ^{235}U is depleted, plutonium is produced, and the isotopics of the plutonium evolve. The LEU fuel plutonium isotopics are initially similar to unirradiated WG MOX fuel, but they rapidly evolve toward RG MOX fuel. For WG MOX fuel, plutonium is depleted, and the isotopics of the plutonium evolve toward unirradiated RG MOX. For RG MOX fuel, the plutonium is depleted, and the isotopics of the plutonium further degrade (i.e., a progressively lower percentage of fissile plutonium). These

characteristics are shown on Figure 3.1, Figure 3.2, and Figure 3.3.

As a result of the changes described above, the source of fissions changes markedly with burnup for LEU fuel. However, both RG MOX and WG MOX fuel have little thermally fissionable uranium, so the fissions in both MOX fuel types are approximately 90% plutonium at any burnup. This effect is shown on Figure 3.4.

The reactivity change of the fuel with burnup results from the change in elemental and isotopic composition. Depletion of ^{235}U and fissile plutonium (^{239}Pu and ^{241}Pu) reduces reactivity, as does buildup of fertile plutonium (^{240}Pu). Conversely, buildup of fissile plutonium and depletion of fertile plutonium increase reactivity. The net result of these factors on the fuel neutronic performance is illustrated in Figure 3.5, which shows the infinite multiplication factors (k_∞) of LEU, RG MOX, and WG MOX fuel assemblies as a function of burnup. LEU fuel reactivity decreases most steeply with burnup, while RG MOX fuel decreases the least. WG MOX fuel behavior lies between that of LEU fuel and RG MOX fuel.

Several important points can be made relative to the different fuel types discussed above.

- LEU fuel, RG MOX fuel, and WG MOX fuel are fundamentally similar and, from a neutronic perspective, differ due to the relative amounts of various fissionable and fertile isotopes of uranium and plutonium.
- Significant plutonium fissions occur in medium- and high-burnup LEU fuel.
- RG MOX fuel has higher initial concentrations of heavy plutonium isotopes than WG MOX fuel. For the same reactivity, the amount of plutonium in RG MOX fuel is significantly greater than the amount of plutonium in WG MOX fuel.
- The reactivity behavior of WG MOX fuel as a function of burnup is between that of LEU fuel and that of RG MOX fuel.

Some important conclusions can be drawn from these points.

- The ability to predict the behavior of cores loaded initially with all-uranium fuel requires the capability to model plutonium fuel behavior.
- RG MOX fuel presents a greater challenge to neutronic modeling methods than WG MOX fuel.
- WG MOX fuel characteristics as a function of burnup are generally bounded by LEU fuel and RG MOX fuel.

Thus it can be concluded that nuclear analysis methods that are demonstrated to model LEU fuel and RG MOX fuel with an acceptable accuracy should also be capable of modeling WG MOX fuel with a similar level of accuracy. This is the approach that has been used by Duke to qualify the CASMO-4 and SIMULATE-3MOX computer codes for application to WG MOX fuel analyses.

Table 3.1 Typical Plutonium Isotopics for the Most Abundant Isotopes

Plutonium Isotope	WG (wt %)	RG (wt %)
^{238}Pu	0.0	1.0
^{239}Pu	93.6	59.0
^{240}Pu	5.9	24.0
^{241}Pu[1]	0.4	10.0
^{242}Pu	0.1	5.0
^{241}Am[1]	0.0	1.0

Table 3.2 Sample Unirradiated Nuclear Fuel Composition

	Mass (kg)		
	LEU	RG MOX	WG MOX
Heavy Metal Loading	458.0	458.0	458.0
Total Uranium	458.0	424.6	438.0
^{235}U	18.3	1.1	1.1
^{238}U	439.5	423.5	436.9
Total Plutonium	0.0	33.0	20.0
^{239}Pu	0.0	22.2	18.7
^{240}Pu	0.0	6.9	1.3
^{241}Pu	0.0	2.6	0.0
^{242}Pu	0.0	1.0	0.0

NOTE: Any discrepancy in the total heavy metal loading is due to the presence of trace quantities of ^{234}U and ^{238}Pu.

1 Amount varies with decay time.

Table 3.3 Sample Unirradiated Nuclear Fuel Isotopics

Isotope	Isotopic Fractions		
	LEU	RG MOX	WG MOX
^{235}U	4.0%	0.25%	0.25%
^{238}U	96.0%	99.75%	99.75%
^{239}Pu	0.0%	67.3%	93.3%
^{240}Pu	0.0%	21.0%	6.5%
^{241}Pu	0.0%	7.8%	0.1%
^{242}Pu	0.0%	3.0%	0.1%

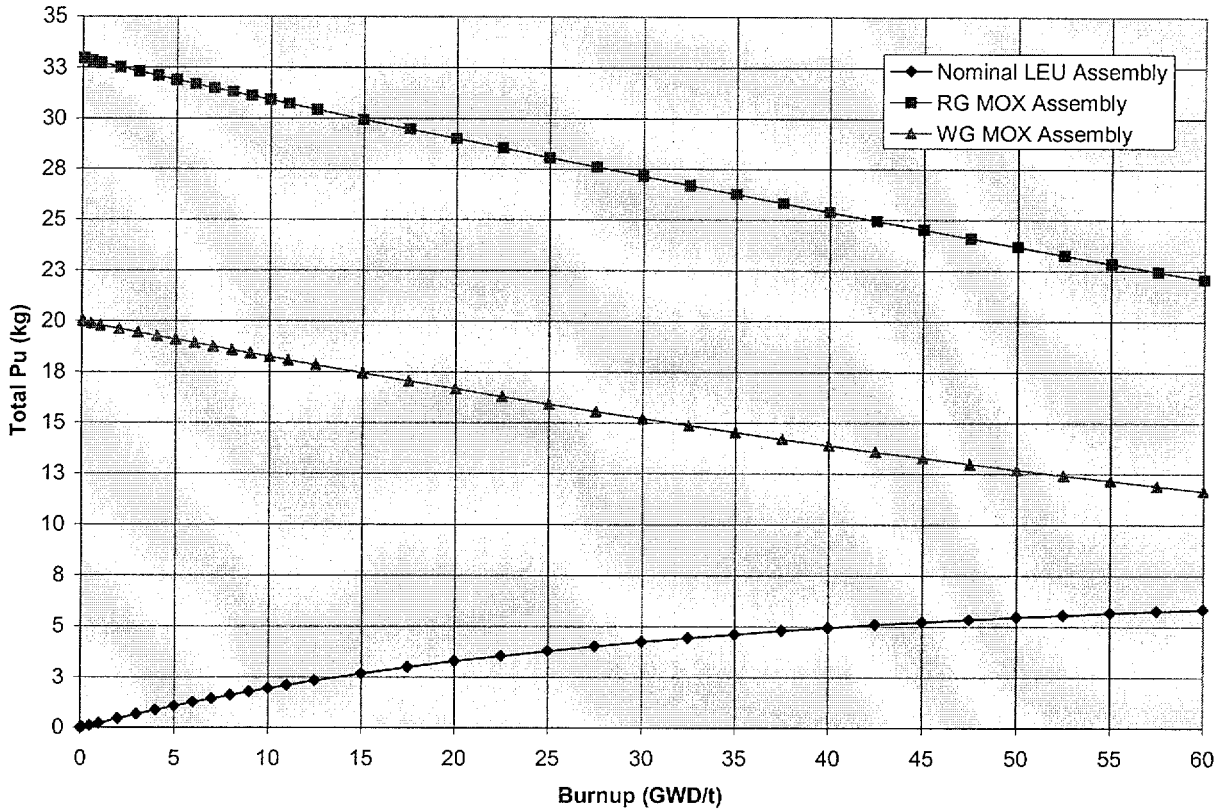

Figure 3.1 Total Plutonium Mass

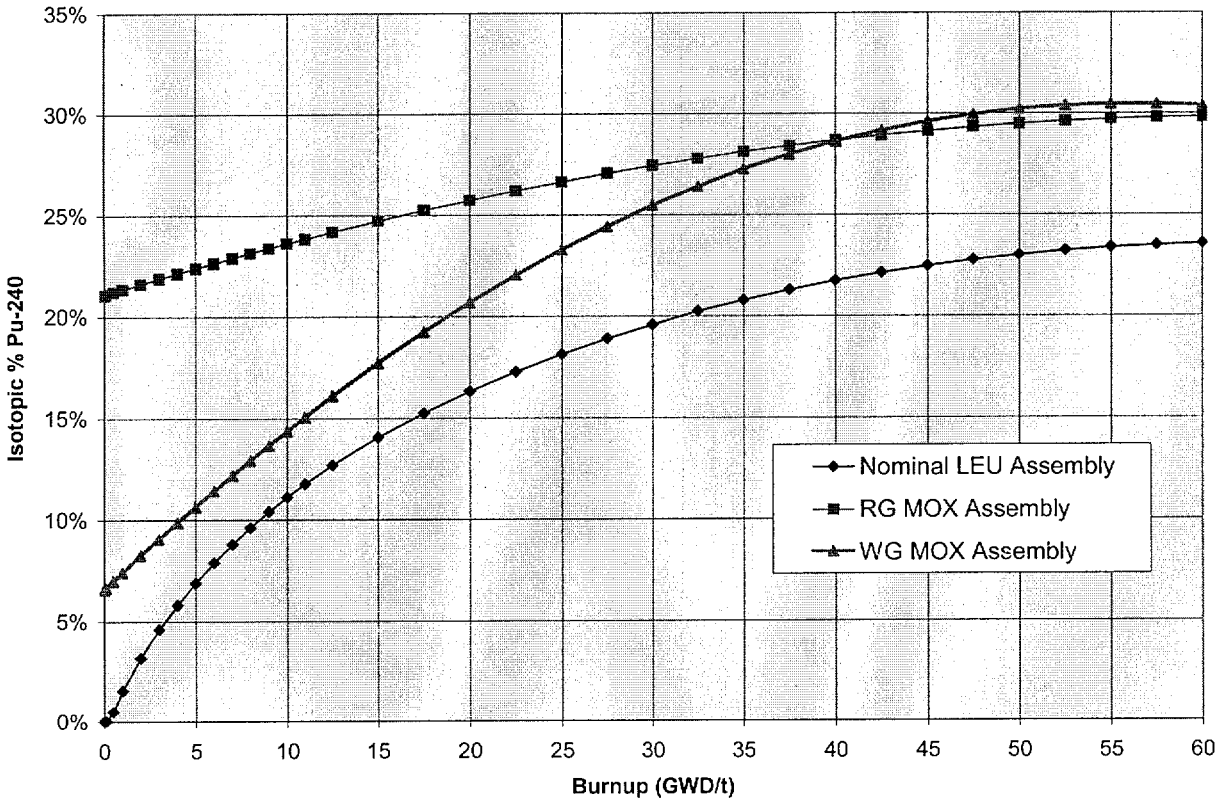

Figure 3.2 ^{240}Pu Concentration

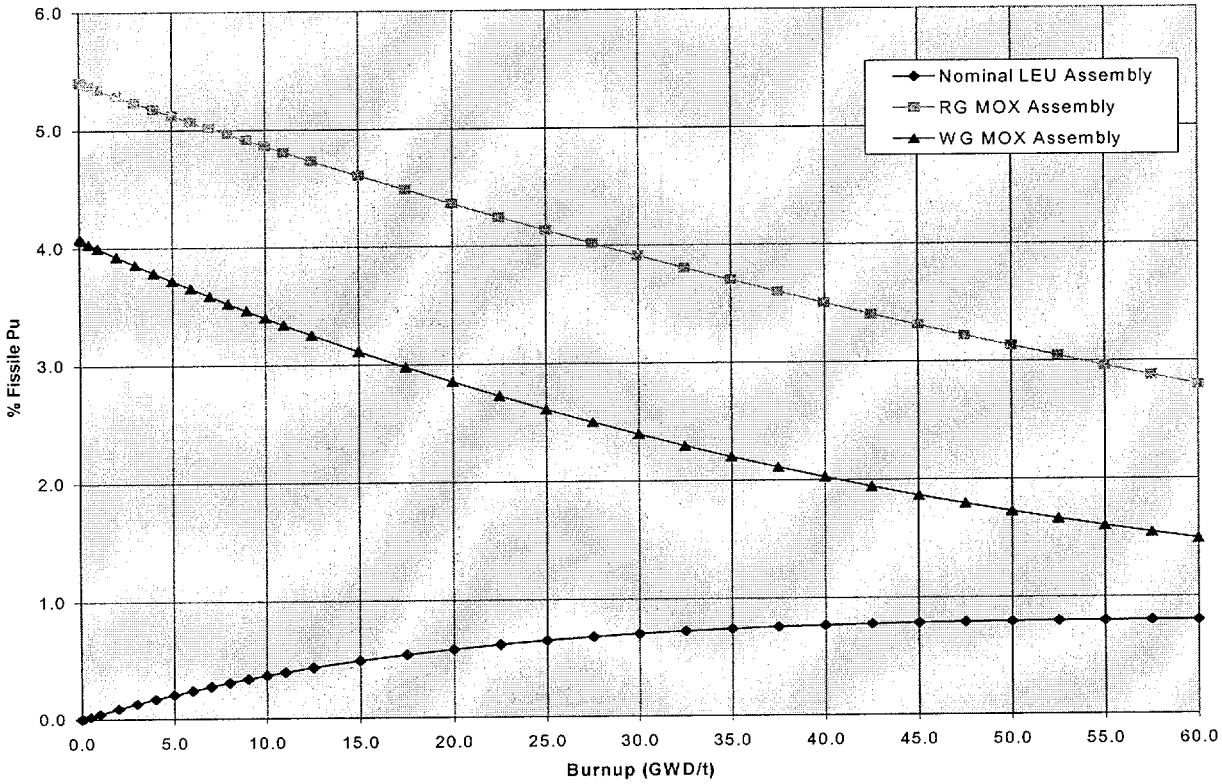

Figure 3.3 Fissile Plutonium

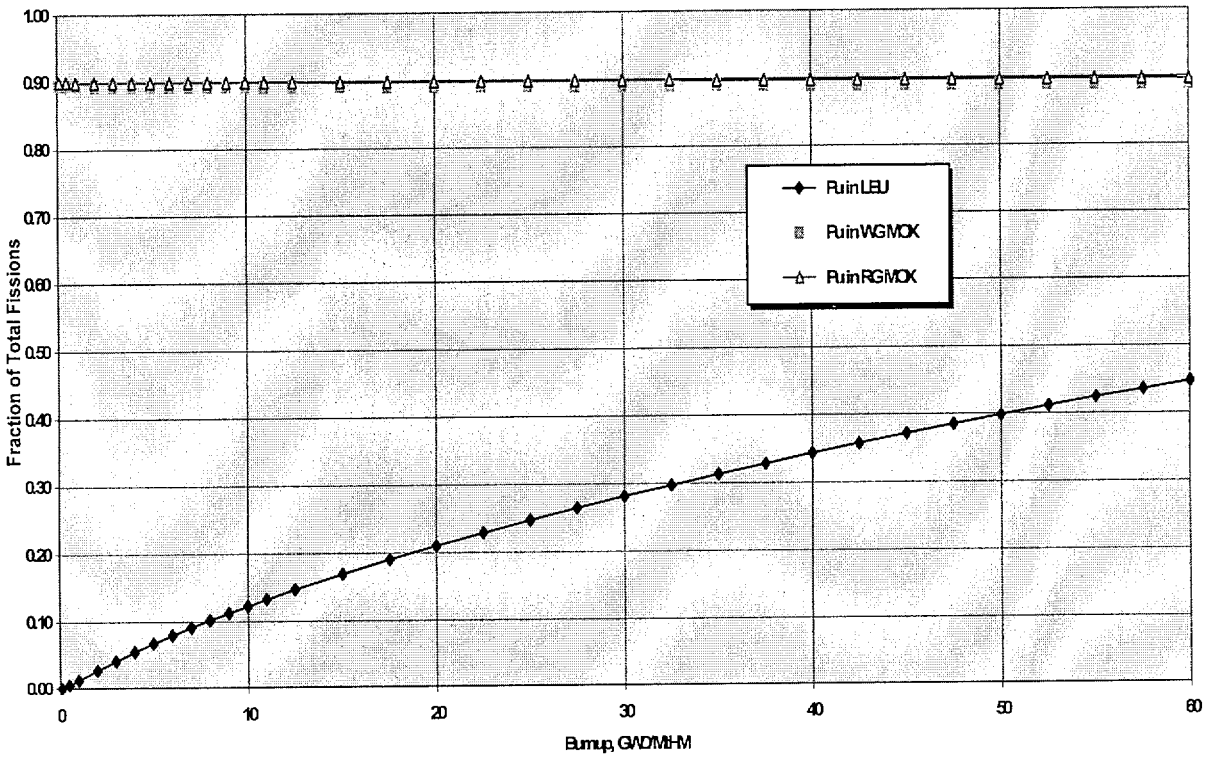

Figure 3.4 Plutonium Fissions – Fraction of Total Fissions

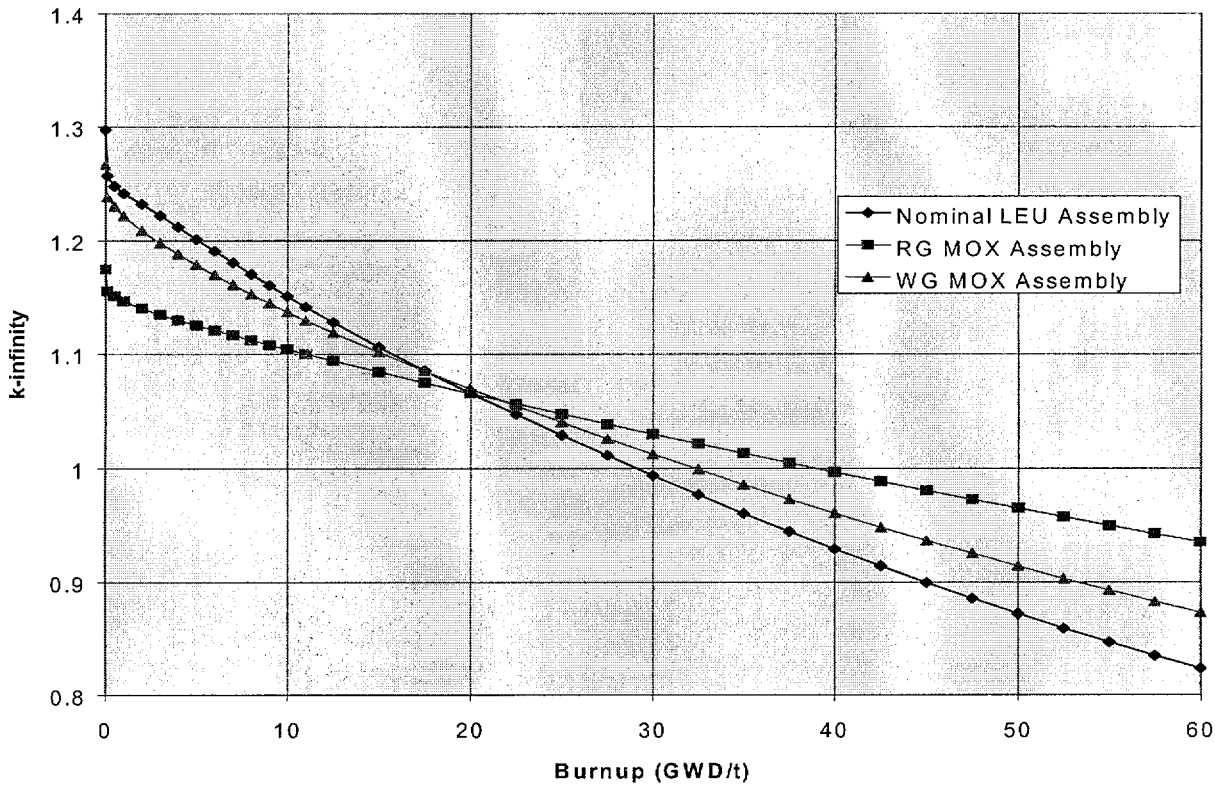

Figure 3.5 k_{oo} vs. Burnup

End of BAW-10238 information

In addition to the information presented above from BAW-10238, it should be noted that the CASMO-4 computer code was used to benchmark critical experiments with a range of plutonium concentrations and isotopics, as shown in Table 4-1 and Table 4-5 of DPC-NE-1005P. The pin power uncertainty calculated by Duke is based on the combined Saxton, EPICURE, and ERASME data set, as shown in Table 4-8. The Saxton critical experiments, in particular, used MOX fuel derived from plutonium that was very close to weapons grade (91.4% fissile). The ERASME experiments also used fuel with higher fissile plutonium (76%) than the reactor grade St. Laurent B1 MOX fuel (approximately 70%), and the ERASME experiments had a very high total plutonium loading (almost 11% of the heavy metal was plutonium). The benchmark data base of DPC-NE-1005P is not exclusively reactor grade MOX fuel, but includes plutonium isotopics that are very similar to the expected isotopics for McGuire and Catawba applications.

Also, Duke Power intends to deploy weapons grade MOX fuel lead assemblies in one of its McGuire or Catawba units prior to large scale use of MOX fuel. The lead assembly program will provide an opportunity to compare measured and predicted powers in a weapons grade MOX fuel assembly. These comparisons will provide additional assurance that the DPC-NE-1005P methodology can adequately predict power in weapons grade MOX fuel.

In conclusion, methods that adequately model both LEU fuel and reactor grade MOX fuel are quite capable of modeling weapons grade MOX fuel, because:

1) The characteristics of weapons grade MOX fuel are similar to both LEU fuel and reactor grade MOX fuel.
2) The nuclear performance of weapons grade MOX fuel (e.g., k_∞ vs. burnup) is generally bounded by LEU fuel and reactor grade MOX fuel.
3) At end of life LEU fuel contains significant amounts of plutonium, so in order to accurately model cores with LEU fuel, it is necessary to accurately model the behavior of that plutonium.
4) Modeling reactor grade MOX fuel is more complicated than modeling weapons grade MOX fuel due to the greater plutonium isotopic variation in fresh reactor grade MOX fuel.
5) The benchmark data base used in DPC-NE-1005P contains a range of plutonium isotopics and concentrations, including fuel with near-weapons grade isotopics and fuel with high plutonium concentrations.

Finally, the MOX fuel lead assembly program will provide additional assurance that the DPC-NE-1005P methodology can adequately model weapons grade MOX fuel.

End of BAW-10238 information

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