Duke Power®

Duke Power

Nuclear Design Methodology Using
CASMO-4 / SIMULATE-3 MOX

DPC-NE-1005-NP-A
Revision 0

Submitted to NRC August 2001

Approved by NRC August 2004

PROPRIETARY NOTICE

Certain data in this report are proprietary to various companies, as noted below.

McGuire/Catawba data	Duke Power
St. Laurent B1 data	Electricite de France
EPICURE and ERASME data	Framatome ANP
Calculation results and statistics	Duke Power
Description of analytical models	Studsvik Scanpower

Proprietary data is bracketed and identified with a subscript indicating which company considers the data proprietary. Where the information is proprietary to more when than one company, multiple subscripts are used. D = Duke Power, E = Electricite de France, F = Framatome ANP, S = Studsvik Scanpower

Page Intentionally Left Blank

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 20, 2004

Mr. H. B. Barron
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church Street
Charlotte, NC 28202

SUBJECT: FINAL SAFETY EVALUATION FOR DUKE TOPICAL REPORT DPC-NE-1005P, "NUCLEAR DESIGN METHODOLOGY USING CASMO-4/SIMULATE-3 MOX"

Dear Mr. Barron:

Enclosed is a copy of the U.S. Nuclear Regulatory Commission staff's Safety Evaluation (SE) for Topical Report DPC-NE-1005P, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX."

A draft of this SE was provided to you by letter dated February 20, 2004. By letter dated March 9, 2004, you provided comments on the draft SE. This final SE responds to those comments and issues the SE in final form. Your letter also stated that the draft SE contained no proprietary information.

In the event of any comments or questions, please contact me at (301) 415-1493.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager
Project Directorate II-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370, 50-413, 50-414

Enclosure: As stated

cc w/encl: See next page

McGuire Nuclear Station
Catawba Nuclear Station

cc:

Ms. Lisa F. Vaughn
Duke Energy Corporation
Mail Code - PB05E
422 South Church Street
P.O. Box 1244
Charlotte, North Carolina 28201-1244

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. C. Jeffrey Thomas, Manager
Regulatory Compliance
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U. S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

Dr. John M. Barry
Mecklenburg County
Department of Environmental Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of Environment,
Health, and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. R. L. Gill, Jr.
Manager - Nuclear Regulatory
Issues and Industry Affairs
Duke Energy Corporation
526 South Church Street
Mail Stop EC05P
Charlotte, North Carolina 28202

NCEM REP Program Manager
4713 Mail Service Center
Raleigh, North Carolina 27699-4713

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Mary Olson
Director of the Southeast Office
Nuclear Information and Resource Service
729 Haywood Road, 1-A
P. O. Box 7586
Asheville, North Carolina 28802

McGuire Nuclear Station
Catawba Nuclear Station

cc:

Lee Keller
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626-0513

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Henry Porter, Assistant Director
Division of Waste Management
Bureau of Solid and Hazardous Waste
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

North Carolina Electric Membership Corp.
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
4830 Concord Road
York, South Carolina 29745

Mr. Dhiaa Jamil
Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Mr. G. R. Peterson
Vice President
McGuire Nuclear Station
Duke Energy Corporation
2700 Hagers Ferry Road
Huntersville, North Carolina 28078

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT DPC-NE-1005P

NUCLEAR DESIGN METHODOLOGY USING CASMO-4/SIMULATE-3 MOX FOR

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS: 50-413, 50-414, 50-369 AND 50-370

1.0 INTRODUCTION

The Duke Power Company (Duke or licensee) submitted by letters dated August 3 (Proprietary), and August 6, 2001 (Non-proprietary), the Topical Report DPC-NE-1005P, Revision 0, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX [mixed oxide]," for review by the U.S. Nuclear Regulatory Commission (NRC) staff. Duke is the license for the Catawba Nuclear Station (Catawba), Units 1 and 2, and the McGuire Nuclear Station (McGuire), Units 1 and 2. Duke submitted additional letters dated September 12 and November 12, 2002, and June 26, August 14 and December 2, 2003 (References 2, 3, 4, 5 and 6).

The Topical Report addresses the use of the Studsvik Core Management System (Studsvik/CMS) code package to support the reload design analyses for Catawba and McGuire. The Studsvik/CMS code package primarily consists of the CASMO-4 and SIMULATE-3 MOX computer codes. The Topical Report demonstrates the validity and accuracy of the Studsvik/CMS code package at Catawba and McGuire for core reload design, core follow, and calculation of key core parameters for reload safety analysis. The NRC staff's review of the topical report considered the topical report's applicability for the use of low-enriched uranium (LEU) fuel at Catawba and McGuire and the use of up to four MOX lead test assemblies (LTAs) in one of the Catawba units. The NRC staff's review findings are based, in part, on licensee commitments included by Duke in Reference 4 as follows:

1. For a lead assembly program containing four MOX fuel assemblies, Duke will place at least two of the MOX fuel lead assemblies in core locations that are measured directly by the movable incore detector system for the first and second cycles of lead assembly irradiation.

2. Duke will perform the physics test program defined in Table 1 [of Reference 4] for all MOX fuel lead assembly cores and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined [in Reference 4] is reached. Core power levels at which low and intermediate power escalation power distribution maps are taken will be consistent from cycle to cycle for each unit (within ± 3% rated thermal power). Core

power level at which power distribution maps are taken may vary among units and between McGuire and Catawba.

3. Duke will prepare a startup report for each operating cycle with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above [in Reference 4] is reached. Each startup report will contain comparisons of predicted to measured data from the zero power physics tests and the power distribution maps taken during power escalation. The reports will include discussions of any parameter that did not meet acceptance criteria. Duke will provide each report to the NRC within 60 days of measurement of the final power distribution map.

4. Duke will prepare an operating report for each operating cycle with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above [in Reference 4] is reached. Each operating report will contain comparisons of predicted to measured monthly power distribution maps and monthly boron concentration letdown values. Duke will provide each cycle operating report to the NRC within 60 days of the end of the fuel cycle.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 34, "Contents of Applications; Technical Information," requires that safety analysis reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload process, licensees perform reload safety evaluations (SE) to ensure that their safety analyses remain bounding for the design cycle. Licensees confirm that the analyses remain bounding by ensuring that the inputs to the safety analyses are conservative with respect to the current design cycle. They check these inputs by using core design codes and methodologies.

The objective of the nuclear design review for the fuel assemblies, control systems, and reactor core is to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients. The NRC staff acceptance criteria are based on Chapter 4.3, "Nuclear Design," of the Standard Review Plan.

3.0 TECHNICAL EVALUATION

Currently, Catawba and McGuire use the CASMO-3/SIMULATE-3 analytical computer codes and various methodologies. In its submittal, Duke requests replacing its current codes with the newer Studsvik/CMS code package. The CASMO-4, CMS-LINK, SIMULATE-3 MOX, and SIMULATE-3K MOX computer codes comprise the Studsvik/CMS package.

The CASMO-4 computer code is the Studsvik Scandpower, Inc., lattice code. The CASMO-4 computer code, a multi-group two-dimensional transport theory code for depletion and branch calculations for a single assembly, is used to generate the lattice physics parameters. These parameters include the cross sections, nuclide concentrations, pin power distributions and other nuclear data used as input to the SIMULATE-3 MOX program for core performance analyses.

New features of CASMO-4 over CASMO-3 are the incorporation of the microscopic depletion of burnable absorbers into the main calculations, and the introduction of a heterogeneous model for the two-dimensional calculation. Also new in CASMO-4, is the use of the characteristics method for solving the transport equation. When MOX fuel is detected in the input, the code automatically uses a more detailed internal calculation to accommodate the larger variation of plutonium (Pu) cross sections and resonances. Studsvik also supplies the SIMULATE-3 MOX code. This code is a two-group, three-dimensional nodal program based on the NRC staff-approved QPANDA neutronics model that employs fourth-order polynomial representations of the intranodal flux distributions in both the fast and thermal neutron groups. The code is based on modified coarse mesh (nodal) diffusion theory calculational technique, with coupled thermal hydraulic and Doppler feedback. The program explicitly models the baffle/reflector region, eliminating the need to normalize to higher-order fine mesh calculations. It also includes the following modeling capabilities: solution of the two group neutron diffusion equation, fuel assembly homogenization, explicit reflector cross-section model, cross-section depletion and pin power reconstruction. The SIMULATE-3 MOX code uses a more refined solution technique to account for steeper flux gradients that exist between the MOX and LEU fuel interfaces.

In order to insure flux continuity at nodal interfaces and perform an accurate determination of pin-wise power distributions, SIMULATE-3 MOX uses assembly discontinuity factors that are pre-calculated by CASMO-4. These factors are related to the ratio of the nodal surface flux in the actual heterogenous geometry to the cell averaged flux in an equivalent homogeneous model, and are determined for each energy group as a function of exposure, moderator density and control-rod-state.

The two group model solves the neutron diffusion equation in three dimensions, and the assembly homogenization employs the flux discontinuity correction factors from CASMO-4 to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution. The flux discontinuity concept is also applied to the baffle/reflector region in both radial and axial directions to eliminate the need for normalization, or other adjustments at the core/reflector interface.

The SIMULATE-3 MOX fuel depletion model uses tabular and functionalized macroscopic or microscopic, or both cross-sections to account for fuel exposure without tracking the individual nuclide concentrations. Depletion history effects are calculated by CASMO-4 and then processed by the CMS-LINK code for generation of the cross-section library used by SIMULATE-3 MOX.

SIMULATE-3 MOX can be used to calculate the three-dimensional pin-by-pin power distribution in a manner that accounts for individual pin burnup and spectral effects. SIMULATE-3 MOX also calculates control rod worth and moderator, Doppler and xenon feedback effects.

SIMULATE-3K MOX is an extension of SIMULATE-3K, which is used for analysis of core transients. The spatial neutronics models in SIMULATE-3K MOX are identical to those in SIMULATE-3 MOX. SIMULATE-3K MOX solves the transient neutron diffusion equation incorporating effects of delayed neutrons, spontaneous fission in fuel, alpha-neutron interactions from actinide decay, and gamma-neutron interactions from long-term fission product decay. For the applications reviewed in Topical Report DPC-NE-1005P, SIMULATE-3K MOX is used only as part of the dynamic rod worth measurement (DRWM) methodology.

3.1 Model Benchmarking

The licensee's submittal, dated August 3, 2001, compares the CASMO-4/SIMULATE-3 MOX predictions of key physics parameters against plant data and critical experiments. For CASMO-4, this benchmarking encompassed criticality and pin power predictions for LEU and MOX fuel. As part of the development of the Catawba and McGuire models, the licensee compared CASMO-4/SIMULATE-3 MOX calculation predictions to plant and/or experimental data for reactivity worth for soluble boron, burnable poison rods, silver-indium-cadmium control rods, Isothermal temperature coefficient, and core power distribution. The licensee provided documentation that contained the results of benchmarking CASMO-4 results to Monte Carlo code calculations and critical experiments for LEU and MOX fuel assembly designs (References 5 and 6).

The licensee performed comparisons between CASMO-4 MOX predictions and data from three MOX critical experiments: Saxton, EPICURE, and ERASME/L. The results of these comparisons were used in the development of the fuel pin power uncertainties that are part of the overall nuclear uncertainty factors. The Saxton critical experiment used Pu that had an isotopic content that is close to current weapons grade Pu fuel. EPICURE used fuel pins that are similar to current 17x17 pressurized-water reactor fuel pins and emulated the hot condition fuel to moderator ratio. ERASME/L used a fissile Pu concentration of 8.28 percent that bounds the fissile Pu content expected in the Duke reactors. SIMULATE-3 MOX could not model the experiments because of their small configurations; therefore, theoretical problems were developed to test the ability of SIMULATE-3 MOX to replicate the CASMO-4 calculations. This provides greater assurance that the CASMO-4/SIMULATE-3 MOX suite of codes will predict the core parameters for a core containing four MOX LTAs with acceptable accuracy.

The comparison of CASMO-4/SIMULATE-3 MOX predictions to measured data incorporates bias and uncertainty for both the predictions and the measured data. The licensee then used statistical methods to account for these uncertainties. For MOX fuel, these methods accounted for the uncertainty from the CASMO comparisons with data and the uncertainty from the CASMO-4 to SIMULATE-3 MOX comparisons for the theoretical problems. Duke also used the CASMO-4/SIMULATE-3 MOX predictions in combination with the normalized flux map reaction rate comparisons to determine appropriate peaking factor uncertainty factors.

Duke intends to use the CASMO-4/SIMULATE-3 MOX programs in licensing applications, including calculations for core reload design, core follow, and calculation of key core parameters for reload safety analyses of Catawba and McGuire. The licensee used data from the Catawba, Unit 1, operating cycles 11 through 13, Catawba, Unit 2, operating cycles 9 through 11, and McGuire, Units 1 and 2, operating cycles 12 through 14, to benchmark the CASMO-4/SIMULATE-3 MOX models for LEU fuel. Duke also used data from the St. Laurent B1 reactor in France, cycles 5 through 10, to benchmark the CASMO-4/SIMULATE-3 MOX models for MOX fuel. These cycles cover core design changes over 17 cycles of operation. Comparison of the St. Laurent parameters to the Catawba and McGuire reactor parameters were provided and demonstrated that the fuel and core parameters important to predicting the core physics response were similar. Loading pattern variations include out-in and low-leakage designs. For model benchmarking, the licensee used critical boron concentration measurements, startup physics testing data, and flux maps. The good agreement between the measured and the calculated values presented in the August 3, 2001, submittal, is used to validate the Duke application of these computer programs for analysis of Catawba and McGuire for LEU and MOX LTAs (maximum four LTAs in one of the Catawba units) fueled cores.

For the parameters compared, the licensee calculated a sample mean and standard deviation of the observed differences. They also determined bias to describe the statistical difference between predicted and reference values.

The St. Laurent reactor uses reactor grade MOX fuel and though similar in composition to the weapons grade MOX fuel, the isotopic composition slightly differs. The Saxton critical experiment uses a Pu isotopic composition that is very close to the weapons grade MOX (90 percent fissile Pu composition.) Both benchmarks demonstrate that the CASMO-4/SIMULATE-3 MOX code can provide close predictions and provides confidence that the code will provide a close prediction of the MOX LTAs. To support future batch implementation, Duke provided a committment in Reference 4 that at least two of the MOX LTAs will be placed in instrumented core locations so that the results from the startup physics tests can be compared to the CASMO-4/SIMULATE-3 MOX predictions to demonstrate the applicability of the codes to analyze LEU and MOX fueled cores. The results of these benchmarks will be submitted to the NRC for review and approval.

The licensee demonstrated that the CASMO-4/SIMULATE-3 MOX models, in conjunction with the indicated reliability factors adequately represent the operating characteristics of Catawba and McGuire. Additionally, Duke did not change key aspects of their core design and analysis methodology, and maintains code and quality assurance practices that provide assurance that future changes to the core, fuel, and burnable poison design will be modeled with accuracy and conservatism. Since the Studsvik/CMS package adequately represents the operating characteristics, the NRC staff finds the use of the Studsvik/CMS package acceptable for Catawba for LEU fuel and up to four MOX LTAs and for McGuire with LEU fuel.

3.2 Statistics

The NRC staff reviewed Duke's application for statistical content. The statistical issues revolved around the 95/95 (probability/confidence) tolerance limit calculations for each parameter of interest. The calculations give 95 percent assurance that at least 95 percent of the population will not exceed the tolerance limit.

The procedure used in the tolerance limits depended on whether the data could be assumed to be distributed normally. The licensee used an established technique for testing normality and assumed normality only if the technique validated that assumption. This approach is acceptable to the NRC staff.

When the normal distribution was applicable, the licensee used the traditional one-sided tolerance calculations. Otherwise, they used a nonparametric method to determine a conservatively large uncertainty (References 9, 10, 11 and 12). Both the parametric and the nonparametric approaches in their proper context are acceptable to the staff.

3.3 Dynamic Rod Worth Measurement

DRWM provides a methodology for the licensee to measure the reactivity worth of the individual control rod banks without changing the boron concentration. The DRWM methodology takes the neutron flux signal from the excore detectors and conditions the excore detector signal through the use of analytical factors to convert the signal into the corresponding rod worth. The SE that approved the Westinghouse DRWM methodology required that anyone applying to use the methodology with their own codes perform calculations comparing their code results to the Westinghouse generated results and that the results must agree within 2 percent or 25 percent mille (pcm) for individual banks, and 2 percent for total bank worth. The acceptance criteria were developed to demonstrate that other parties that used the methodology were applying the codes and methodology correctly. The final test of using the methodology correctly is developing analytical factors that are consistent with the corresponding Westinghouse computations. This consistency is demonstrated by the measured rod worth comparisons.

Duke used the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX codes to generate comparisons to the Westinghouse generated results that used the ALPHA/PHOENIX/ANC codes per the DRWM topical requirements. Duke's analysis showed that 3 percent of the computational results did not meet the criteria. All of the comparisons that did not meet the criteria were for predictions of the rod worth. The comparisons between the measured rod worth CASMO-4/SIMULATE-3 MOX /SIMULATE-3K MOX and the Westinghouse results demonstrated that the analytical factors developed using the CASMO-4/SIMULATE-3 MOX/ SIMULATE-3K MOX code very closely mirror the Westinghouse results. All of the measured rod worth comparisons met the acceptance criteria.

When the underlying causes of the computational results which did not meet the criteria were investigated, it was noted that the predicted and measured rod bank worth deviations were consistent with the differences in the predicted radial Hot Zero Power (HZP) power distribution between Westinghouse and Duke. Relative to the Westinghouse calculation, Duke under-predicts the power of the assemblies on the core periphery which results in a calculated lower rod worth for the associated rod banks (banks SA, CD, SD, and SC) and over-predicts the power of the assemblies in the center of the core which results in a calculated higher rod worth for the associated rod banks (banks CC, CA, and SB.) In all cases where the predicted rod worth computational results did not meet the criteria, Duke predicted a lower bank rod worth that was consistent with the radial power distribution difference between Westinghouse and Duke. Likewise, the impact of the radial distribution caused Duke to consistently calculate a lower total bank worth relative to the Westinghouse calculation since a greater number of rod banks are on the periphery.

The parameter of greatest interest for correct application of DRWM is the calculation of the analytical factor. Correct determination of the analytical factor is shown by close agreement in the measured rod worth comparisons. All of the measured rod worth comparisons met the acceptance criteria. Since all of the measured rod worth comparisons met the acceptance criteria and the deviations in the predicted rod worth comparisons were consistent with the radial power distribution predictions, the NRC staff finds the use of the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX code acceptable for use with the DRWM methodology.

The NRC staff finds the use of the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX methodology acceptable for use with the DRWM methodology for McGuire with LEU fuel and for Catawba with LEU fuel and up to four MOX LTAs.

4.0 RESPONSE TO DUKE'S COMMENTS ON DRAFT SE

Duke's letter dated March 9, 2004, provided comments on the NRC staff's Draft SE. Those comments, and the NRC staff's response to them, are provided below.

Duke Comment No. 1

The NRC has restricted approval of the methodology for the use of up to four MOX fuel LTA at Catawba only. Duke has the following comments concerning this restriction.

(i) NRC has chosen not to provide Duke with the approval that was sought for application of the methodology to partial MOX fuel cores. It is noted that NRC has not provided a technical basis for this action. If NRC restricts the methodology approval to four MOX fuel lead assemblies, Duke requests that NRC indicate in the SE what is considered necessary for extending that approval to larger-scale use of MOX fuel. For example, it could be assumed that NRC anticipates extending the approval to partial MOX fuel cores, provided that the results of the MOX fuel lead assembly program are satisfactory. If NRC has additional expectations beyond the lead assembly program, it is requested that NRC should make those expectations clear. As currently written, the SE provides no clarity on the application of the methodology to partial MOX fuel cores.

NRC Staff Response

Duke's comment, as written, is not completely correct. The NRC staff's approval of the Topical Report extended to the use of the methodology for LEU fuel at Catawba and McGuire and to the use of four MOX LTAs in one of the Catawba units. The basis for the approval of the methodology for MOX LTAs in one of the Catawba units, and not in the McGuire units, is Duke's letter dated September 23, 2003, wherein Duke removed McGuire from the MOX LTA program and indicated that MOX LTAs would be used in one of the Catawba units.

The NRC staff is aware that industry core reload design and analysis practices are continually evolving. Considering the potential changes that may take place between the time of Duke's submittal of the Topical Report and the time of potential use of partial MOX cores, and other information that may be developed on the predictive capabilities of the code package, such as discussed in the sixth paragraph of Section 3.1 above and the following paragraph, the NRC staff elects to delay approval of the methodology for partial MOX fuel cores until such more specific information on the design of partial MOX cores becomes available.

Duke committed to place two LTAs in instrumented locations for the first and second core cycles as required by condition one. The purpose of taking the incore measurements of the LTAs is to be able to compare the measured results with the CASMO-4/SIMULATE-3 MOX calculated results to demonstrate the impact of using weapons grade material versus reactor grade material and to demonstrate that the reactor grade MOX database for calculating core reload design is appropriate for use with weapons grade MOX.

(ii) Duke believes that the methodology approval for MOX fuel lead assemblies should not be constrained to one unit at Catawba only. As a practical matter, Duke intends to use MOX fuel lead assemblies at one Catawba unit only. However, this is not a nuclear analysis methodology issue. Furthermore, it is conceivable (though not likely) that MOX fuel lead assembly circumstances could change. The Duke report has justified application of the CASMO-4/SIMULATE-3MOX methodolgoy to MOX fuel at either McGuire or Catawba, assuming that the other necessary regulatory approvals are in place to support the loading of MOX fuel lead assemblies. Duke believes that the SE for the analytical methodology is an undesirable place for restrictions on the use of MOX fuel for reasons that have nothing to do with the methodology. At a minimum, if the "Catawba-only" restriction is retained, the SE should make it clear that the restriction on location of MOX fuel lead assembly use has no basis related to the analytical methodology, but is due to other considerations. In Attachment 2, Duke has included as markups that would make the MOX LTA approval applicable to all four units.

<u>NRC Staff Response</u>

As noted above, Duke has removed McGuire from the MOX LTA program. Therefore, an explicit approval of the methodology for LTAs at McGuire would constitute approval of a methodology for a usage that the licensee indicates will never be exercised. As a matter of policy, the NRC staff elects not to issue such approvals. However, the NRC staff has not identified any technical issues that would preclude approval and use of this methodology for McGuire, had MOX LTAs been chosen for McGuire.

<u>Duke Comment No. 2</u>

The "cc" list should include McGuire Nuclear Station as well.

<u>NRC Staff Response</u>

This report will also be distributed to the McGuire Mailing list.

<u>Duke Comment No. 3</u>

With respect to Sections 3.0, 3.3, and 4.0, of the Draft SE, the SIMULATE-3K MOX computer code is an integral part of the methodology for DRWM. In order to ensure clarity, the SIMULATE-3K MOX code should be specifically mentioned. Duke has included clarifying markups in Attachment 2.

NRC Staff Response

Duke proposes the addition of the following paragraph at the end of Section 3.0:

> SIMULATE-3K MOX is an extension of SIMULATE-3K, which is used for analysis of core transients. The spatial neutronics models in SIMULATE-3K MOX are identical to those in SIMULATE-3 MOX. SIMULATE-3K MOX solves the transient neutron diffusion equation incorporating effects of delayed neutrons, spontaneous fission in fuel, alpha-neutron interactions from actinide decay, and gamma-neutron interactions from long term fission product decay. For the applications reviewed in Topical Report DPC-NE-1005P, SIMULATE-3K MOX is used only as part of the dynamic rod worth measurement (DRWM) methodology.

The first three sentences of the above paragraph appear in the Topical Report, Section 2.4, "SIMULATE 3K MOX," as a description of the SIMULATE-3K MOX code's capabilities. The NRC staff issued a request for additional information (RAI) on Section 2.4 and Duke responded on September 12, 2002. The information in the last sentence of the proposed paragraph above is included in that RAI (No. 5) response. The NRC staff finds this description of the SIMULATE 3K code's capabilities to be acceptable for inclusion in the SE.

Duke's proposed changes to SE Sections 3.3 and 4.0 on this matter consist of adding "SIMULATE-3K MOX," to the code package name. The NRC staff finds this to be consistent with the Topical Report and the NRC staff's review and, therefore, acceptable.

Duke Comment on Section 3.3, Paragraph 1

As currently written, the beginning of Section 3.3 could give the impression that meeting the criteria for comparison to Westinghouse results (e.g., 2%/25 pcm) is an absolute requirement for applying DRWM with non-Westinghouse codes. As noted in Duke's December 2, 2003 letter on DRWM (Canady to U.S. NRC), the absolute need to meet those criteria was modified by the Safety Evaluation Report on WCAP-13360. This point should be clarified in the current SE to avoid creating an impression that Duke has failed to meet the appropriate DRWM requirements. By addressing those limited instances in which the acceptance criteria were not met, Duke has satisfied the pertinent requirements. Duke has included a markup addressing this point in Attachment 2.

NRC Staff Response

Duke proposes to add the following to the first paragraph of the SE, Section 3.3:

> A subsequent Safety Evaluation of Westinghouse WCAP-13360 accepted the clarification that deviations from the above acceptance criteria (comparison to Westinghouse generated results) may be acceptable if appropriately justified.

Duke's proposed clarification is accurate and the NRC staff finds it acceptable for inclusion into the SE.

Duke Comment on Section 3.3, Paragraph 3

Duke requests that NRC provide proper context for the discussion of "Duke under-predictions" in the second-to-last paragraph of Section 3.3. The "under-predictions" are relative to another analytical method (Westinghouse calculations), not data. Duke has included a clarifying markup in Attachment 2.

NRC Staff Response

The licensee's submittals include (a) comparison of *predictions* of control rod worths made by Westinghouse analytical methods to predictions made by Duke methods and, (b) comparison of *measurements* of control rod worths determined by Westinghouse to those determined by Duke. Duke proposes to add the words "Relative to Westinghouse," to the second sentence of the third paragraph to clarify that the discussion refers to a comparison of two analytical methods and not to a comparison of measured data. Duke's proposal is consistent with its discussion of the issue in its letter dated December 2, 2003, and is consistent with the NRC staff's understanding of the issue, and is, therefore, acceptable.

Duke Comment on Section 3.3

Duke considers it essential that the SE clarify that the DRWM methodology is approved for application to cores including, at a minimum, four MOX fuel lead assemblies. Duke has included a clarifying markup in Attachment 2.

NRC Staff Response

Duke proposes the inclusion of the following paragraph in Section 3.3:

> The NRC staff finds the use of the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX methodology acceptable for use with the DRWM methodology for McGuire and Catawba with LEU fuel and up to four MOX LTAs.

The NRC staff approves the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K methodology for DRWM methodology for McGuire and Catawba with LEU fuel and for Catawba with up to four MOX LTAs and has added a clarifying statement to Section 3.3 to this effect.

As noted above, Duke has removed McGuire from the MOX LTA program. Therefore, an explicit approval of the methodology for LTAs at McGuire would constitute approval of a methodology for a usage that the licensee indicates will never be exercized. As a matter of policy, the NRC staff elects not to issue such approvals. However, the NRC staff has not identified any technical issues that would preclude approval and use of this methodology for McGuire, had MOX LTAs been chosen for McGuire.

Duke Comment on Section 4.0

In order to ensure clarity, the conclusion section should specifically address approval to use CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX computer codes for DRWM. Duke has included clarifying markups in Attachment 2.

NRC Staff Response

Duke proposed to add the following to Section 4.0:

> In addition, the NRC staff concluded that the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX methodology can be applied to Catawba and McGuire DRWM.

This is essentially the same issue for SE Section 4.0 as discussed above for SE Section 3.3. For the same reasons as discussed above, the NRC staff has added the same clarifying statement made in SE Section 3.3 to SE Section 4.0.

5.0 CONCLUSION

Duke submitted the Topical Report (Reference 1) and supplementary information in References 2, 3, 4, 5 and 6 for review by the NRC staff. The licensee performed extensive benchmarking using the CASMO-4/SIMULATE-3 MOX methodology. The licensee's effort consisted of conducting detailed comparisons of calculated key physics parameters with measurements obtained from several operating cycles of Catawba and McGuire, the St. Laurent reactor in France, and several MOX critical experiments. These results were then used to determine the set of 95/95 (probability/confidence) tolerance limits for application to the calculation of the stated physics parameters.

Based on the review of the analyses and results presented in References 1, 2, 3 and 4, the NRC staff has concluded that the CASMO-4/SIMULATE-3 MOX methodology, as validated by Duke, can be applied to the Catawba and McGuire steady-state physics calculations for reload applications as described in the above technical evaluation. The NRC staff finds the use of the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX methodology acceptable for use with the DRWM methodology for McGuire with LEU fuel and for Catawba with LEU fuel and up to four MOX LTAs. The NRC staff's approval is limited to the range of fuel configurations and core design parameters as stated and referenced by the August 3, 2001, submittal. Introduction of significantly different fuel designs will require further validation of the above-stated physics methods for application to Catawba and McGuire by the licensee and will require review by the NRC staff. Additionally, the results of the LTA in-core performance and predictive capabilities of CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX for weapons grade MOX will need to be demonstrated and submitted to the NRC for review and approval as part of any application for partial MOX cores.

This approval is subject to the conditions listed above in Section 1.0 that have been provided by Duke in Reference 4.

6.0 REFERENCES

1. Letter from K. S. Canady, Duke Power to the U.S. NRC, "Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," August 3, 2001 (Proprietary). A non-proprietary version was submitted by letter dated August 6, 2001.

2. Letter from K. S. Canady, Duke Power to the U.S. NRC, "Response to Request for Additional Information - Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," September 12, 2002.

3. Letter from K. S. Canady, Duke Power to the U.S. NRC, "Response to Request for Additional Information - Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," November 12, 2002.

4. Letter from M. S. Tuckman, Duke Power to the U.S. NRC, "Physics Testing Program in Support of Topical Report DPC-NE-1005P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," June 26, 2003.

5. Letter from K. S. Canady, Duke Power to the U.S. NRC, "Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," August 14, 2003

6. Letter from K. S. Canady, Duke Power to the U.S. NRC, "Additional Information Related to Duke Topical Report DPC-NE-1005P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," December 2, 2003.

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

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

9. M. G. Natrella, "Experimental Statistics," National Bureau of Standards Handbook 91, October 1966.

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.126, Revision 1, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification," March 1978.

11. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandia Corporation, March 1963.

12. ANSI-N15.15-1974, "Assessment of the Assumption of Normality (Employing Individual Observed Values)," October 1973.

Principal Contributors: A. Attard
U. Shoop

Date: August 20, 2004

Page Intentionally Left Blank

DPC-NE-1005-PA
Revision 0

List of Attached Correspondence

1. NRC letter to Duke requesting additional information, July 29, 2002.
2. Duke letter to NRC providing additional information, September 12, 2002.
3. Duke letter to NRC providing additional information, November 12, 2002.
4. Duke letter to NRC documenting startup testing program to be used for cores containing MOX fuel assemblies, June 26, 2003.
5. Duke letter to NRC clarifying results from dynamic rod worth measurement benchmark, December 2, 2003.

Page Intentionally Left Blank

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 29, 2002

Mr. K. S. Canady
Duke Energy Corporation
526 South Church St
Charlottte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: 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)

Dear Mr. Canady:

The Nuclear Regulatory Commission is reviewing your submittal dated August 3, 2001, entitled "Topical Report DPC-NE-1005P, Revision 0, Nuclear Design Methodology Using CASMO-4/SIMLATE-3 MOX (Proprietary)" and has identified a need for additional information as identified in the Enclosure. These issues were discussed with your staff on June 25, 2002. Please provide a response to this request within 45 days of receipt of this letter so that we may complete our review.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

TOPICAL REPORT DPC-NE-1005P, REVISION 0

NUCLEAR DESIGN METHODOLOGY USING CASMO-4/SIMULATE-3 MOX

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

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.

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?

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.

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.

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?

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?

Enclosure

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.

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?

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?

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?

11. In Table 3-5, it appears that there are large differences between the measured and predicted hot zero power isothermal temperature coefficients. Please explain.

12. In Table 3-2, it appears that CASMO-4/SIMULATE-3 is over-predicting the boron concentrations and thus is non-conservative. This is also the case in Table 3-8. Please explain.

13. The last sentence of the last paragraph of Section 6.2, "Impact of MOX Fuel on DRWM," suggests that there is little difference between an LEU fuel core and an LEU/MOX fuel core. However, no data was provided to support this claim. Please provide quantitative technical justification to support this claim.

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.

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.

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.

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.

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.

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.

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 withhheld and that meets the other requirements of the Commission's regulations in 10 CFR 2.790, "Public inspections, exemptions, requests for withholding."

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.

22. Please provide all documentation and the code for CASMO4 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.

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.

McGuire Nuclear Station

cc:

Ms. Lisa F. Vaughn
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

County Manager of
Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Dr. John M. Barry
Mecklenburg County
Department of Environmental
Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinshouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health and Natural
Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Page Intentionally Left Blank

Duke Power®
A Duke Energy Company

Ken S. Canady
Vice President
Nuclear Engineering

Duke Energy Corporation
526 South Church St.
Charlotte, NC 28202
EC08H
P.O. Box 1006
(704) 382-4712 OFFICE
(704) 382-7852 FAX

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.

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

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 D. Smith

Kord Smith

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

Shawna Lee Smith

Notary Public

My Commission Expires: 3-15-08

SHAWNA LEE SMITH
NOTARY
PUBLIC
STATE OF IDAHO

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

Electric Power Research Center
College of Engineering

NC STATE UNIVERSITY

Campus Box 7917
Raleigh, NC 27695-7917

919.515.3517
919.515.5108 (fax)

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

Paul Turinsky

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

Thelma B. Burrell

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?

<u>Response</u>:

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?

<u>Response</u>:

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.

<u>Response</u>:

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 LEU	MOX	No MOX Bias 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.

When flux maps are taken, the measured signals are stored in the plant computer system.

The conversion factors are applied through a post-processing computer code, prior to performing a comparison between measured and predicted powers.

11. In Table 3-5, it appears that there are large differences between the measured and predicted hot zero power isothermal temperature coefficients. Please explain.

    Response:

    The acceptance criterion on the beginning of cycle (BOC) hot zero power (HZP) isothermal temperature coefficient (ITC) test is ± 2 pcm/°F. All of the deviations in Table 3-5 are within this criterion. The data provided in the answer to Question 7 shows that the fidelity of ITC predictions from CASMO-4 / SIMULATE-MOX is consistent with the currently approved methodology.

12. In Table 3-2, it appears that CASMO-4/SIMULATE-3 is over-predicting the boron concentrations and thus is non-conservative. This is also the case in Table 3-8. Please explain.

    Response:

    CASMO-4 and SIMULATE-3 MOX are best estimate predictive tools. Some boron predictions are higher than measured data, and other predictions are lower. The predictions are not considered to be inherently conservative or non-conservative. Standard industry practice is to bias the predicted boron concentrations to reflect anticipated differences between predictions and measurements. This practice further reduces the already small boron predictive error.

    The acceptance criterion on HFP boron deviation is 1% Δk/k. This acceptance criterion translates to a boron deviation on the order of 125 ppm. All of the deviations in Tables 3-2 and 3-8 are well within this criterion. The data provided in the response to Question 7 shows that the fidelity of soluble boron concentration predictions from CASMO-4 / SIMULATE-3 MOX is consistent with or better than the currently approved methodology.

13. The last sentence of the last paragraph of Section 6.2, "Impact of MOX Fuel on DRWM," suggests that there is little difference between an LEU fuel core and an LEU/MOX fuel core. However, no data was provided to support this claim. Please provide quantitative technical justification to support this claim.

    Response:

    The simulations referenced in Section 6.2 of the topical report are discussed in Sections 4 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			
FΔh	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.

<u>Response</u>:

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 $[0.8\% \text{ RMS}]_S$.

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%

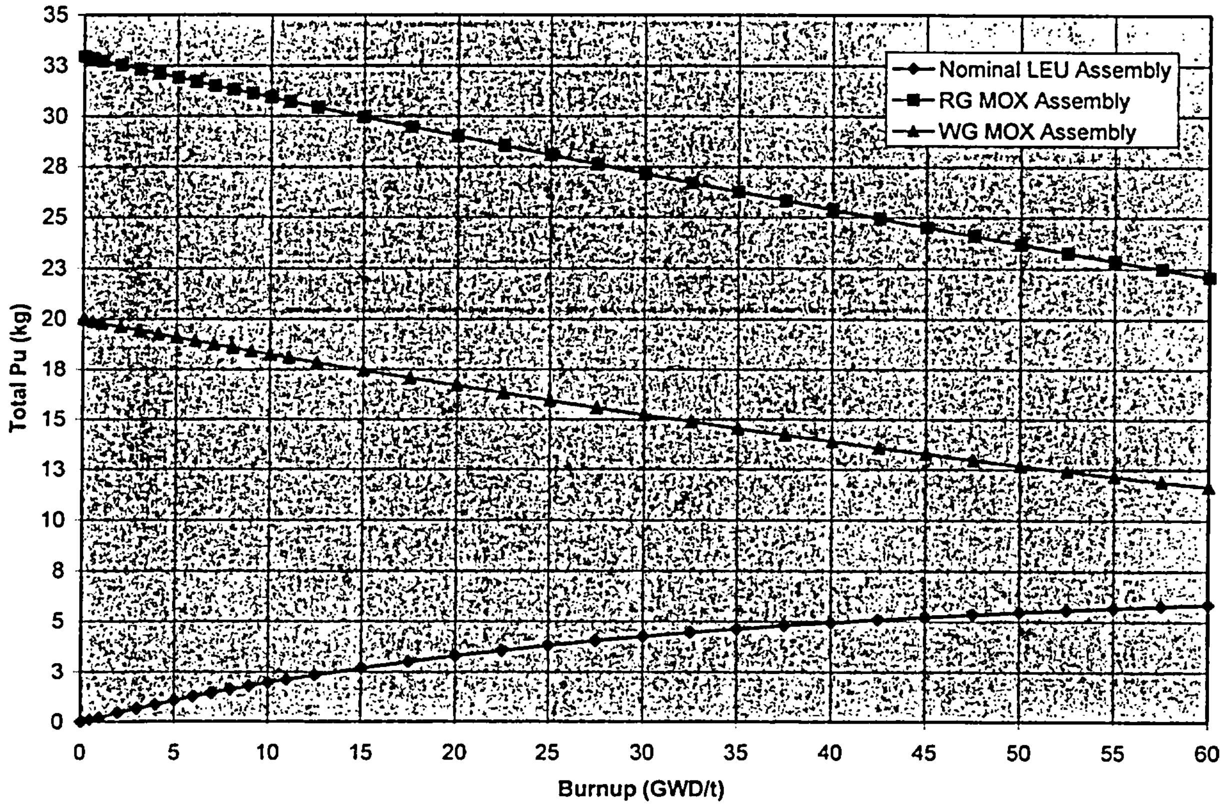


Figure 3.1 Total Plutonium Mass

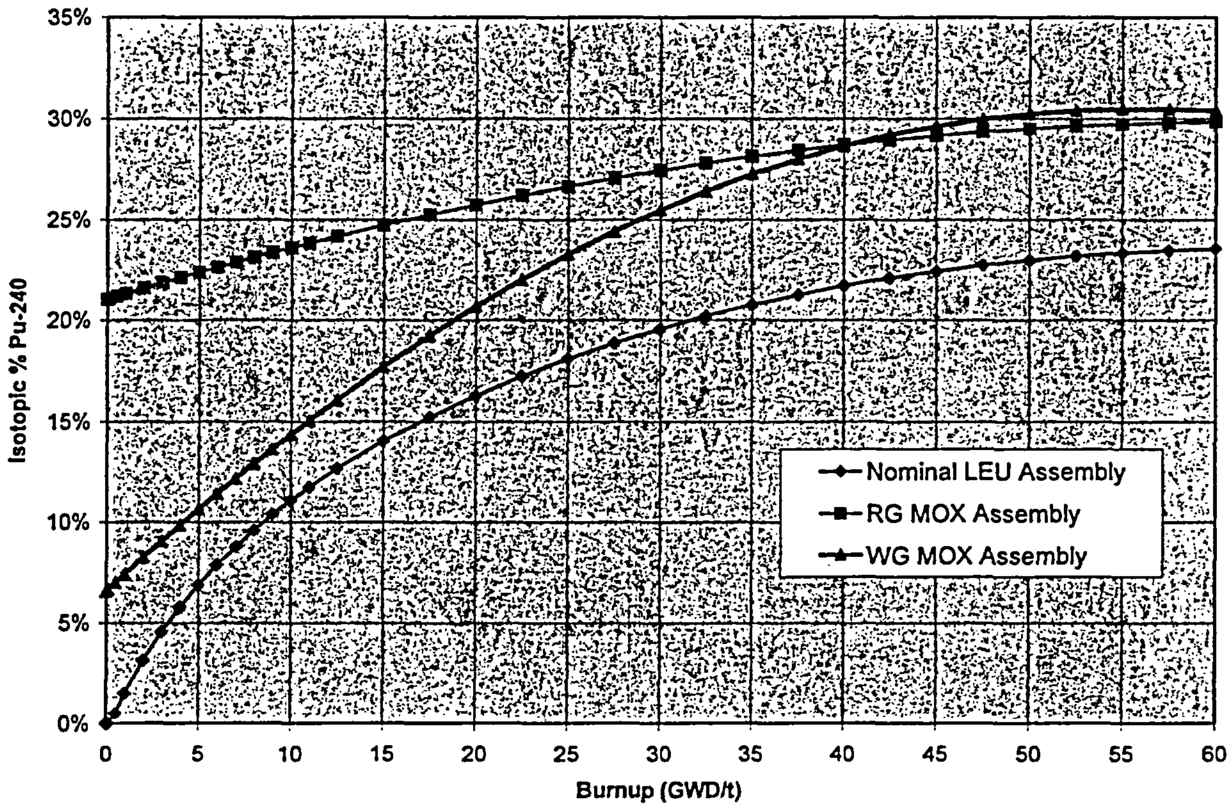


Figure 3.2 ^{240}Pu Concentration

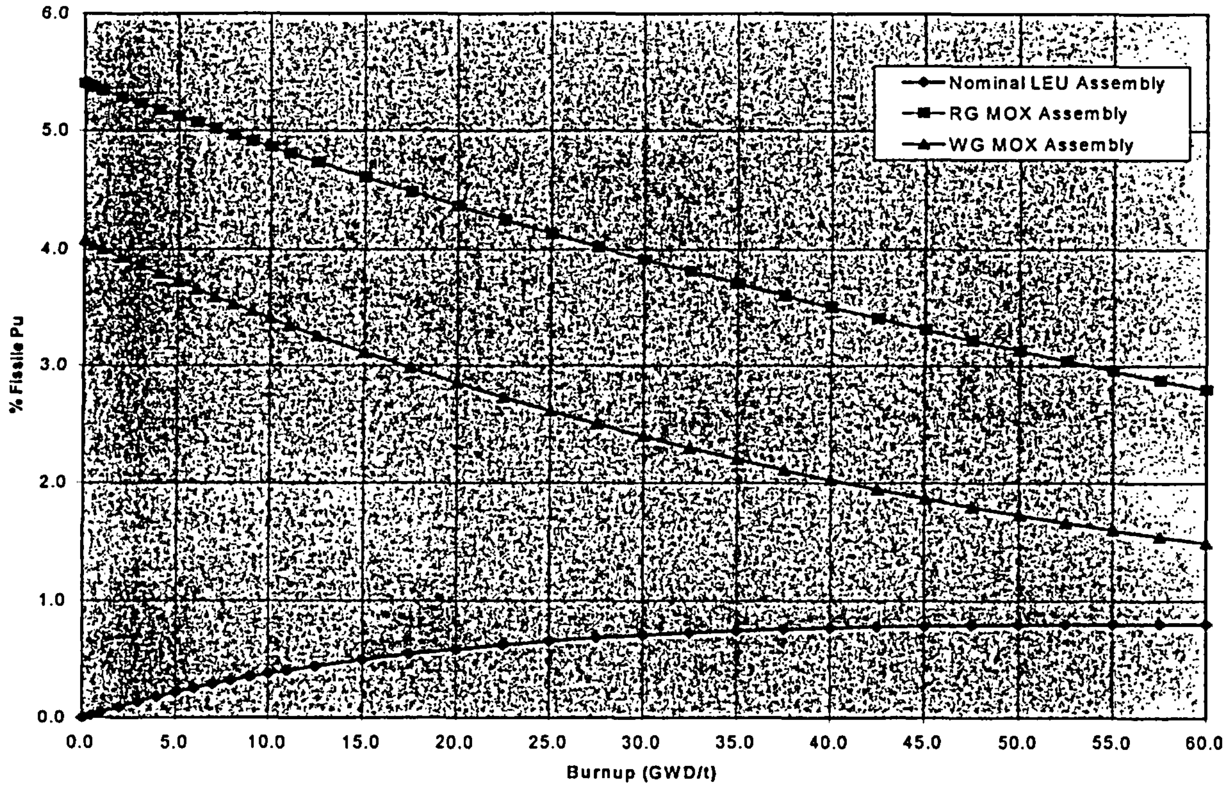


Figure 3.3 Fissile Plutonium

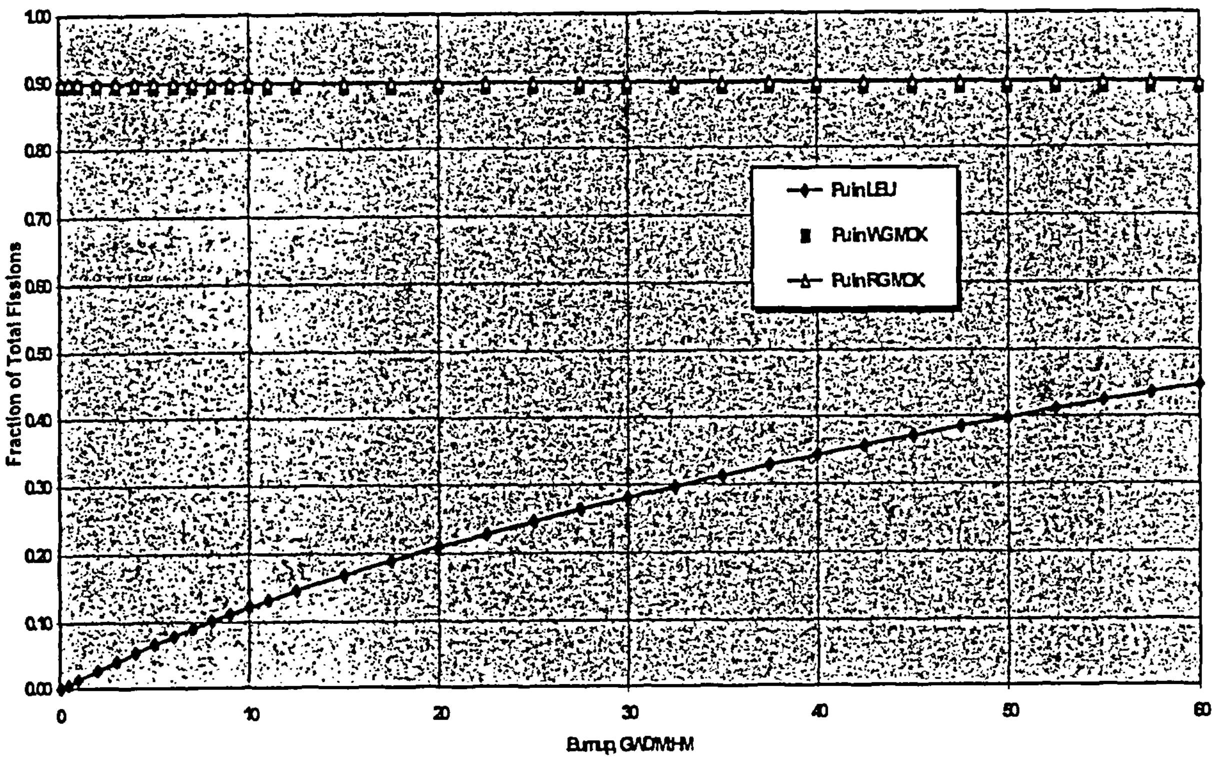


Figure 3.4 Plutonium Fissions – Fraction of Total Fissions

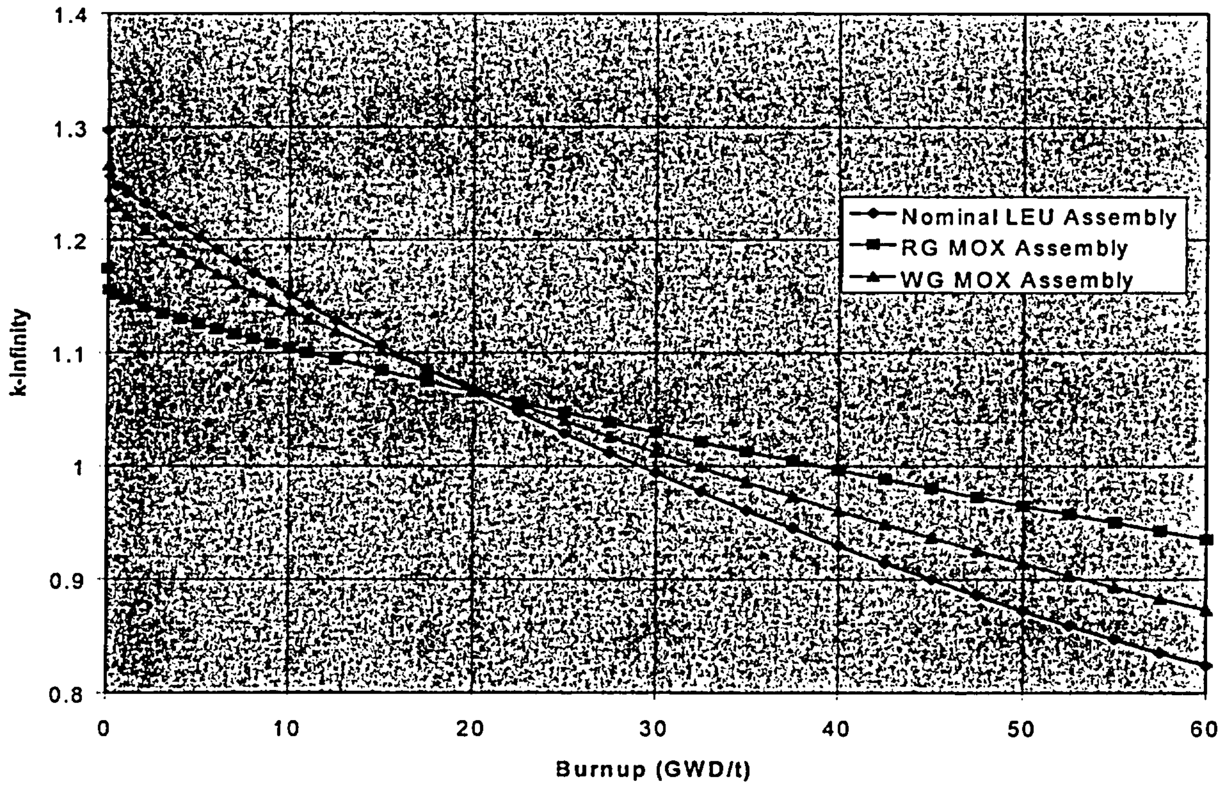


Figure 3.5 k_{∞} vs. Burnup

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.

Page Intentionally Left Blank

Duke Power®
A Duke Energy Company

Ken S. Canady
Vice President
Nuclear Engineering

Duke Energy Corporation
526 South Church St.
Charlotte, NC 28202
EC08H
P.O. Box 1006
(704) 382-4712 OFFICE
(704) 382-7852 FAX

November 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 Using CASMO-4/ SIMULATE-3 MOX* (Proprietary), (TAC Nos. MB2578, MB2579, MB2726 and MB2729)

Reference: 1) 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*
2) Duke Letter dated September 12, 2002, Response to Request for Additional Information Re: Topical Report DPC-NE-1005P, Revision 0, *Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX* (PROPRIETARY)

Duke's initial response to Reference 1 was provided in Reference 2. However, this response was incomplete because some of the information was proprietary to Electricite de France (EDF) and Duke did not have an affidavit from EDF supporting the withholding of the proprietary information. The missing information consisted of References 19 and 20 from Duke Topical Report DPC-NE-1005P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX* (requested in Question 20 of Reference 1) and the response to Question 8 from Reference 1.

Enclosed are two copies each of References 19 and 20 from Duke Topical Report DPC-NE-1005P. In addition, supplementary information to that contained in Reference 20 is enclosed in order to provide the NRC staff with operating data from Saint Laurent B1 operating cycles (campaigns) eight, nine, and ten, which were benchmarked in the topical report. Attachment 1 is the proprietary response to Question 8 from Reference 1 and Attachment 2 is the redacted version of the response with all of the EDF proprietary information deleted.

This submittal contains information that is proprietary to EDF. In accordance with 10 CFR 2.790, Duke requests that this information be withheld from public disclosure. An affidavit from EDF is included that attests to the proprietary nature of the information.

PROPRIETARY
Material Attached

In a related matter, Duke has requested that the NRC staff provide written confirmation of the planned schedule for the review of the subject topical report. To date, Duke has not received this information. This schedule is still needed in order for Duke to plan for the use of the methodology in the topical report for future fuel cycle designs.

In addition, Question 22 of Reference 1 pertained to the provision of documentation and software for the CASMO-4 and SIMULATE-3 MOX computer codes. As noted in the initial Duke response (Reference 2), Duke would like to work with the NRC staff to identify an arrangement that will enable the NRC staff to perform its review, while at the same time addressing Duke and Studsvik Scandpower, Inc., concerns associated with the NRC staff request. Duke is still awaiting the NRC staff's response to Duke's proposal on this issue.

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

Very truly yours,

K. S. Canady

Attachments

Enclosures
1) Operating Data for MOX Fuel Power Plant (Reference 19) – 2 copies
2) Operation Monitoring and Power Diagrams Saint Laurent B1:
Campaigns 3-4-5-6-7 (Reference 20) – 2 copies
3) Reference 20 Supplementary Information – 2 copies

EDF
Electricité de France

AFFIDAVIT OF CATHERINE GAUJACQ

1 My name is Ms Catherine Gaujacq. I am President of Electricité de France International North America de France.

2 Mr. Michel Ponticq who has the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing has asked me to sign in his lieu et place.

3 Thus I am authorized on the part of EDF to apply for this withholding.

4. 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.

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

6. 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 EDF and has been held in confidence by EDF and its consultants.

(ii) The information is of a type that would customarily be held in confidence by EDF. The information consists of operating data for reactor cores with a mixture of mixed oxide and low enriched uranium fuel that were developed at significant cost to EDF and which provide a competitive advantage to EDF.

(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 not available in public to the best of our knowledge and belief

(v) The proprietary information sought to be withheld is that which is marked in the proprietary version of Duke Energy's 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.* The proprietary information sought to be withheld from public disclosure has substantial commercial value to EDF because the information

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

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

1/2 *[handwritten initials]*

EDF
Electricité de France

(Continued)

Catherine Gaujacq

(c) Consists of reactor operating data, which is not readily available to others and therefore has value to EDF,

7. Public disclosure of this information is likely to cause harm to EDF because it would allow other competitors in the nuclear industry to benefit from the results of an extensive reactor monitoring and measurement program without requiring commensurate expense or allowing EDF to recoup a portion of its expenditures or benefit from the sale of the information.

Catherine Gaujacq, being duly sworn, states that she is the person who subscribed her name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of her knowledge.

C. Gaujacq
Catherine Gaujacq

Subscribed and sworn to before me on this 4th day of November, 2002
Witness my hand and official seal.

Ann M. Bistodeau
Notary Public

My Commission Expires: ANN M. BISTODEAU
NOTARY PUBLIC, DISTRICT OF COLUMBIA
My Commission Expires 12-14-2005

SEAL

2/2 C.G

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 statement in Section 3.2.5 is made based on the similarity of those detector characteristics which are important to the nuclear performance of the device. The St. Laurent detectors are slightly smaller than the McGuire/Catawba detectors since the instrument thimbles in St Laurent are smaller. Table 2 below summarizes the important parameters of the incore instrument in each reactor.

Table 2
Incore Instruments Comparison

	McGuire/Catawba	**St. Laurent**
Instrument Thimble Tubes		
Material	Stainless Steel	[ ]E
Outer Diameter	.762 cm	[ ]E
Detectors		
Shell Material	Stainless Steel	[ ]E
External Diameter	.478 cm	[ ]E
Fill Gas	Argon	[ ]E
Electrode Material	Stainless Steel	[ ]
Length	5.33 cm	[ ]E (includes welded end plug)
Active Length	2.54 cm	[ ]E
Coating Material	U_3O_8	[ ]E
U-235 Content	> 90 w/o	[ ]E
Minimum Neutron Sensitivity	1.0×10^{-17} amps / nv	[ ]E
Max Gamma Sensitivity	3.0×10^{-14} amps / R / hr	[ ]E

Page Intentionally Left Blank

Duke Power
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power
526 South Church St. EC07H
Charlotte, NC 28202
P. O. Box 1006 EC07H
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
(704) 382-4360 FAX

June 26, 2003

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
Physics Testing Program in Support of Topical Report DPC-NE-1005P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX*

The Nuclear Regulatory Commission (NRC) staff as part of their review of the subject topical report requested a description of the physics testing programs that Duke Power intends to perform for mixed oxide (MOX) fuel lead assembly and batch use that will provide verification of the nuclear analysis methodology described in the topical report. The proposed test program and the reports generated from this program are described in the attachment to this letter. This testing program and the associated reporting were developed based on discussion and communication with the NRC staff.

The proposed test program, while similar in scope to the current Duke physics testing program, contains some restrictions on power levels for neutron flux measurement during power escalation. The proposed test program also includes additional requirements for documentation. As described in the attachment, the proposed test program would be carried out for numerous cycles over many years at the four McGuire and Catawba units, and it would generate a large amount of nuclear data related to the performance of cores containing a mixture of MOX fuel and low enriched uranium fuel. As the test program progresses, Duke may propose modifications to the duration of the program, if such changes are warranted based on the results of the testing and associated analyses.

Four regulatory commitments that relate to the content, performance, timing, and documentation of the described physics testing program are listed at the end of the attachment. Inquiries on this matter should be directed to G.A Copp at (704) 373-5620.

Very truly yours,

M. S. Tuckman

M.S. Tuckman

Attachment

cc with attachment:

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

S. M. Shaeffer
U.S. Nuclear Regulatory Commission
Senior Resident Inspector
McGuire Nuclear Station

E.F. Guthrie
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

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

Duke Power
Mixed Oxide Fuel Project
Core Physics Testing and Validation Program

Introduction

As part of the initiative to dispose of surplus weapons plutonium in the United States and Russia, Duke Power is planning to use mixed oxide (MOX) fuel derived from surplus weapons grade plutonium at the McGuire and Catawba nuclear power reactors. Duke plans to conduct a lead assembly program in which four MOX fuel lead assemblies will be used for two operating cycles at one of the nuclear units, and one or more of the assemblies will be irradiated for a third cycle. As part of the MOX fuel lead assembly program, core conditions will be measured and monitored during plant startup and operation, and post-irradiation examinations of the fuel will be performed. Following a successful fuel qualification program that includes two cycles of lead assembly irradiation, Duke plans to begin batch-scale use of MOX fuel, contingent on (i) regulatory approval, and (ii) availability of the fuel. In batch implementation, Duke will load a mixture of fresh MOX fuel and fresh low enriched uranium (LEU) fuel, similar to the ongoing practice in European reactors using MOX fuel derived from reactor grade plutonium. The McGuire and Catawba MOX fuel core fractions would increase to approximately 40% over several cycles.

The use of batch quantities of MOX fuel will represent a major change in the core designs of Catawba and McGuire. Accordingly, the physics testing program has been reviewed with respect to (i) protecting public health and safety, and (ii) obtaining measured data to validate computer code predictions. The physics testing programs planned for the MOX lead assembly cores and then the partial batch MOX fuel cores are discussed below.

Physics Testing for MOX Lead Assembly Cores

One of the primary goals of the MOX fuel lead assembly program will be to collect measured neutronic data to validate the computer code predictions. The effect of four MOX fuel assemblies on global core reactivity parameters will be minimal, as demonstrated by analyses that are summarized in Reference 1, Attachment 3, Section 3.7.2.3. Therefore, the valuable neutronic measured data from the MOX fuel lead assembly program will be core power distributions derived from incore neutron flux measurements (flux maps). For a lead assembly program containing four MOX fuel assemblies, Duke will place at least two of the MOX fuel lead assemblies in core locations that are measured directly by the movable incore detector system for the first and second cycles of lead assembly irradiation. For the third cycle of irradiation, core design constraints will dictate whether the placement of the MOX fuel lead assembly is in a measured core location, or not. Not placing the MOX lead assembly in a measured location during the third cycle of irradiation is acceptable because the purpose of this irradiation is to achieve a high burnup on the MOX lead assembly to assess mechanical performance of the fuel assembly.

The physics test program to be used at Catawba and McGuire for cores containing MOX fuel lead assemblies is based on the American Nuclear Society (ANS) Standard for Reload Startup Physics Tests for Pressurized Water Reactor (Reference 2) and the Nuclear Regulatory Commission (NRC)-approved Duke Power reload startup physics test program for McGuire and Catawba (Reference 3), modified to include the dynamic rod worth measurement (DRWM) technique as described in Reference 4. A summary of the planned startup physics testing program is shown in Table 1.

The physics testing program will provide data to assess various physics parameters important to confirming the core design predictions. The physics data will be measured at hot zero power (HZP) and at various power levels during initial cycle power escalation. The HZP measurements will include the all-rods-out critical boron concentration, the isothermal temperature coefficient, and individual bank worths for each of the nine control and safety banks. During the power escalation phase of startup, power distribution measurements will be made at a minimum of two intermediate power levels as well as full power. The final physics test is the measurement of the full power critical boron concentration.

The bank worth measurements will be performed using the Westinghouse DRWM technique. DRWM provides integral and differential bank worth data for all banks. The applicability of the DRWM technique to cores containing MOX fuel assemblies was evaluated in the Duke CASMO-4/SIMULATE-3 MOX nuclear analysis methodology topical report (Reference 5, Section 6).

As noted earlier, the four MOX fuel lead assemblies will have a very small effect on the global reactivity parameters such as boron concentration and temperature coefficients. For these MOX fuel lead assembly cores, the principal measurement of interest will be flux maps, which will provide for a comparison of predicted to measured power distributions.

The power level plateaus for power distribution measurements are chosen based on a number of considerations. The best quality flux maps are taken at full power, steady-state conditions. Obtaining good flux maps at very low power is challenging because core conditions are not as stable and the detector signal strength is low, particularly in peripheral, low power assemblies. Since each full flux map takes several hours to perform at a stable power level, standard practice has been to take the lower power map while the reactor is being held at constant power for other plant system evolutions; *e.g.*, turbine heatup, turbine overspeed tests, and chemistry hold points.

The primary purpose of the first flux map is to provide additional confirmation that the core has been loaded as designed. The types of misloadings that are considered include both assembly misplacement and assembly manufacturing errors. MOX fuel assemblies provide a coincidental enhancement to this check because the MOX fuel instrument tube reaction rates are uniquely lower than comparable LEU fuel assemblies. This effect results from thermal neutron flux depression in MOX fuel due to the higher thermal absorption cross section of plutonium, relative to uranium.

The first power distribution measurement will be made at a power level that is sufficiently low such that it is not credible to exceed a power peaking related safety limit. Successfully meeting the acceptance criteria will provide assurance that the core is loaded properly, that it is operating as designed, and that it is acceptable from a safety perspective to proceed to the next power plateau for further testing.

The second power distribution measurement must be made between 50% and 80% full power. This power plateau is chosen to allow for a quality measurement using the movable incore detector system, while not challenging thermal margin limits. The data from this flux map is analyzed with the following objectives:

1. Confirm that the measured core peaking is within safety analysis limits.

2. Confirm that the predicted power distribution is within the acceptance criteria established for the test.

3. Confirm the trend of changes in the power distribution as a function of power level. This provides assurance that next power plateau will be acceptable.

The third power distribution measurement will be made above 90% full power (generally, this measurement is performed at full power). The acceptance criteria are the same as the second measurement, and this flux map provides further assurance and the core is operating as designed, in accordance with assumptions made in steady-state and transient safety analyses.

Once startup testing is successfully completed, flux maps are taken monthly during cycle operation, typically at steady-state, full power conditions. These core power distribution measurements will provide the primary data base against which the core power distribution predictions of the CASMO-4/SIMULATE-3 MOX codes will be assessed. The assessment of the full power data will be performed as part of the normal core follow program which evaluates important parameters such as core reactivity, $F_{\Delta h}$ and F_q peaking factors, radial power distribution, and core average axial power shape on a monthly basis. For MOX fuel lead assembly cores, the program will be expanded to include the analysis of axial power shapes for the MOX fuel assemblies in instrumented locations.

While the flux map data collected on MOX fuel assemblies during power escalation will provide useful data, the best and most appropriate data for confirming computer code predictions will be obtained from the monthly full power flux maps during cycle operation. This is because the core conditions for the full power flux maps are more stable with respect to spatial transients of fission product poison distributions (e.g., xenon and samarium), core flow, and core temperature distributions. In addition, the McGuire and Catawba reactors operate almost entirely at full power conditions, and it is at full power that steady-state thermal margins are smallest. Therefore, full power operation is the primary condition of concern with respect to the uncertainty associated with computer code predictions.

In summary, at least three core power distribution measurements will be taken during startup of the core with MOX fuel lead assemblies - two measurements at intermediate power conditions and one above 90% power. These measurements, coupled with monthly flux maps during cycle operation, will provide a substantial data base against which core physics calculations of weapons grade MOX fuel assembly performance can be validated.

Physics Testing for Partial Batch MOX Cores

In Reference 5 the CASMO-4/SIMULATE-3 MOX codes were validated against startup testing and operating data from the St. Laurent B1 reactor. Those St. Laurent cores contained a mixture of LEU and reactor grade MOX fuel assemblies. Prior to loading batch quantities of MOX fuel, the MOX fuel lead assembly program will provide additional data. It is expected that these data will further confirm the ability of the Duke nuclear design methods to predict the performance of MOX fuel in mixed cores. Accordingly, the same startup testing program will be followed for partial MOX fuel cores as for the MOX fuel lead assembly cores. This program will be performed on each McGuire and Catawba unit starting with the first operating cycle containing batch quantities of MOX fuel and continuing through the first equilibrium cycle. For the purposes of this test program description, an equilibrium cycle is defined as an operating cycle with a core containing 76 MOX fuel assemblies (feed and reload), or 39.4% of the core.

MOX Core Startup and Operating Reports

Duke will prepare startup reports for all cycles operating with MOX fuel lead assemblies and for all cycles for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above is reached. Each startup report will contain comparisons of predicted to measured data from the zero power physics tests and the three power distribution maps taken during power escalation. The reports will include discussions of any parameter that did not meet acceptance criteria. Duke will provide each report to the NRC within 60 days of measurement of the final power distribution map.

Duke will also prepare operating reports for all cycles operating with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above is reached. Each operating report will contain comparisons of predicted to measured monthly power distribution maps and monthly boron concentration letdown values. As noted earlier, these data provide the most benefit with respect to benchmarking the computer code predictions. Duke will provide each cycle operating report to the NRC within 60 days of the end of the fuel cycle.

Summary of Startup Test Commitments for McGuire/Catawba Cores Containing MOX Fuel

The following is a summary of NRC commitments made in this document related to physics testing for MOX fuel cores:

1. For a lead assembly program containing four MOX fuel assemblies, Duke will place at least two of the MOX fuel lead assemblies in core locations that are measured directly by the movable incore detector system for the first and second cycles of lead assembly irradiation.

2. Duke will perform the physics test program defined in Table 1 for all MOX fuel lead assembly cores and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above is reached. Core power levels at which low and

intermediate power escalation power distribution maps are taken will be consistent from cycle to cycle for each unit (within ± 3 % rated thermal power). Core power level at which power distribution maps are taken may vary among units and between McGuire and Catawba.

3. Duke will prepare a startup report for each operating cycle with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above is reached. Each startup report will contain comparisons of predicted to measured data from the zero power physics tests and the power distribution maps taken during power escalation. The reports will include discussions of any parameter that did not meet acceptance criteria. Duke will provide each report to the NRC within 60 days of measurement of the final power distribution map.

4. Duke will prepare an operating report for each operating cycle with MOX fuel lead assemblies and for each unit operating with partial MOX fuel cores until the equilibrium cycle defined above is reached. Each operating report will contain comparisons of predicted to measured monthly power distribution maps and monthly boron concentration letdown values. Duke will provide each cycle operating report to the NRC within 60 days of the end of the fuel cycle.

References

1. Letter, Tuckman, M. S. (Duke Power) to U. S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50, February 27, 2003.

2. ANSI/ANS-19.6.1, Reload Startup Physics Tests for Pressurized Water Reactors, American National Standard, 1997.

3. Letter, Hood, D. S. (U. S. Nuclear Regulatory Commission) to Tucker, H. B. (Duke Power), Transmittal of Safety Evaluation Report for McGuire and Catawba Reload Startup Physics Test Program, May 18, 1988.

4. WCAP-13360-P-A, Revision 1, "Westinghouse Dynamic Rod Worth Measurement Technique," October 1998.

5. DPC-NE-1005P, Duke Power Nuclear Design Methodology Using CASMO-4/ SIMULATE-3 MOX, August 2001.

6. BAW-10231P, COPERNIC Fuel Rod Design Computer Code, Framatome Cogema Fuels, September 1999.

7. DPC-NE-2005P-A, Revision 3, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, September 2002.

Table 1
Physics Test Program for
McGuire and Catawba MOX Fuel Cores

Physics Test	Core Condition	Acceptance Criteria
Critical Boron Concentration - All Rods Out	Hot Zero Power	Predicted +/- 50 PPM
Isothermal Temperature Coefficient	Hot Zero Power	Predicted +/- 2 PCM/F
Bank Worth Measurements	Hot Zero Power	**Review Criteria** Individual Banks $\pm$ 15% or 100 PCM (whichever is greater) Sum of all banks $\pm$ 8% of Predicted **Acceptance Criteria:** Sum of all banks $\geq$ 90% of Predicted
Low Power Flux Map (0-40% FP) Full core map including all operable instrument locations	Between 0 and 40 percent Full Power	Normalized reaction rates or assembly power: $\pm$ 10% of Predicted, Root Mean Square error: $\leq$ 0.05
Intermediate Flux Map 1 (50-80% FP) Full core map including all operable instrument locations	Between 50 and 80 percent Full Power	Normalized reaction rates or assembly power: $\pm$ 10% of Predicted, Root Mean Square error: $\leq$ 0.05
High Power Flux Map (> 90% FP) Full core map including all operable instrument locations	Greater than 90 percent Full Power	Normalized reaction rates or assembly power: $\pm$- 10% of predicted, Root Mean Square error $\leq$ 0.05
Critical Boron Concentration - All Rods Out	Greater than 90 percent Full Power	Predicted +/- 50 PPM

Duke Power
A Duke Energy Company

Ken S. Canady
Vice President
Nuclear Engineering

Duke Energy Corporation
526 South Church St.
Charlotte, NC 28202
EC08H
P.O. Box 1006
(704) 382-4712 OFFICE
(704) 382-7852 FAX

December 2, 2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

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
Additional Information Related to Duke Topical Report DPC-NE-1005P, *Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX* (TAC Nos. MB2578, MB2579, MB2726 and MB2729)

The Nuclear Regulatory Commission (NRC) is currently reviewing the Duke Power (Duke) Topical Report DPC-NE-1005P (Reference 1), which among other things addresses the application of the CASMO-4, SIMULATE-3 MOX, and SIMULATE-3K MOX computer codes to dynamic rod worth measurement (DRWM) at the McGuire and Catawba Nuclear Stations. In a telephone conversation on November 10, 2003, NRC personnel noted that, in a few instances, Duke DRWM calculations differed from Westinghouse DRWM predictions by an amount greater than the acceptance criteria for such comparisons. NRC requested that Duke provide an explanation for the deviations. The explanation is provided in the attached write-up.

If you have any questions regarding this matter, please contact G. A. Copp at (704) 373-5620.

Sincerely,

K. S. Canady

Attachment

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

J. B. Brady
NRC Sr. Resident Inspector
McGuire Nuclear Station

E. F. Guthrie
NRC Sr. Resident Inspector
Catawba Nuclear Station

R. E. Martin, Senior Project Manager (Addressee Only)
U. S. Nuclear Regulatory Commission (Mail Stop O-8 G9)
11555 Rockville Pike
Rockville, MD 20852-2738

Dynamic Rod Worth Measurement
Acceptance Criteria

The Nuclear Regulatory Commission (NRC) is currently reviewing the Duke Power (Duke) Topical Report DPC-NE-1005P (Reference 1), which among other things addresses the application of the CASMO-4, SIMULATE-3 MOX, and SIMULATE-3K MOX computer codes to dynamic rod worth measurement (DRWM) at the McGuire and Catawba Nuclear Stations. In Reference 2 Westinghouse proposed five criteria "... for a utility to use to demonstrate competencies to perform Dynamic Rod Worth Measurement (DRWM) design calculations." Criterion 4 addresses comparison calculations and states:

> Prior to the first application by a utility using their own methods to perform physics calculations in support of DRWM for LPPT, the utility will demonstrate its ability to use the methods supplied by Westinghouse by comparing its calculated results with the analyses and results obtained by Westinghouse during the first, or subsequent, application(s) of DRWM at the utility's plant.

In Reference 3 Westinghouse clarified Criterion 4. Among other things, Westinghouse noted that utility calculations exceeding the acceptable variation does not imply that the utility DRWM calculations are unacceptable. In such an instance the utility should report to the NRC the reasons for exceeding the acceptable deviations, and that "the review need not be complicated."

In Reference 4 the NRC transmitted to Westinghouse a Safety Evaluation Report (SER) for WCAP-13360 "Westinghouse Dynamic Rod Worth Measurement Technique" and related submittals with initial DRWM results. In the SER, the NRC accepted the provisions for technology transfer as described in References 2 and 3.

Original Duke DRWM Results

The Duke Power topical report DPC-NE-2012A (Reference 5) included benchmarks against Westinghouse DRWM calculations for six startups. The Duke calculations were performed using the CASMO-3, SIMULATE-3, and SIMULATE-3K computer codes. The report addressed the five acceptance criteria from Reference 2. Concerning Criterion 4, the difference between the Duke and Westinghouse results exceeded the individual rod bank worth criterion in 6 out of 108 instances. In the report, Duke explained that the bank worth differences are consistent with differences in predicted radial power distribution at hot zero power (HZP) between the Duke and Westinghouse methodologies. The NRC approved the Duke application of DRWM to McGuire and Catawba by the Reference 6 SER.

DPC-NE-1005P DRWM Results

Section 6 of Reference 1 includes new DRWM results for the same McGuire and Catawba cycles that were originally benchmarked in Reference 5. As noted above, the more recent Duke work used the CASMO-4, SIMULATE-3 MOX, and SIMULATE-3K MOX codes. As before, the Duke calculations showed good agreement with the Westinghouse results, especially when considering the independence of the underlying physics methods. In most but not all cases, Acceptance Criterion 4 was satisfied. Table 6.1 of Reference 1 shows the comparison between SIMULATE-3 MOX and Westinghouse predicted bank worths. The ± 2% or ± 25 pcm criterion were met in 116 of the 120 comparisons (54 predicted bank worths, 54 measured bank worths, six predicted total bank worths, and six measured total bank worths). The four comparisons that do not meet the criterion are:

- two predicted bank worths from McGuire 2, Cycle 13 (M2C13) (Banks CD and SA),
- one predicted bank worth from Catawba 1, Cycle 12 (C1C12) (Bank CD), and
- one total predicted bank worth (C1C11).

Individual Rod Bank Worths

The M2C13 and C1C12 predicted bank worth differences are slightly larger than the other cycle comparisons, although the magnitude of the differences is small and acceptable. No deficiencies were identified in either the Duke or Westinghouse nuclear models that explain the slightly larger power distribution and bank worth differences for those cycles. The differences are likely the result of code and methodology differences between SIMULATE and ANC (the Westinghouse nodal code). A review of the radial power distribution predictions showed slightly higher differences in the power distribution comparison for assemblies that operated near the periphery for more than one cycle. Both M2C13 and C1C12 contained more assemblies of these types, located at or near control rod locations, than the other benchmarked cycles. It is possible that the different spectral history treatments between ANC and SIMULATE are partially responsible for the larger differences in the predicted power distributions.

In all cases in which the acceptance criterion on differences in predicted individual bank worth were not met, Duke predicted a lower bank worth than Westinghouse. Furthermore, all measured individual bank worths easily met the acceptance criterion.

Total Worths of All Banks

The differences between the Duke and Westinghouse predicted total bank worths met the ± 2% criterion in all but one instance (five out of six predicted worths and six out of six measured worths met the criterion). The C1C11 core exceeded the criterion with a 2.2% difference in total predicted bank worth. A review of the Duke and Westinghouse C1C11 HZP radial power distribution predictions shows that the powers of many of the assemblies in peripheral region of the core were over predicted by Westinghouse, relative to the Duke predictions. More of the control banks are located near the periphery; this tends to emphasize the contribution of the peripheral assemblies to the calculation of the total bank worth. Therefore, the trend of Westinghouse's predicted total bank worth being slightly higher than the Duke prediction is consistent with the HZP radial power distribution differences.

For all six of the benchmark cores the Duke total bank worth predictions were lower than the Westinghouse total bank worth predictions, including the single case in which the total bank worth acceptance criterion was not met. Furthermore, all measured total bank worths easily met the acceptance criterion.

Conclusions

The trend in predicted bank worth deviations is consistent with the observed differences in the predicted redial Hot Zero Power (HZP) power distribution between Duke and Westinghouse. Relative to Westinghouse, Duke typically under predicts the relative power of assemblies located near the core periphery (assemblies containing banks SA, CD, SD, and SC), and over predicts the powers of assemblies near the core interior (assemblies containing banks CC, CA and SB). Duke typically predicts lower worths for banks SA, CD, SD and SC than Westinghouse due to differences in the radial power distribution.

The four deviations represent only a small deviation from an extremely tight criterion, and they are a very small percentage of the total benchmarking data. The slight differences observed are well within the expected range for a comparison of two independent core analysis methodologies.

The DRWM results with the Reference 1 methodology are very similar to the results obtained using the previously approved Westinghouse and Duke methodologies. Most importantly, the Duke and Westinghouse measured bank worths were in excellent agreement, with the largest difference between Duke and Westinghouse total measured bank worth for the six cycles being only 0.2%. This clearly demonstrates that Duke has implemented the DRWM analytical factor methodology consistent with the Westinghouse approved methodology.

References

1. DPC-NE-1005P, *Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX*, August 2001.
2. Letter, N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), December 9, 1996.
3. Letter, N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), October 2, 1997.
4. Letter, T. H. Essig, (NRC) to N. J. Liparulo (Westinghouse), July 30, 1998.
5. DPC-NE-2012A, *Dynamic Rod Worth Measurement Using CASMO/SIMULATE,* August 1999.
6. NRC Safety Evaluation Report on DPC-NE-2012A dated Febuary 15, 2000.

Page Intentionally Left Blank

Duke Power
Nuclear Design Methodology
Using
CASMO-4/SIMULATE-3 MOX

DPC-NE-1005-NP

Revision 0

Submitted to NRC August 2001

Approved by NRC August 2004

PROPRIETARY NOTICE

Certain data in this report are proprietary to various companies, as noted below.

McGuire/Catawba data	Duke
St. Laurent B1 data	Electricite de France
EPICURE and ERASME data	Framatome Advanced Nuclear Power
Calculation results and statistics	Duke

Proprietary data is bracketed and identified with a subscript (D - Duke, E - Electricité de France , and F – Framatome) indicating which company considers the data proprietary, e.g., $[xxx]_{D.}$ Where the information is proprietary to more when than one company, multiple subscripts are used, e.g., $[xxx]_{D,\,F.}$

ABSTRACT

This report presents alternative models for calculating nuclear physics data for the McGuire and Catawba nuclear units. The new models are based on the CASMO-4 /SIMULATE-3 MOX software package. The report provides benchmark comparisons to operating data from McGuire and Catawba fuel cycles with low-enriched uranium cores, Saint Laurent B1 fuel cycles with mixed cores of LEU and mixed oxide fuel assemblies, and data from critical experiments. These benchmark comparisons characterize the fidelity of the models for both low enriched uranium and mixed oxide fuels. From this benchmarking a set of biases and uncertainty factors are developed that are used in different aspects of reactor core reload design and plant operation. These biases and uncertainty factors can be updated if necessary using the methodology described in this report as new operating data is collected from subsequent McGuire and Catawba fuel cycles.

This report also describes the use of the CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX code package in the measurement of control rod worths using the dynamic rod worth measurement methodology.

Duke Power intends to use the models and methods described in this report for performing nuclear design calculations on McGuire and Catawba reactor cores containing low enriched uranium fuel and cores containing a mixture of low enriched uranium and mixed oxide fuel.

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES

Page

LIST OF FIGURES·

LIST OF FIGURES

Page

LIST OF FIGURES

Page

ACRONYMS

Acronym	Meaning
AIC	Silver-indium-cadmium
ARO	All rods out
B&W	Babcock & Wilcox
BOC	Beginning of cycle
BP	Burnable poison
BWR	Boiling water reactor
CEA	(French) Commissariat à l'Énergie Atomique
CNS	Catawba Nuclear Station
DRWM	Dynamic rod worth measurement
EDF	Electricité de France
F	Fahrenheit
GWd	Gigawatt days
HFP	Hot full power
HZP	Hot zero power
IFBA	Integral fuel burnable absorber
ITC	Isothermal temperature coefficient
Kw/ft	Kilowatts per foot
LEU	Low-enriched uranium
MeV	Million electron volts
Mlb/hr	Million pounds per hour
MNS	McGuire Nuclear Station
MSMG	Mid-span mixing grids
MOX	Mixed oxide
Mthm	Metric tons heavy metal
MWd	Megawatt days
MWe	Megawatts electric
MWt	Megawatts thermal
NRC	Nuclear Regulatory Commission
OD	Outside diameter

ACRONYMS

Acronym	Meaning
ONRF	Observed nuclear reliability factor
pcm	Percent mille
ppm	Parts per million
ppmb	Parts per million boron
PWR	Pressurized water reactor
RG	Reactor grade
SCUF	Statistically combined uncertainty factor
SLB1	Saint Laurent B1
SPND	Self-powered neutron detector
VIP	VENUS International Program
WG	Weapons grade
YAEC	Yankee Atomic Electric Corporation

1.0 INTRODUCTION

The design of a commercial pressurized water reactor core determines the characteristics of a specific number of fuel assemblies which are generally similar in design but differ in the amount of fissile material content. The refueling of a reactor core involves removing some of the fuel assemblies and replacing them with fresh fuel and possibly previously burned fuel assemblies. In a reload core the fuel enrichment, burnup, and burnable absorber content may be different for each fuel assembly in the core. In general, the neutronic and operating parameters of the new core are different from the previous core. The reload design analysis defines the characteristics of the new core and confirms that it can be operated safely while meeting design power generation requirements.

Neutronic analyses are performed to define the number of feed assemblies, their enrichment, burnable poison loading, and the arrangement of fuel and control components within the reactor core. Calculations are performed which verify core safety parameters, determine reactor protection system setpoints, and provide necessary startup and operational information. This report presents a state of the art package of analytical models which may be used to develop these analyses. The fidelity of the analytical models is demonstrated by comparison of calculated nuclear parameters to available measurements from power reactor operation and laboratory experiments.

Duke Power currently performs reload design analysis for the McGuire and Catawba nuclear stations with methodologies defined by References 1 through 7. Reference 1 describes the overall reload design methodology. Reference 2 describes the current core physics methodology which uses CASMO-3/SIMULATE-3 analytical models. Reference 3 describes Duke Power's current Nuclear Regulatory Commission (NRC)-approved methodology for performing dynamic rod worth measurements. References 4 through 7 and Reference 36 address other specific aspects of reload design for the McGuire and Catawba nuclear units.

As part of a continuous effort to improve design methods and to prepare for the use of mixed oxide (MOX) fuel, the use of CASMO-4/SIMULATE-3 MOX is presented in this report. Section 2 describes the CASMO-4, CMS-LINK, SIMULATE-3 MOX, and SIMULATE-3K MOX computer codes that are used in this reload design methodology. Section 3 presents benchmarks of the methodology against power reactor data and demonstrates the ability of the methodology to predict core physics parameters and power distributions in low enriched uranium (LEU) fueled cores as well as cores containing a mixture of LEU and MOX fuel. Section 4 presents benchmarks of the methodology against critical experiment data and demonstrates the ability of the methodology to predict relative fuel pin power in all-LEU lattices as well as lattices containing MOX fuel pins. Section 5 describes the development of power distribution uncertainty factors for both LEU and mixed LEU-MOX cores. Section 6 presents benchmarks of the methodology against dynamic rod worth measurements and predictions and also justifies the application of dynamic rod worth measurement to mixed LEU-MOX cores. Section 7 summarizes the results and conclusions of this report. Appendix A provides a description of a typical McGuire and Catawba LEU core design and shows the currently planned fuel management pattern for mixed LEU-MOX cores at those plants.

2.0 DESCRIPTION OF ANALYTICAL MODELS

As part of the reload design process, reactor physics calculations are performed on a cycle-specific basis to develop the core nuclear design and ensure safety.
The cycle design is set by specifying the number and enrichment(s) of the feed assemblies and the core locations of the feed and reinserted assemblies. Calculations are performed to verify core safety parameters, generate operational and reactor protection system (RPS) limits, and identify the core loading pattern. Calculations are also performed to support startup testing, including rod worth measurement, and for core follow activities during reactor operation. Details of these calculations have previously been described in References 1, 3, 4, 5, and 36.

This section provides a brief description of the CASMO-4/SIMULATE-3 MOX computer codes and the supporting programs that are used to perform the above calculations. The NRC has approved Duke's current reactor physics calculation methodology, which includes the use of CASMO-3/SIMULATE-3. The methodology described in this report is basically the same as Duke's current methodology with the substitution of the four codes listed below. These codes contain improved features for both LEU and partial MOX fuel cores as discussed in Sections 2.1 and 2.3.

The core modeling package is made up of four computer programs:

- CASMO-4
- CMS-LINK
- SIMULATE-3 MOX
- SIMULATE-3K MOX

These programs were developed by Studsvik Scandpower Incorporated. Various forms of these computer programs have a long history of utilization in both the United States and international nuclear industries. In the calculation sequence, CASMO-4 generates nuclear data for each unique fuel assembly lattice. CMS-LINK collects this data into a single library for use by SIMULATE-3 MOX and SIMULATE 3K MOX. SIMULATE-3 MOX is utilized to deplete the fuel cycle and predict critical boron concentration, rod worth, reactivity coefficients, core power distribution, as well as for 3-D analysis for generation of operational and reactor protection system limits. SIMULATE-3K MOX is used to model core transients and support the dynamic rod worth measurement technique.

2.1 CASMO-4

CASMO-4 is a multi-group, two dimensional transport theory model for burnup calculations on fuel assemblies or fuel pin cells as described in References 8 and 9. The code accommodates a geometry consisting of cylindrical rods of varying composition in a square pitch array. CASMO-4 can model fuel pins, burnable absorber rods, control rods, guide tubes, in-core instruments, water gaps, and reflectors. The nuclear data library input to CASMO-4 is based mainly on data from ENDF/B-IV. It contains cross sections for more than 100 materials commonly found in light water reactors. The cross sections are collected into 70 energy groups covering neutron energies from 0 to 10 million electron volts (MeV). CASMO-4 supports NRC-approved methodologies at Palo Verde Nuclear Station and Prairie Island Nuclear Station.

Important new features of CASMO-4 over CASMO-3 are the incorporation of microscopic depletion of burnable absorbers into the main calculation, use of a geometrically heterogeneous model for the entire calculation, and use of the characteristics method for solving the transport equation. CASMO-4 provides a convenient method for describing MOX fuel compositions and Pu^{241} decay time. For a MOX fuel lattice the program automatically adjusts the detail of appropriate internal calculations to accommodate the larger variation of plutonium cross sections in the thermal energy region and the presence of significant plutonium resonances in the epithermal energy region. CASMO-4 also edits several additional coefficients which are required by the modified nodal methods used in SIMULATE-3 MOX.

A series of CASMO-4 cases is executed for each unique fuel assembly lattice configuration. A typical case set characterizes the effect of fuel burnup, moderator temperature, fuel temperature, soluble boron concentration, and control rod presence.

For core reflector regions the impact of changes in moderator temperature and soluble boron concentration are typically modeled.

2.2 CMS - LINK

CMS-LINK processes data generated by CASMO-4 and produces a nuclear data library for input to the SIMULATE-3 MOX core model as described in Reference 10. The code collects the following data for each unique fuel lattice configuration.

- Macroscopic cross sections in two energy groups
- Discontinuity factors at fuel assembly boundaries in two energy groups
- Yields and microscopic cross sections for important fission products
- Incore detector constants
- Kinetics data
- Pin by pin power distributions

For any fuel type used in mixed cores of MOX and LEU fuels, the program also collects additional data required by the nodal methods used in SIMULATE-3 MOX.

The data is collected into multi-dimensional tables that characterize the effect of both instantaneous and integrated perturbations to local core conditions. The precise functionalization of the data varies depending on the type of data and the amount that a given data type changes as core conditions change.

2.3 SIMULATE-3 MOX

SIMULATE-3 MOX is a three-dimensional diffusion theory reactor core simulator described in Reference 11. The program calculates core wide power distribution and fuel depletion with macroscopic cross sections in two energy groups. The nodal solution is performed on a geometric mesh of either one or four nodes per assembly in the radial plane and an appropriate axial mesh in the active fuel column. Explicit models of top, bottom, and radial reflector regions allow analytic solutions for flux and leakage at the core boundary. A microscopic depletion model is used to track iodine, xenon, promethium, and samarium during anticipated core transients. Pin power distributions are constructed by synthesizing results of the nodal mesh solution with heterogeneous lattice solutions extracted from CASMO-4.

SIMULATE-3 MOX is an extension of the standard SIMULATE program as described in Reference 12. The CASMO-4/SIMULATE-3 MOX programs are used in different capacities in Germany, Switzerland, the United Kingdom, and Japan. Reference 37 provides a summary of work that has been performed to validate the CASMO-4/SIMULATE-3 MOX codes for MOX fuel applications in Japan. The modifications required are not due to the MOX fuel itself but are necessary to more accurately model the interaction of MOX fuel with adjacent LEU fuel assemblies. The large difference in thermal absorption cross sections of MOX and LEU fuel causes steep thermal flux gradients at the fuel assembly interface. Changes made to accommodate these flux gradients are discussed briefly below.

SIMULATE-3 MOX uses a transverse integration procedure to reduce the multi-dimensional diffusion equations to a set of coupled one-dimensional equations. For non-MOX problems, SIMULATE-3 MOX solves the one-dimensional equations by representing the flux with fourth-order polynomial expansions and then using weighted residual methods to determine the coefficients for each of the two energy groups. For problems with very large flux gradients such as face adjacent MOX and LEU fuel assembles, polynomial expansions may not accurately model dramatic spatial changes in neutron flux. SIMULATE-3 MOX supplements the polynomial expansion method with additional terms derived from purely analytic nodal solution methods. The fast flux is represented by a polynomial expansion and the thermal flux is represented by an expansion containing both polynomial and hyperbolic terms. The hyperbolic terms allow very large changes in thermal flux level to be characterized more accurately than can be accomplished with only polynomial expansions.

The use of single assembly lattice calculations to produce homogenized cross sections for the nodal model can introduce errors when the single assembly spatial flux shape is dramatically different from the actual flux distribution across the assembly. SIMULATE-3 MOX reduces this spatial homogenization error by recalculating homogeneous two group cross sections with the actual local flux shape determined for the reactor configuration. In a general sense this re-homogenization of cross sections is consistent with the traditional technique of superpositioning intra-nodal flux shapes and single assembly flux form functions to construct accurate predictions of individual pin power distributions.

Several modifications are made to pin power reconstruction techniques in SIMULATE-3 MOX to more accurately model the impact of local flux gradients at MOX-LEU fuel assembly interfaces. An improved method of estimating nodal corner point fluxes makes use of empirically determined coefficients. Conventional reconstruction methods used a single total power form function from CASMO-4. This approach works well when the fast to thermal flux ratio is relatively constant throughout the core. In MOX fuel, interactions with fast neutrons produce three times as much power as fast neutron interactions in LEU fuel. SIMULATE-3 MOX accounts for this imbalance by utilizing separate CASMO-4 form functions for each neutron energy group.

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.

2.4 SIMULATE-3K MOX

SIMULATE-3K MOX is an extension of SIMULATE-3K (References 13 and 14), which is used for analysis of core transients. The spatial neutronics models in SIMULATE-3K MOX are identical to those in SIMULATE-3 MOX. SIMULATE-3K MOX solves the transient neutron diffusion equation incorporating effects of delayed neutrons, spontaneous fission in fuel, alpha-neutron interactions from actinide decay, and gamma-neutron interactions from long term fission product decay. The thermal-hydraulics module consists of a fuel pin heat conduction model, fission product decay heat generation model, and a channel hydraulics model.

The fuel pin conduction model calculates the radial temperature distribution and the fuel pin surface heat flux using a finite difference model of the nonlinear cylindrical heat conduction equation. An explicit fuel pin conduction calculation is performed for the average fuel pin in each nodal mesh, and optionally for the hot pin in each fuel assembly. The axial nodalization of the fuel pin conduction solution is identical to that of the neutronics model. Fuel, gap, and clad thermal properties are treated as functions of node-averaged fuel pin burnup and local temperature. Convective heat transfer coefficients are computed using regime-dependent correlations. The coupling between the pin conduction calculation and the heat transfer coefficient calculation is fully resolved at each time step by nonlinear iteration.

An explicit hydraulic calculation is performed for each nodal mesh, using the average fuel pin heat flux and hydraulic characteristics of the node. The axial nodalization of the hydraulic solution is identical to that of the neutronics model. For pressurized water reactor (PWR) applications, SIMULATE-3K MOX utilizes a fully-implicit, five-equation hydraulics model (liquid mass and energy, vapor mass and energy, and mixture momentum).

The SIMULATE-3K MOX neutronics model uses the same nuclear data library as SIMULATE-3 MOX. The thermal and hydraulic models are coupled to the neutronics model via the fuel pin heat generation rate which is directly determined from the calculated neutron power. In turn, the thermal hydraulics module provides the nuclear calculation with the appropriate hydraulic data to permit nuclear feedback with local thermal conditions. Boundary conditions for the hydraulic calculations are defined by moderator core inlet conditions and upper plenum pressure.

SIMULATE-3K MOX is capable of modeling core transients initiated by changes in soluble boron concentration, control rod placement, moderator temperature, moderator flow, and/or system pressure. Incore and excore instrumentation may be modeled for the purpose of driving the reactor control system and allowing realistic comparison to actual core transients. SIMULATE-3K MOX is a best estimate model by nature, however conservatism may be applied via individual scalar multipliers to important parameters such as fuel conductivity, specific heat, gap conductance, convective heat transfer, fuel temperature, moderator temperature, void fraction, delayed neutron yields, and control rod worths.

[Page intentionally left blank]

3.0 POWER REACTOR BENCHMARK ANALYSES

3.1 McGuire and Catawba Benchmark Analysis

This section compares measured core physics parameters from McGuire Nuclear Station (MNS) and Catawba Nuclear Station (CNS) to predictions from the SIMULATE-3 MOX analytical model. Comparisons are made for the following recent operating cycles:

MNS Unit 1 Cycles 12, 13, 14	CNS Unit 1 Cycles 11, 12, 13
MNS Unit 2 Cycles 12, 13, 14	CNS Unit 2 Cycles 9, 10, 11

Measurements of critical boron concentration, control rod bank worth, and isothermal temperature coefficient are made during initial startup of each fuel cycle at hot zero power (HZP) conditions. Measurements of critical boron concentration and core wide power distribution are made throughout the depletion of each fuel cycle at nominal hot full power (HFP) operating conditions.

3.1.1 Description of Reactors

MNS and CNS are operated by Duke Power, a division of Duke Energy, and are located within 30 miles of Charlotte, North Carolina. Each reactor is a four loop pressurized water reactor operating at 3411 megawatts thermal (MWt) and approximately 1215 megawatts electric (MWe). Average moderator temperature at HFP is approximately 586 °F. Each reactor core contains 193 fuel assemblies and 53 control rod clusters. The MNS/CNS core configuration is shown in Figure 3-1. Each fuel assembly is comprised of a 17x17 square lattice having 264 fuel pins, 24 guide tubes, and a central instrument tube. In general terms all fuel cycles analyzed may be characterized as 18 month fuel cycles utilizing a low leakage fuel management technique.

3.1.2 Critical Boron Concentrations

Critical boron concentrations are measured during cycle startup testing and throughout cycle operation by an acid-base titration of a reactor coolant system sample. Critical boron concentrations are measured during startup tests at beginning of cycle (BOC) under peak samarium, no xenon, HZP conditions, with all control rods out of the core (ARO). All four reactors are operated as base-loaded units and thus most mid-cycle critical boron measurements are at near HFP nominal operating conditions. Natural boron was used in all fuel cycles analyzed. The measured full power critical boron concentrations were corrected for B^{10} depletion during operation.

Table 3-1 compares measured to predicted critical boron concentrations for BOC HZP conditions. Tables 3-2 and 3-3 compare measured to predicted critical boron concentrations for HFP conditions throughout the depletion of each fuel cycle. The deviation is defined as measured minus predicted expressed in parts per million (ppm) boron. Figure 3-2 plots the HFP deviations versus cycle burnup. The calculated results with SIMULATE-3 MOX were consistent with the performance of previously approved Duke methodologies.

3.1.3 Control Rod Worths

Individual control rod bank worths are measured at BOC HZP conditions during startup testing for each fuel cycle. Rod worth measurements for all fuel cycles except McGuire 1 Cycle 12 and Catawba 2 Cycle 9 were performed using a dynamic rod worth measurement (DRWM) technique. This is a relatively fast method that measures individual control rod bank worths by inserting and withdrawing the bank at the maximum stepping speed without changing boron concentration. Excore detector signals are processed by a reactivity computer with appropriate analytical compensation for significant space-time effects that occur during control rod insertion. A more detailed discussion of DRWM is provided in Section 6.

For McGuire 1 Cycle 12 and Catawba 2 Cycle 9, control rod bank worths were measured by the rod swap technique as described in Reference 37. This technique compensates for a continuous decrease in boron concentration by inserting the control rod bank in small, discrete steps. The change in reactivity due to each insertion was determined from reactivity computer readings before and after the insertion. These individual or differential rod worths were integrated to define a reference bank worth versus bank insertion. Other individual control rod banks were then inserted without changes in boron concentration by offsetting their worth with removal of the reference bank. The amount of reference bank withdrawal was used to infer the worth of other individual control rod banks.

Table 3-4 compares measured and predicted rod worths at BOC HZP conditions for each fuel cycle. The deviation is defined as measured minus predicted divided by the measured worth expressed in percent. The accuracy of control rod worth predictions with SIMULATE-3 MOX is very similar to the accuracy of previously approved Duke methodologies.

3.1.4 Isothermal Temperature Coefficient

Isothermal temperature coefficients (ITC) are measured at BOC, HZP, ARO conditions during startup testing for each fuel cycle. The ITC is determined by altering the average moderator temperature and measuring the change in reactivity with a reactivity computer. Table 3-5 compares measured and predicted ITC at BOC HZP conditions for selected McGuire and Catawba fuel cycles. The deviation is defined as measured minus predicted expressed as pcm per degree F. The accuracy of ITC predictions with SIMULATE-3 MOX is very similar to the accuracy of previously approved Duke methodologies.

3.1.5 Fuel Assembly Power Distribution Analysis and Uncertainty Factors

Core power distributions are measured at regular intervals during operation of each fuel cycle. The measured power distributions are derived from electrical signals produced by moveable incore fission chambers as they pass through the instrument guide tube of individual fuel assemblies during reactor operation.

The MNS/CNS incore system uses six fission chambers to make measurements in 58 instrumented locations distributed among the 193 fuel assemblies in the core. Core locations with incore instrument tubes are shown in Figure 3-1. More than 600 individual signals are recorded as the detector passes through each instrumented fuel assembly.

Raw measured signals are processed to remove clearly spurious information and any data taken above or below the active fuel column. The remaining information is normalized to account for differences in individual fission chamber performance and changes in reactor power level that may have occurred while the data was taken. The normalized signals are converted to normalized relative power by applying signal to power conversion factors that are derived from cycle specific core models. These conversion factors are dependent upon core location, burnup, and control rod presence.

The final product is a full core, assembly mesh, three-dimensional measured relative power distribution. These data are used to calculate three types of power peaking factors which characterize important radial and axial properties of the measured power distribution. Assembly $F_{\Delta h}$ or assembly radial power is simply the average relative power in each fuel assembly. Assembly F_q or assembly maximum power is the largest relative power in each assembly. Assembly F_z or assembly axial power is the assembly F_q normalized to the assembly average power ($F_z = F_q / F_{\Delta h}$) for each assembly. Measured assembly $F_{\Delta h}$, F_q, and F_z may be compared directly to equivalent edits generated by SIMULATE-3 MOX.

SIMULATE-3 MOX is used to model reactor conditions for 74 power distribution measurements taken during operation of 12 MNS/CNS fuel cycles. Comparison of measured and predicted peaking factors define the relative error in the predicted value for each fuel assembly in each power distribution measurement. One-sided upper tolerance limit uncertainties are developed to insure with a 95% confidence level that 95% of local power predictions are equal to or larger than the measured value. This statistical method requires that the data set pass a test for normality which is performed at a 1% level of significance. If a given data set fails this normality test, a conservatively large uncertainty is determined by a non-parametric evaluation of the data. These statistical methods are described in References 15 through 18.

Representative comparisons of calculated and measured assembly average power for MNS and CNS are provided in Figures 3-3 through 3-14. Biases and uncertainties are derived by comparing the calculated power to the measured power for all 74 measured power distributions. Observed nuclear reliability factors (ORNFs) or assembly uncertainty factors for $F_{\Delta h}$, F_q, and F_z are then calculated using the following expression:

$$ONRF = 1 - bias + K_a\sigma_a$$

where $K_a\sigma_a$ is the statistical deviation of the calculated to measured power comparisons. These values are summarized in Table 3-12.

3.2 St. Laurent Benchmark Analysis

This section compares measured core physics parameters from Saint Laurent B1 (SLB1) Cycles 5 through 10 (References 19 and 20) to predictions from the SIMULATE-3 MOX analytical model. Measurements of critical boron concentration, control rod bank worth, and isothermal temperature coefficient are made during startup testing for each fuel cycle at HZP conditions. Measurements of critical boron concentration and core wide power distribution are made throughout the depletion of each fuel cycle at nominal HFP operating conditions.

3.2.1 Description of Reactor

St. Laurent B1 is operated by Electricité de France (EDF) and is located in north central France. It is a three loop pressurized water reactor operating at 2775 MWt and 915 MWe. Average moderator temperature at HFP is approximately 580 °F. The reactor core contains 157 fuel assemblies and 57 control rod clusters. The St. Laurent B1 core configuration is shown in Figure 3-15. Each fuel assembly is comprised of a 17x17 square lattice having 264 fuel pins, 24 guide tubes, and a central instrument tube. In general terms the first 10 cycles of St. Laurent B1 may be characterized as annual fuel cycles utilizing an out-in-in fuel management technique with MOX fuel assemblies loaded at least one assembly in from the periphery. Assemblies containing MOX fuel were first introduced in Cycle 5 with initial startup in November 1987. A typical reload consists of 36 LEU and 16 MOX fuel assemblies. All MOX fuel assemblies are burned in three fuel cycles before permanent discharge from the core. Key reactor characteristics for St. Laurent B1 are compared to McGuire and Catawba in Table 3-6.

Except for the number of fuel assemblies, St. Laurent B1 core components are very similar to MNS and CNS. Fuel assembly, control rod, and core structural materials for the three stations are neutronically similar. The arrangement of the fuel assembly lattice and the locations of guide tubes, instrumentation tube, and spacer grids are very similar to MNS and CNS. The design and function of incore instrumentation in the St. Laurent B1 reactor are equivalent to the Duke reactors. Moderator temperature and pressure are essentially equal in each reactor. Thus the interaction of LEU and MOX fuel in the St. Laurent B1 core is representative of the behavior that is expected for MOX fuel use in the MNS and CNS reactors.

3.2.2 Critical Boron Concentration

Critical boron concentrations are measured during cycle startup testing and throughout cycle operation by an acid-base titration of a reactor coolant system sample. The measurements are made during startup testing at BOC, peak samarium, no xenon, HZP conditions, with ARO and with the regulating control rod bank (Bank R) inserted. The St. Laurent B1 reactor was operated as a base-loaded unit in Cycles 5 through 10 and thus many mid-cycle critical boron measurements are at near-HFP nominal operating conditions. St. Laurent B1 used natural boron and did not recycle soluble boron during these fuel cycles. The measured critical boron concentrations were not corrected for B^{10} depletion during operation.

Table 3-7 compares measured to predicted critical boron concentrations for BOC HZP conditions. Table 3-8 compares measured to predicted critical boron concentrations for HFP conditions throughout the depletion of each fuel cycle. The deviation is defined as measured minus predicted expressed in ppm boron. Figure 3-16 shows the HFP critical boron deviations versus cycle burnup. These results are consistent with past experience on Duke reactors using exclusively LEU fuel and previously approved methodologies.

3.2.3 Control Rod Worth

Individual control rod bank worths are measured at BOC HZP conditions by the rod swap technique during startup tests for each fuel cycle. The control rod bank with the highest predicted worth is measured with a boron dilution technique. This technique compensates for a continuous decrease in boron concentration by inserting the control rod bank in small, discrete steps. The change in reactivity due to each insertion is determined from reactivity computer readings before and after the insertion. These differential rod worths were integrated to define a reference bank worth versus bank insertion. Other individual control rod banks were then inserted without changes in boron concentration by offsetting their worth by removal of the reference rod bank. The amount of reference bank withdrawal was used to infer the worth of other individual control rod banks.

Varying numbers of MOX fuel assemblies were placed under control rods in each of the St. Laurent B1 fuel cycles. Some of the fuel assemblies under the reference bank in Cycles 8, 9, and 10 were burned MOX assemblies. Control rod worth was measured in rods that were in fresh and burned MOX fuel assemblies.

Table 3-9 compares measured and predicted rod worths at BOC HZP conditions for each fuel cycle containing MOX fuel. The deviation is defined as measured minus predicted divided by the measured worth expressed in percent. The quality of control rod worth predictions for St. Laurent B1 is consistent with past experience on Duke reactors using exclusively LEU fuel and previously approved methodologies. The accuracy of predicted control rod worths is not significantly affected by the introduction of MOX fuel.

3.2.4 Isothermal Temperature Coefficient

Isothermal temperature coefficients are measured at BOC HZP conditions during startup tests for each fuel cycle. ITC is measured with ARO and with control rod Bank R fully inserted. The ITC is determined by altering the average moderator temperature and measuring the change in reactivity with a reactivity computer. Table 3-10 compares measured and predicted ITC at BOC HZP conditions for each fuel cycle containing MOX fuel. The deviation is defined as measured minus predicted expressed as pcm per °F. The accuracy of predicted ITC is not affected by the introduction of MOX fuel.

3.2.5 Fuel Assembly Power Distribution Analysis and Uncertainty Factors

Measured core power distributions are determined at regular intervals during operation of each fuel cycle. The measured power distributions are derived from electrical signals produced by moveable incore fission chambers as they pass through the instrument guide tube of individual fuel assemblies during reactor operation. The St. Laurent B1 incore system and fission chambers are very similar, both in terms of design and performance, to McGuire and Catawba systems.

The St. Laurent B1 incore system uses five fission chambers to make measurements in 50 instrumented locations distributed among the 157 fuel assemblies in the core. More than 500 individual signals are recorded as the detector passes through each instrumented fuel assembly. The measured power distributions used in the St. Laurent B1 benchmark analysis were reconstructed from these raw signals with the same general methods used for the McGuire and Catawba reactors. This ensured a consistent comparison of measured and predicted power distribution information among the five reactors.

Raw measured signals were processed to remove clearly spurious information and any data taken above or below the active fuel column. The remaining information was normalized to account for differences in individual fission chamber performance and changes in reactor power level that may have occurred while the data was taken. The significantly larger neutron absorption cross section of MOX fuel results in fission chamber signals that were 1/3 to 1/2 of those from LEU fuel. This means that the relative importance of gamma and background signals varies depending on fuel type. A small bias was applied to measured signals from MOX core locations to account for these effects. The normalized signals were converted to normalized relative power by applying signal to power conversion factors that were derived from cycle specific core models. These conversion factors were dependent upon core location, burnup, and control rod presence.

The final product was a full core, assembly mesh, three-dimensional measured relative power distribution. This data was used to calculate three types of power peaking factors which characterize important radial and axial properties of the measured power distribution. Assembly $F_{\Delta h}$ or assembly radial power is simply the average relative power in each fuel assembly. Assembly F_q or assembly maximum power is the largest relative power in each assembly. Assembly F_z or assembly axial power is the assembly F_q normalized to the assembly average power ($F_z = F_q / F_{\Delta h}$) for each assembly. Measured assembly $F_{\Delta h}$, F_q, and F_z may be compared directly to equivalent edits generated by SIMULATE-3 MOX.

SIMULATE-3 MOX was used to model reactor conditions for 58 power distribution measurements taken during operation of St. Laurent B1 Cycles 5 through 10. Comparison of measured and predicted peaking factors defined the relative error in the predicted value for each fuel assembly in each power distribution measurement. One sided upper tolerance limit uncertainties were developed to insure with a 95% confidence level that 95% of local power predictions were equal to or larger than the measured value. This statistical approach requires that the data set pass a test for normality which was performed at a 1% level of significance. If a given data set fails this normality test, a conservatively large uncertainty is determined by a non-parametric evaluation of the data. These statistical methods are described in References 15 through 18.

Representative comparisons of calculated and measured assembly average power for St. Laurent B1 are provided in Figures 3-17 through 3-22. Biases and uncertainties are derived by comparing the calculated power to the measured power for all 58 measured power distributions. ONRFs or assembly uncertainty factors for $F_{\Delta h}$, F_q, and F_z are then calculated using the following expression:

$$ONRF = 1 - \text{bias} + K_a\sigma_a$$

where $K_a\sigma_a$ is the statistical deviation of the calculated to measured power comparisons. These values are summarized in Table 3-12.

3.3 Summary Comparison of Benchmark Results

The average deviation between measured and calculated values and the associated standard deviation for each of the four reactor physics parameters evaluated (HZP critical boron concentration, HFP boron concentration, control rod worth, isothermal temperature coefficient) were determined for both the McGuire/Catawba and St. Laurent B1 benchmark calculations. These deviations are shown in Table 3-11. The average and standard deviations calculated for McGuire/Catawba fuel cycles with LEU fueled cores are consistent with the average and standard deviations for St. Laurent B1 partial MOX fuel cores. These results demonstrate the ability of CASMO-4/SIMULATE-3 MOX to adequately model the behavior of partial MOX fuel cores.

Excellent results were also obtained from the power distribution benchmark analyses. The assembly uncertainty factors or ORNFs for $F_{\Delta h}$, F_q, and F_z that were developed from comparisons of the McGuire, Catawba, and St Laurent B1 measured power distribution data and CASMO-4/SIMULATE-3 MOX models are summarized in Table 3-12. As new operating data are collected from subsequent McGuire and Catawba fuel cycles these values can be updated if necessary using the methodology described in this report.

Table 3-1 McGuire & Catawba Beginning of Cycle Hot Zero Power Critical Soluble Boron Comparisons

Unit Cycle	All CR Out		Cycle Number	All CR Out
M1 C12		Measured PPMB	C1 C11	
		Predicted PPMB		
		PPMB Deviation		
M1 C13		Measured PPMB	C1 C12	
		Predicted PPMB		
		PPMB Deviation		
M1 C14		Measured PPMB	C1 C13	
		Predicted PPMB		
		PPMB Deviation		
M2 C12		Measured PPMB	C2 C09	
		Predicted PPMB		
		PPMB Deviation		
M2 C13		Measured PPMB	C2 C10	
		Predicted PPMB		
		PPMB Deviation		
M2 C14		Measured PPMB	C2 C11	
		Predicted PPMB		
	D	PPMB Deviation		D

Table 3-2 McGuire Hot Full Power Critical Soluble Boron Comparisons vs Cycle Burnup

McGuire 1 Cycle 12

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
13			
35			
62			
93			
117			
145			
172			
202			
230			
260			
282			
309			
335			

D

McGuire 1 Cycle 13

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
6			
36			
65			
91			
121			
147			
175			
204			
232			
258			
288			
316			
342			
372			
400			
428			

D

McGuire 1 Cycle 14

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
5			
22			
50			
78			
106			
134			
162			
191			
215			
243			
271			
299			
327			
355			
383			
404			

D

McGuire 2 Cycle 12

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
8			
36			
70			
82			
112			
138			
166			
194			
215			
243			
271			
299			
327			
356			
383			
411			

D

McGuire 2 Cycle 13

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
5			
24			
45			
73			
100			
128			
155			
184			
212			
234			
256			
272			
294			
313			
342			
370			
391			
419			

D

McGuire 2 Cycle 14

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
6			
31			
57			

D

Table 3-3 Catawba Hot Full Power Critical Soluble Boron Comparisons vs Cycle Burnup

Catawba 1 Cycle 11

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
5			
32			
59			
60			
88			
119			
124			
149			
170			
194			
207			
225			
252			
280			
308			
335			
364			
392			
419			

D

Catawba 1 Cycle 12

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
5			
21			
43			
64			
92			
120			
148			
174			
202			
230			
258			
283			
311			
339			
367			
395			
423			
451			
479			

D

Catawba 2 Cycle 9

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
7			
24			
52			
77			
102			
129			
157			
185			
212			
238			
268			
296			
321			
324			
352			
380			
408			
436			
464			

D

Catawba 2 Cycle 10

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
7			
25			
52			
64			
81			
108			
136			
164			
191			
219			
239			
268			
295			
323			
349			
377			
398			
426			
441			

D

Catawba 2 Cycle 11

Cycle EFPD	Meas PPMB	Pred PPMB	PPMB Deviation
1			
7			
29			
57			
72			
81			
109			
137			
165			
193			
218			
247			

D

Table 3-4 McGuire and Catawba Beginning of Cycle Hot Zero Power Control Rod Worth Comparisons

Cycle Number		Control Rod Bank									Total Worth	
		C D	C C	C B	C A	S E	S D	S C	S B	S A		
M1 C12	Measured, pcm											D
	Predicted, pcm											
	% Deviation											
M1 C13	Measured, pcm	587	731	693	299	529	468	477	1033	271	5088	
	Predicted, pcm											D
	% Deviation											
M1 C14	Measured, pcm											D
	Predicted, pcm											
	% Deviation											
M2 C12	Measured, pcm	620	761	660	293	485	508	505	1067	303	5204	
	Predicted, pcm											D
	% Deviation											
M2 C13	Measured, pcm	595	816	680	339	501	473	461	979	278	5122	
	Predicted, pcm											D
	% Deviation											
M2 C14	Measured, pcm											D
	Predicted, pcm											
	% Deviation											
C1 C11	Measured, pcm	688	857	627	370	456	458	460	883	235	5063	
	Predicted, pcm											D
	% Deviation											
C1 C12	Measured, pcm											D
	Predicted, pcm											
	% Deviation											
C1 C13	Measured, pcm											D
	Predicted, pcm											
	% Deviation											
C2 C09	Measured, pcm											D
	Predicted, pcm											
	% Deviation											
C2 C10	Measured, pcm	556	886	596	374	473	400	400	1001	237	4924	
	Predicted, pcm											D
	% Deviation											
C2 C11	Measured, pcm											D
	Predicted, pcm											
	% Deviation											

Table 3-5 McGuire & Catawba Beginning of Cycle Hot Zero Power Isothermal Temperature Coefficient Comparisons

Unit Cycle			Unit Cycle	
M1 C12		Measured, pcm / F	C1 C11	
		Predicted, pcm / F		
		Deviation, pcm / F		
M1 C13		Measured, pcm / F	C1 C12	
		Predicted, pcm / F		
		Deviation, pcm / F		
M1 C14		Measured, pcm / F	C1 C13	
		Predicted, pcm / F		
		Deviation, pcm / F		
M2 C12		Measured, pcm / F	C2 C09	
		Predicted, pcm / F		
		Deviation, pcm / F		
M2 C13		Measured, pcm / F	C2 C10	
		Predicted, pcm / F		
		Deviation, pcm / F		
M2 C14		Measured, pcm / F	C2 C11	
		Predicted, pcm / F		
		Deviation, pcm / F		
	D			D

Table 3-6 McGuire/Catawba and Saint Laurent B1 Reactor Characteristics

	St. Laurent B1	McGuire / Catawba
Core Thermal Power, MW	2775	3411
Number of Fuel Assemblies	157	193
Fuel Pin Pitch, inches	0.496	0.496
Fuel Assembly Pitch, inches	8.47	8.47
Fuel Pin Array	17 x 17	17 x 17
Fuel Pins per Assembly	264	264
Active Fuel Height, inches	144	144
Core Average Linear Heat Rate, kw/ft	5.58	5.58
Core Inlet Moderator Temperature, F	550	555
Core Average Moderator Temperature, F	580	586
Primary System Pressure, psia	2250	2250
Total Core Flow, Mlb/hr	113	139
Average Assembly Flow, Mlb/hr	0.717	0.718
Incore Instrumentation	Highly Enriched U3O8	Highly Enriched U3O8
Number of Instrumented Locations	50	58
Control Rod Materials	AIC / SS	AIC / B4C
Number of Control Rods	57	53

Table 3-7 St. Laurent B1 Beginning of Cycle Hot Zero Power Critical Soluble Boron Comparisons

Cycle Number		All CR Out	CR Bank R Inserted
5	Measured PPMB	[	]E
	Predicted PPMB		
	PPMB Deviation		]D
6	Measured PPMB	[	]E
	Predicted PPMB		
	PPMB Deviation		]D
7	Measured PPMB	[	]E
	Predicted PPMB		
	PPMB Deviation		]D
8	Measured PPMB	[	]E
	Predicted PPMB		
	PPMB Deviation		]D
9	Measured PPMB	[	]E
	Predicted PPMB		
	PPMB Deviation		]D
10	Measured PPMB	[	]E
	Predicted PPMB		
	PPMB Deviation		]D

Table 3-8 St. Laurent B1 Hot Full Power Critical Soluble Boron Comparison vs Cycle Burnup

Cycle 5

Burnup MWd/Mthm	Meas PPMB	Pred PPMB	PPMB Deviation
142			
527			
1062			
2151			
3200			
4349			
5467			
6494			
7580			
8581			
9366			

Cycle 6

Burnup MWd/Mthm	Meas PPMB	Pred PPMB	PPMB Deviation
840			
1650			
2033			
2639			
3179			
4341			
5368			
6523			
7000			
8161			
9455			
10145			
11026			

Cycle 7

Burnup MWd/Mthm	Meas PPMB	Pred PPMB	PPMB Deviation
356			
1519			
2876			
4172			
5400			
6400			
7323			
8470			
9595			
10370			
10548			

Cycle 8

Burnup MWd/Mthm	Meas PPMB	Pred PPMB	PPMB Deviation
313			
1080			
2346			
3376			
4442			
5516			
6222			
7301			
8333			
9174			
10207			
10972			

Cycle 9

Burnup MWd/Mthm	Meas PPMB	Pred PPMB	PPMB Deviation
170			
702			
968			
1409			
2437			
3381			
4520			
5740			
6828			
7886			
8940			
10153			
10498			

Cycle 10

Burnup MWd/Mthm	Meas PPMB	Pred PPMB	PPMB Deviation
154			
1066			
1822			
3111			
3874			
4990			
5943			
6971			
7997			
8898			
10023			

Table 3-9 St. Laurent B1 Beginning of Cycle Hot Zero Power Control Rod Worth Comparisons

Control Rod Bank								
R	G 1	G 2	N 1	N 2	S B	S C	S A + S D	Total Worth
								E
								D
								E
								D
								E
								D
								E
								D
								E
								D
								E
								D

Table 3-10 St. Laurent B1 Beginning of Cycle Hot Zero Power Isothermal Temperature Coefficient Comparisons

Cycle Number		All CR Out	CR Bank R Inserted	
5	Measured, pcm / F			E
	Predicted, pcm / F			
	Deviation, pcm / F			D
6	Measured, pcm / F			E
	Predicted, pcm / F			
	Deviation, pcm / F			D
7	Measured, pcm / F			E
	Predicted, pcm / F			
	Deviation, pcm / F			D
8	Measured, pcm / F			E
	Predicted, pcm / F			
	Deviation, pcm / F			D
9	Measured, pcm / F			E
	Predicted, pcm / F			
	Deviation, pcm / F			D
10	Measured, pcm / F			E
	Predicted, pcm / F			
	Deviation, pcm / F			D

Table 3-11 Summary Comparison of Benchmark Results

	Average Deviation	Standard Deviation
McGuire/Catawba		
BOC HZP Soluble Boron (ppmb)	[ ]D	[ ]D
HFP Soluble Boron (ppmb)	[ ]D	[ ]D
BOC HZP Control Rod Worth (%)	[ ]D	[ ]D
BOC HZP ITC (pcm / °F)	[ ]D	[ ]D
St. Laurent B1		
BOC HZP Soluble Boron (ppmb)	[ ]D	[ ]D
HFP Soluble Boron (ppmb)	[ ]D	[ ]D
BOC HZP Control Rod Worth (%)	[ ]D	[ ]D
BOC HZP ITC (pcm / °F)	[ ]D	[ ]D

Table 3-12 Assembly Uncertainty Factors for McGuire/Catawba and St Laurent B1 Reactor Cores

Parameter	Bias	Statistical Deviation ($K_a\sigma_a$)	Assembly Uncertainty Factor (ONRF)
MNS/CNS LEU Fuel			
$F_{\Delta h}$	[ ]D	[ ]D	[ ]D
F_q	[ ]D	[ ]D	[ ]D
F_z	[ ]D	[ ]D	[ ]D
St Laurent B1 LEU Fuel			
$F_{\Delta h}$	[ ]D	[ ]D	[ ]D D
F_q	[ ]D	[ ]D	[ ]D
F_z	[ ]D	[ ]D	[ ]D
St Laurent B1 MOX Fuel			
$F_{\Delta h}$	[ ]D	[ ]D	[ ]D
F_q	[ ]D	[ ]D	[ ]D
F_z	[ ]D	[ ]D	[ ]D

Figure 3-1 McGuire/Catawba Core Configuration

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

Incore Detector Location
Control Rod Bank Location

Figure 3-2 McGuire and Catawba Hot Full Power Boron Deviations (ppm)

D

Figure 3-3 McGuire-1 Cycle 12 Assembly Average Power Distributions

D

Figure 3-4 McGuire Unit-1 Cycle 13 Assembly Average Power Distributions

D

Figure 3-5 McGuire Unit-1 Cycle 14 Assembly Average Power Distributions

D

Figure 3-6 McGuire Unit-2 Cycle 12 Assembly Average Power Distributions

D

Figure 3-7 McGuire Unit-2 Cycle 13 Assembly Average Power Distributions

D

Figure 3-8 McGuire Unit-2 Cycle 14 Assembly Average Power Distributions

D

Figure 3-9 Catawba Unit-1 Cycle 11 Assembly Average Power Distributions

D

Figure 3-10 Catawba Unit-1 Cycle 12 Assembly Average Power Distributions

D

Figure 3-11 Catawba Unit-1 Cycle 13 Assembly Average Power Distributions

D

D

Figure 3-12 Catawba Unit-2 Cycle 9 Assembly Average Power Distributions

D

Figure 3-13 Catawba Unit-2 Cycle 10 Assembly Average Power Distributions

D

Figure 3-14 Catawba Unit-2 Cycle 11 Assembly Average Power Distributions

D

Figure 3-15 Saint Laurent B1 Core Configuration

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1								46							
2						N2		R		45 N2					
3				26			36 SB	35	SB		SD	43			
4				N1	24	G2		38 G1		48 G2		N1			
5			15 SD		14 SC		25 SD				34 SC	44		42	
6		N2		G2	18	R		37 N1		31 R		G2		N2	
7			3 SB				13		23		SD	33	SB	41	
8	6	R	17	G1	16	N1		SA		39 N1		G1	49	47 R	
9			SB		20 SD				27	21			SB		50
10		N2	8	G2		R	19	N1		R		40 G2		32 N2	
11					1 SC			11	SD	29	28 SC		SD		
12			7	N1		G2	5	G1		G2		30 N1	22		
13					SD		SB	4	SB	12					
14					9	N2		R	2	N2					
15							10								

Incore Detector Location
Control Rod Bank Location

Figure 3-16 Saint Laurent B1 Hot Full Power Boron Deviations (ppm)

D

Figure 3-17 Saint Laurent B1 Cycle 5 Assembly Average Power Distributions

D, E

Figure 3-18 Saint Laurent B1 Cycle 6 Assembly Average Power Distributions

D, E

Figure 3-19 Saint Laurent B1 Cycle 7 Assembly Average Power Distributions

D, E

Figure 3-20 Saint Laurent B1 Cycle 8 Assembly Average Power Distributions

D, E

Figure 3-21 Saint Laurent B1 Cycle 9 Assembly Average Power Distributions

D, E

Figure 3-22 Saint Laurent B1 Cycle 10 Assembly Average Power Distributions

D, E

[Page intentionally left blank]

4.0 FUEL PIN POWER DISTRIBUTION BENCHMARK ANALYSIS

CASMO-4 and its predecessor, CASMO-3, have been used worldwide to model both LEU and MOX fuel, including predictions of fuel pin power distributions, for more than ten years. Reference 21 describes the results of benchmarking CASMO-4 against the B&W PWR and KRITZ PWR and boiling water reactor (BWR) critical experiments. Reference 22 documents results of benchmarks of CASMO-4/SIMULATE-3 against the VENUS International Program (VIP) PWR MOX fuel critical experiments. Reference 23 presents the results of benchmarking CASMO-4 against MCNP for several types of MOX fuel assemblies, including a fuel assembly design similar to the design proposed for use by Duke.

This section describes the development of pin power distribution uncertainty factors using CASMO-4/SIMULATE-3 MOX. Two separate pin power uncertainty factors were developed; one applicable to LEU fuel, and one applicable to MOX fuel.

4.1 Methodology

The pin power uncertainty factors were developed based on comparisons of predicted and measured pin power distributions from several critical experiments for both LEU and MOX fuel. Table 4-1 provides a tabulation of the fuel parameters for each of the critical experiments along with the parameters of the proposed Duke MOX fuel for comparison. Separate uncertainty factors were calculated for LEU fuel and for MOX fuel. These pin power uncertainties were calculated using 95/95 statistical methodology consistent with that described in Section 3.1.5.

For LEU fuel, the pin power uncertainty was calculated using data from the B&W critical experiments (Reference 24). The pin power uncertainty calculation for LEU fuel used the same methods that were used in Reference 2 except that CASMO-4 and SIMULATE-3 MOX were used in lieu of CASMO-3 and SIMULATE-3. The pin power uncertainty was determined by direct comparison of the SIMULATE-3 MOX calculations with the measured data from the critical experiments. The derivation of the LEU fuel pin power uncertainty factor is described in Section 4.2.

For MOX fuel, the pin power uncertainty was calculated using data from the Saxton, EPICURE, and ERASME/L series of critical experiments. These critical experiments were much smaller than the B&W experiments and as a result could not be adequately modeled with SIMULATE-3 MOX. Therefore, the pin power uncertainty factor for MOX fuel was calculated in two steps. First, each of the critical experiments was modeled using CASMO-4, and pin power uncertainty factors were calculated. The derivation of these uncertainty factors is presented in Section 4.3.

Second, a set of theoretical infinite lattice cases (colorsets) was modeled using both CASMO-4 and SIMULATE-3 MOX in order to determine the SIMULATE-3 MOX pin uncertainty with respect to CASMO-4 predictions. The uncertainty factor derived from the colorset calculations was combined with the uncertainty factor from the CASMO-4 calculations of the MOX fuel critical experiments to obtain the overall MOX fuel pin power uncertainty factor. This work is described in Section 4.4.

4.2 Low Enriched Uranium Fuel Critical Experiments

Pin power distribution comparisons between measured and calculated values were performed for LEU fuel pins using the Babcock & Wilcox (B&W) Urania Gadolinia critical experiments that are documented in Reference 24. These experiments are the same as those evaluated in Reference 2 and which formed the basis for the fuel pin power uncertainty that is currently applied to Duke Power LEU core designs. A series of three B&W critical experiments were evaluated to assess the capability of the CASMO-4/SIMULATE-3 MOX programs to accurately calculate pin power distributions. The B&W critical experiment configurations that were evaluated are as follows:

A. Core 1 - Consists entirely of 2.46 weight percent (w/o) LEU fuel pins arrayed to simulate a 15x15 B&W type fuel assembly lattice.

B. Core 12 - Identical to Case 1 except that the central 31x31 region contains 4.02% LEU fuel pins with the rest consisting of 2.46% LEU fuel pins.

C. Core 18 - Consists of a central 32x32 region of 4.02% LEU fuel pins with peripheral 2.46% LEU fuel. This experiment is arranged to simulate a Combustion Engineering 16x16 fuel assembly with 2x2 water holes.

Table 4-2 shows the critical conditions for these experiments. Measured pin powers were obtained from Reference 24 for Cores 1, 12, and 18. Figure 4-1 illustrates the layout of the Core 1 experiment. The other experiments are similar, except for the differences described above. The power distribution was measured by counting the fission product gamma radiation produced from each fuel pin following irradiation. Each fuel pin was measured three times and the results averaged, and then normalized to an average relative power of 1.0. All of the measurements were performed at the point on the fuel pin corresponding to the midplane of the experiment.

The experiments were modeled using two different methods. Method 1 involved developing CASMO-4 models of the critical configurations and using these models to calculate the fuel pin power distributions directly. Small deviations from the as-built configurations were necessary in order to execute the CASMO-4 model. Specifically, the code input required a square geometry, so inputs defining the water peripheral to the experiments were adjusted to meet this requirement. These CASMO-4 models of the B&W critical experiments were developed for comparison purposes only.

In Method 2, separate CASMO-4 models were used to create cross sections and assembly discontinuity factors for three-dimensional SIMULATE-3 MOX models of the critical configurations. Method 2 is essentially the same approach that was used in Reference 2. As in Reference 2, a small number of peripheral fuel pins were relocated to provide a better model of partial fuel assemblies at the exterior of the experiment. This addressed the fact that SIMULATE-3 MOX was not designed to model partial fuel assemblies with very few fuel pins. Also, SIMULATE-3 MOX requires that reflectors have no fuel and at least a trace amount of moderator. All of these changes were restricted to the periphery of the core.

The two methods provided consistent predictions of fuel pin power distributions for the three configurations that were analyzed. Comparisons between the measured pin power distributions and the predicted pin power distributions for Cores 1, 12, & 18 are presented for the two methods in Figures 4-2, 4-3, and 4-4, respectively. The measured and predicted data in these figures has been normalized to an average value of 1.0 in order to provide a consistent comparison. The results of the statistical evaluation of the calculated vs. measured power distributions for both methods are shown in Table 4-3. From these results, the conclusion is that the CASMO-4/SIMULATE-3 MOX code system is shown to accurately predict the pin power distribution for individual pins within a LEU fuel assembly. The 95/95 pin power uncertainty ($K\sigma$) for LEU fuel modeled with SIMULATE-3 MOX (Method 2) is [ ]$_D$, which is only slightly higher than the CASMO-4 model uncertainty (Method 1) for these experiments.

4.3 Mixed Oxide Fuel Critical Experiments

For MOX fuel, most of the available recent critical experiment data consists of reactor grade (RG) MOX fuel experiments performed in France, Belgium, and Sweden. In addition, the older Westinghouse Saxton experiments (References 25 and 26) used near weapons grade MOX fuel (approximately 90% Pu^{239}) that was very comparable to the isotopic composition of the proposed Duke MOX fuel.

Pin power distribution comparisons between measured and calculated values were made for MOX fuel pins using three sets of available critical experiments. These experiments were:

A. Westinghouse Saxton (References 25 and 26)

B. French Commissariat à l'Énergie Atomique (CEA) EPICURE (References 27 through 32)

C. French CEA ERASME/L (References 33 and 34)

Unlike the B&W experiments described in Section 4.2, most of these experiments were too small to model accurately with SIMULATE-3 MOX. The largest of these was only about half the size of the B&W critical experiments. Therefore, Duke Power modeled the MOX fuel critical experiments using CASMO-4 only. Section 4.4 addresses the ability of SIMULATE-3 MOX to replicate CASMO-4 calculations.

4.3.1 Saxton Critical Experiments

The Saxton critical experiments are described in References 25 and 26. These experiments consisted of a series of single region UO_2-PuO_2 and multi-region UO_2-PuO_2/UO_2 fueled geometries with several different pin pitches. The results were evaluated at the Critical Reactor Experiment facility at the Westinghouse Reactor Evaluation Center in Saxton, Pennsylvania in 1965. These experiments are the smallest and oldest of the MOX fuel critical experiments evaluated in this topical report. These experiments are useful because they involve MOX fuel with a 90% Pu^{239} isotopic assay which approaches the isotopic composition of weapons grade (WG) plutonium.

The Saxton experiments used MOX fuel pins of 6.6% PuO_2 in natural UO_2. Multi-region geometries also included 5.74% U^{235} LEU fuel. In addition to variations in pin layouts, these experiments evaluated perturbations in the lattice structure that included water slots, aluminum slab spacers, and silver-indium-cadmium (AIC) rods in the lattice. These perturbations were introduced by removing five central pins in the single region experiments and five pins at the interface in multi-region experiments. The relative powers of the fuel pins were determined by measurement of gamma activity normalized to a reference fuel pin. Measurements of irradiated foil activity as well as thermal measurements were used to verify the gamma activity measurements.

The Saxton critical experiment cases from Reference 26 that were evaluated are as follows:

A. Case 2 - 19x19 MOX fuel pin array

B. Case 3 - 19x19 MOX fuel pin array, with 5x1 water slot in center

C. Case 4 - 19x19 MOX fuel pin array, with 5x1 aluminum plate in center

D. Case 5 - 21x21 MOX fuel pin array, 5x1 AIC rods in center

E. Case 21 - 19x19 LEU fuel pin array with 11x11 MOX interior pin array

F. Case 22 - 19x19 LEU fuel pin array with 11x11 MOX interior pin array, 5x1 aluminum plate at interface

G. Case 24 - 27x27 LEU fuel pin array with 19x19 MOX interior pin array

H. Case 25 - 27x27 LEU fuel pin array with 19x19 MOX interior pin array, 5x1 aluminum plate at interface

I. Case 26 - 27x27 LEU fuel pin array with 19x19 interior MOX fuel pin array, 5x1 water slot at interface

J. Case 27 - 27x27 LEU fuel pin array with 19x19 interior MOX fuel pin array, and L-shaped (nine pins) LEU inserts in MOX fuel region, also two flux wire rods in MOX lattice and one flux wire rod in LEU lattice.

K. Case 28 - 27x27 LEU fuel pin array with 19x19 interior MOX fuel pin array, 3x3 LEU insert in center of MOX fuel region

L. Case 30 - 19x19 LEU fuel pin array with 3x3 MOX insert

All of these experiments had a pin pitch of 0.56 inches (1.42 cm). Figures 4-5 through 4-16 illustrate the geometries and show the results of the CASMO-4 calculations for the Saxton experiments. The measured and predicted data in these figures have been normalized to an average value of 1.0 in order to provide a consistent comparison. Table 4-4 contains a summary of the CASMO-4 results for MOX fuel pins in each of the Saxton configurations that were evaluated by Duke. For comparison, the results of MCNP calculations from Reference 26 are also included in Table 4-4.

The 95/95 pin power uncertainty derived from the Saxton experiments modeled with CASMO-4 is [ ]D. Since the Saxton experiments did not pass the D' test, the 95/95 uncertainty is defined by non-parametric methods as the [ ]D most negative result, in this case [ ]D. Had the results passed the D' test the 95/95 uncertainty using Kσ would have been [ ]D.

4.3.2 EPICURE Critical Experiments

The EPICURE experiments were performed at a facility in Cadarache, France between 1987 and 1994. Figure 4-17 shows the typical EPICURE critical experiment layout, in this case for the UMZONE configuration. The EPICURE experiments consist of cylindrical arrays of fuel pins with a diameter of between 43 and 55 pin pitches. The MOX fuel pins were centrally located and surrounded by a buffer region of LEU fuel pins. The EPICURE experiments were designed to be representative of plutonium recycling in PWRs. The objectives of these experiments were (i) to obtain accurate measurements of individual fuel pin flux distributions in fuel assembly configurations that are identical to those in PWRs and (ii) to obtain accurate reactivity measurements for various absorbers and partial/total void fractions. Several of the EPICURE experiments used fuel pin layouts comparable to the proposed Duke MOX fuel assembly arrangement.

The fuel pins used in the EPICURE experiments are similar to fuel pins used in 17x17 reactor fuel. They are 0.950 cm in outside diameter (OD) with a pin pitch of 1.26 cm., and have similar fuel cladding and pellet dimensions as production 17x17 fuel pins. In addition, the EPICURE experiments used aluminum overcladding on each fuel pin to displace moderator in the lattice in order to approximate the hot condition fuel to moderator ratio. Four different fuel pin designs were used in the EPICURE experiments, differing only in fuel pellet composition. These consisted of 3.70% LEU fuel pins and MOX fuel pins containing 4.3%, 7.0%, and 8.7% plutonium in a depleted UO_2 matrix. These experiments used RG MOX fuel, which has a more complex plutonium isotopic mix than the WG MOX fuel that will ultimately be used by Duke Power. Table 4-5 shows the plutonium and americium isotopic composition for the MOX fuel pins used in the EPICURE experiments.

The relative powers of the fuel rods were determined by measurement of gamma activity. These included both total gamma activity as well as measurements of the La^{140} and Sr^{92} gamma peaks. Comparisons were performed at octant symmetry. Where measurements of symmetric pins existed, the average of these measurements was used. Finally, both measured and calculated fission rates for each experiment were normalized to an average value of 1.0 for the pins evaluated in order to provide for a consistent comparison.

The EPICURE critical experiment configurations that were evaluated are as follows:

A. UMZONE (Reference 27) is a test of a typical reactor fuel pin layout. The central zone contains a 17x17 MOX fueled PWR pin arrangement with three plutonium concentrations, 4.3%, 7%, and 8.7%. The 17x17 region has 24 guide tubes and a central instrument tube. This configuration is surrounded by 3.7% LEU fuel pins that also approximate the typical 17x17 fuel geometry.
B. UMZONE B_4C (Reference 28) is similar to the UMZONE experiment with 24 B_4C control rods added to the 17x17 MOX fuel region in the 24 guide tubes.
C. UMZONE AIC (Reference 29) is similar to the UMZONE experiment with 24 AIC control rods added to the central 17x17 MOX fuel region in the 24 guide tubes.
D. MH1.2-93 (Reference 30) is an experiment with an homogeneous cylindrical MOX fuel region surrounded by a buffer region of LEU fuel pins. The MOX fuel pins in this experiment were 7% plutonium with an UO_2 enrichment of approximately 0.24% U^{235}.
E. UM 17x17/7% (Reference 31) is a 17x17 homogeneous MOX experiment with fuel pins containing 7% plutonium surrounded by a LEU fuel pin buffer region.
F. UM 17x17/11% (Reference 32) is a 17x17 homogeneous MOX experiment with fuel pins containing 11% plutonium surrounded by a LEU fuel pin buffer region. The 11% pins came from the ERASME/L experiments described in Section 4.3.3.

Figures 4-18 through 4-23 show the results of the CASMO-4 calculations for the MOX fuel pins in the EPICURE experiments. A summary of the uncertainty calculations is shown in Table 4-6. The 95/95 pin power uncertainty for the EPICURE experiments modeled with CASMO-4 is [ ]D.

4.3.3 ERASME/L Critical Experiments

The ERASME/L experiments (References 33 and 34) have a slightly smaller pin pitch (1.19-cm) than 17x17 PWR fuel and used RG MOX fuel pins with 11% plutonium. These experiments are nonetheless considered valuable because their fissile plutonium concentration of 8.28% bounds that which Duke Power expects to use. The ERASME/L experiments were cylindrical, 45 pin pitches across, and were similar in setup to the EPICURE experiments. They consisted entirely of MOX fuel pins, except for guide tube locations that accommodate control rod and burnable poison (BP) rod locations. The configurations analyzed included nine B_4C rods with three different spacings and a configuration with one B_4C rod. The pin powers were measured by gamma scans. Table 4-5 shows the plutonium and americium isotopic composition for the MOX fuel pins used in the ERASME/L experiments.

Figure 4-24 illustrates the layout of the ERASME/L experiment, Case D. The other experiments are similar. The ERASME/L critical experiment configurations that were evaluated are as follows:

A. Central region containing one B_4C poison rod.

B. Central region containing a square array of nine B_4C poison rods spaced at every other rod location.

C. Central region containing a square array of nine B_4C poison rods spaced at every third rod location.

D. Central region containing a square array of nine B_4C poison rods spaced at every fourth rod location.

Figures 4-25 through 4-28 show the results of the CASMO-4 calculations for the MOX fuel pins in the ERASME/L experiments. A summary of the uncertainty calculation results for the ERASME/L experiments is shown in Table 4-6. The 95/95 pin power uncertainty for the ERASME/L experiments modeled with CASMO-4 is [ ]D.

4.4 Theoretical Benchmark of SIMULATE-3 MOX to CASMO-4

As noted in Section 4.3, the MOX fuel critical experiments were too small to model accurately with SIMULATE-3 MOX. Accordingly, fuel pin power distribution uncertainties were based on CASMO-4 calculations. Because the calculated uncertainty factors described in Section 4.3 only apply to CASMO-4, it was necessary to assess and account for the ability of SIMULATE-3 MOX to reproduce the pin power distributions from CASMO-4. CASMO-4 output data was processed by CMS-LINK to create an input library for use by SIMULATE-3 MOX.

4.4.1 Description of Benchmarks

A series of five theoretical benchmarks (colorsets) were evaluated in order to assess the ability of SIMULATE-3 MOX to replicate CASMO-4 pin power calculations. These theoretical benchmarks consisted of 2x2 assembly infinite lattice calculations executed with each code. The benchmark problems were designed to approximate the various combinations of feed and reload fuel assembly loading patterns that are expected in a typical reload core over a burnup range typical of MOX fuel assemblies during a full cycle of operation. These combinations included (i) a feed MOX fuel assembly in checkerboard and (ii) face-adjacent feed/reinsert layouts that included other MOX and LEU fuel assemblies. The MOX fuel assemblies consisted of layouts with no burnable poisons and layouts heavy with burnable poison. Figure 4-29 shows the configurations evaluated for these theoretical benchmarks.

The results from the evaluation of these colorsets validate the ability of SIMULATE-3 MOX to adequately replicate the CASMO-4 calculations. Individual CASMO-4 executions for each of the fuel assembly types in the colorsets were run for input to the SIMULATE-3 MOX model. A separate CASMO-4 model of each of the colorsets was then run for comparison with the SIMULATE-3 MOX model of the colorsets. Each of the colorsets was depleted to a burnup of 20 GWd/Mthm. The reinserted assemblies in each colorset had calculated burnups in excess of 40 GWd/Mthm. Comparisons of individual pin power for each MOX fuel assembly were made at the beginning, middle, and end burnup of the execution for each fuel pin.

Quantitative results of the theoretical benchmark comparisons for each of the five colorsets evaluated are shown in Table 4-7. The table shows the standard deviation of the fuel pin power calculations for each MOX fuel pin in the five theoretical benchmark cases. The results are consistent with Reference 12, which documented the ability of SIMULATE-3 MOX to replicate pin powers from CASMO-4 with MOX and LEU assemblies heterogeneously placed in both infinite lattice and quarter core calculations to a root mean square error of 1% or less.

4.4.2 Statistical Evaluation of Benchmark Data

The statistical methods employed were based on those described in Reference 2. The data was evaluated using a 95/95 statistical method to develop a SIMULATE-3 MOX to CASMO-4 uncertainty factor. The distribution of individual pin comparisons was tested for normality according to Reference 18 using a 1% level of significance as described in Reference 16. Where the test for normality was acceptable, the uncertainty factor was defined as follows:

$$\text{Pin uncertainty factor} = 1 - \text{bias} + K_p\sigma_p$$

where:

K_p = 95/95 one-sided factor based on sample size from Reference 17

σ_p = standard deviation of the population of individual pin comparisons.

Since both the measured and calculated data from the CASMO-4 and SIMULATE-3 MOX comparisons to critical experiment data were both normalized to 1.0 for each critical experiment, the bias term is zero.

Where the test for normality was unacceptable, the non-parametric evaluation described in Reference 16 was used for the 95/95 one-sided tolerance as described in Reference 15. The 95/95 uncertainty of a distribution is the m^{th} worst comparison where m is a function of the number of comparisons. Values of m were derived from References 15 and 16. The calculated value for the 95/95 uncertainty for the SIMULATE-3 MOX theoretical comparison with CASMO-4 is [ ]D as shown in Table 4-9.

4.5 Fuel Pin Power Distribution Uncertainty Factors

4.5.1 LEU Fuel Pin Uncertainty

The pin power uncertainty factor for LEU fuel was calculated using the results from the B&W critical experiment benchmarks that are described in Section 4.2. Table 4-3 contains a summary of the statistical analysis of the LEU fuel critical experiment benchmarks. The 95/95 pin power uncertainty factor was developed using the data from Cases 1, 12, and 18 of the B&W critical experiments for both SIMULATE-3 MOX and CASMO-4 (Method 2 from Section 4.2). The normality of the data was confirmed using the D' normality test. The calculated pin power uncertainty is [ ]D for SIMULATE-3 MOX. This uncertainty compares well with the direct CASMO-4 calculations (Method 1 from Section 4.2) that are also presented in Table 4-3.

4.5.2 MOX Fuel Pin Uncertainty

The uncertainty calculation for CASMO-4 modeling of MOX fuel pins in the three sets of critical experiments is shown in Table 4-8. The population satisfies the D' normality test. As shown in Table 4-8, the CASMO-4 uncertainty for MOX fuel pins is [ ]D. The benchmark of SIMULATE-3 MOX against CASMO-4 calculations is described in Section 4.4. The statistical calculation of the uncertainty for SIMULATE-3 MOX benchmarked against CASMO-4 is [ ]D as shown in Table 4-9. The pin power uncertainty for SIMULATE-3 MOX is determined by the statistical combination of these two contributing uncertainties as follows:

Pin uncertainty = [ ]D

This uncertainty is applied in Section 5 to calculate the uncertainty factors on total peaking and radial peaking.

Table 4-1 Mission Reactor and Critical Experiment Fuel Parameters

Fuel Parameter	MNS/CNS MOX Fuel*	EPICURE	ERASME/L	Saxton	B&W
Material and Type	Ceramic MOX & LEU fuel pellets	Ceramic MOX & LEU fuel pellets	Ceramic MOX Fuel Pellets	Ceramic MOX & LEU Fuel Pellets	Ceramic LEU Fuel Pellets
Pu Isotopic %Fissile (Pu^{239} & Pu^{241})	92% to 93%	67%	76%	91%	N/A
Theoretical Density (g/cm^3)	10.99	11.00	11.01	10.99	10.96
Actual Density, (g/cm^3)	10.37	10.37	10.496	10.19	9.46 & 10.24
% of Maximum Theoretical Density	94.4%	94.3%	95.3%	92.7%	86.3% & 93.4%
Initial Plutonium Concentration	4.9%, 3.4%, & 2.4%	8.7%, 7%, & 4.3%	10.89%	6.6%	N/A
Fissile Plutonium Concentration	4.6%, 3.1%, & 2.2%	5.9%, 4.8%, & 2.9%	8.28%	6.0%	N/A
Plutonium Zoning	Three Zone Enriched	Three Zone Enriched & Single Enrichment	Single Enrichment	Single Enrichment, some with LEU Zones	N/A
Lattice Pitch	1.265 cm	1.26 cm	1.19 cm	1.42 cm	1.64 cm
Fuel Pin OD	0.950 cm	0.950 cm (Zirc. Clad) 1.08 cm (Al overclad)	0.804 cm & 0.848 cm (pin double clad with SS)	0.993 cm	1.21 cm
Pellet OD	0.819 cm	0.819 cm	0.714 cm	0.857 cm	1.03 cm & 1.13 cm
Effective Fuel/Water Volume Ratio	0.591	0.557	0.334	0.330	0.422
Critical Soluble Boron Concentration	0 to 2500 ppm	50 to 579 ppm	750 to 1300 ppm	0 to 1453 ppm	1300 to 1900 ppm
Poison Rods in Lattice	AIC & B_4C Control Rods; B_4C BP Rods	AIC & B_4C Control Rods	B_4C Control Rods	AIC Rods	None
Moderator Temperature	557°F to 625°F	71°F to 78°F	79°F	59°F to 70°F	77°F

* Based on currently planned MOX fuel and core designs as summarized in Appendix A

Table 4-2 B&W Experiment Configurations and Critical Conditions

Case	Configuration	Water Temperature (Degrees F)	Boron Concentration (ppm)
1	15X15 2.46% LEU	77	1338
12	15X15 4.02% LEU 2.46% LEU peripheral	77	1899
18	16X16 4.02% LEU 2.46% LEU peripheral	77	1777

Table 4-3 Uncertainty Calculation Summary for B&W Critical Experiments

Standard Deviation Summary

Case	CASMO-4 (Method 1)	SIMULATE (Method 2)
1	[ ]D	[ ]D
12	[ ]D	[ ]D
18	[ ]D	[ ]D

D' Test Results

Parameter	CASMO-4 (Method 1)	SIMULATE (Method 2)
n	[ ]D	[ ]D
D' (P=0.005)	[ ]D	[ ]D
D'	[ ]D	[ ]D
D'(P=.995)	[ ]D	[ ]D
Evaluation	[ ]D	[ ]D

Uncertainty Calculation Results

Parameter	CASMO-4 (Method 1)	SIMULATE (Method 2)
Std. Dev.	[ ]D	[ ]D
K	[ ]D	[ ]D
Uncertainty	[ ]D	[ ]D

Table 4-4 Uncertainty Calculation Summary for Saxton Critical Experiments

Standard Deviation Summary

Experiment/ Case No.	No. of data points	CASMO-4 MOX Std. Dev.	MCNP MOX Std. Dev. (Reference 26)
2	13	[ ]D	1.13%
3	13	[ ]D	1.26%
4	15	[ ]D	1.22%
5	12	[ ]D	1.50%
21	14	[ ]D	0.96%
22	14	[ ]D	1.33%
24	16	[ ]D	1.42%
25	15	[ ]D	1.13%
26	15	[ ]D	1.48%
27	18	[ ]D	3.08%
28	8	[ ]D	0.87%
30	3	[ ]D	0.87%

CASMO-4 D' Test Results

Parameter	Value
n	[ ]D
D' (P=0.005)	[ ]D
D'	[ ]D
D'(P=.995)	[ ]D
Evaluation	[ ]D

CASMO-4 Uncertainty Calculation

Parameter	Value
Std. Dev.	[ ]D
K	[ ]D
n	[ ]D
m	[ ]D
Uncertainty (Non-parametric)	[ ]D

Table 4-5 Isotopic Assay of MOX Fuel Pins in Saxton, EPICURE, and ERASME/L Critical Experiments

Total Plutonium Concentration	Pu-238 (w/o)	Pu-239 (w/o)	Pu-240 (w/o)	Pu-241* (w/o)	Pu-242 (w/o)	Am-241* (w/o)
Saxton						
6.6%	0%	90.49%	8.57%	0.89%	0.04%	0%
EPICURE						
4.3%	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F
7.0%	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F
8.7%	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F
11%	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F	[[ ]F
ERASME/L						
11%	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F	[ ]F

Note: The ERASME/L fuel pins were also used in the EPICURE UH 17x17/11% experiment.

* Isotopic composition was adjusted in CASMO-4 to account for decay time from initial assay.

Table 4-6 Uncertainty Calculation Summary for EPICURE and ERASME/L Critical Experiments

Standard Deviation Summary

Experiment	No. of Data Points	CASMO-4 MOX Std. Dev.
EPICURE - UMZONE	[ ]F	[ ]D
EPICURE - UMZONE-B_4C	[ ]F	[ ]D
EPICURE - UMZONE-AIC	[ ]F	[ ]D
EPICURE - MH 1.2-93	[ ]F	[ ]D
EPICURE - UH 17x17 / 7%	[ ]F	[ ]D
EPICURE - UH 17x17 / 11%	[ ]F	[ ]D
ERASME/L - 1 B_4C Rod	[ ]F	[ ]D
ERASME/L - 9 B_4C Rods- small spacing	[ ]F	[ ]D
ERASME/L - 9 B_4C Rods- medium spacing	[ ]F	[ ]D
ERASME/L - 9 B_4C Rods- large spacing	[ ]F	[ ]D

D' Test Results

Parameter	EPICURE	ERASME/L
n	[ ]F	[ ]F
D' (P=0.005)	[ ]D	[ ]D
D'	[ ]D	[ ]D
D' (P=.995)	[ ]D	[ ]D
Evaluation	[ ]D	[ ]D

Uncertainty Calculation

Parameter	EPICURE	ERASME/L
Std.Dev.	[ ]D	[ ]D
K	[ ]D	[ ]D
Uncertainty	[ ]D	[ ]D

Table 4-7 SIMULATE-3 MOX to CASMO-4 Colorset Comparisons

Case	Burnup GWd/Mthm	No. of Burnable Poison Rods	Std. Dev.
1a	25	None	[ ]D
	34.2	None	[ ]D
	43.4	None	[ ]D
1b	15	24 pulled	[ ]D
	25.2	24 pulled	[ ]D
	35.4	24 pulled	[ ]D
1c	0	24	[ ]D
	9.2	24	[ ]D
	18.9	24	[ ]D
2a	0	24	[ ]D
	9.8	24	[ ]D
	19.9	24	[ ]D
2b	0	None	[ ]D
	11.7	None	[ ]D
	23.1	None	[ ]D
3a	20	None	[ ]D
	30	None	[ ]D
	40.1	None	[ ]D
3b	0	None	[ ]D
	12.1	None	[ ]D
	23.9	None	[ ]D
4a	25	24 pulled	[ ]D
	34.4	24 pulled	[ ]D
	43.9	24 pulled	[ ]D
4b	15	None	[ ]D
	24.5	None	[ ]D
	34.2	None	[ ]D
4c	0	24	[ ]D
	9.7	24	[ ]D
	19.7	24	[ ]D
5a	0	None	[ ]D
	11.3	None	[ ]D
	22.3	None	[ ]D
5b	0	24	[ ]D
	9.3	24	[ ]D
	19	24	[ ]D

Table 4-8 Combined Uncertainty Calculation for Saxton, EPICURE, and ERASME/L Critical Experiments

D' Test Results

Parameter	Value
n	[ ]F
D' (P=0.005)	[ ]D
D'	[ ]D
D' (P=.995)	[ ]D
Evaluation	[ ]D

Uncertainty Calculation

n	Normal Distribution (Y/N)	K	Std. Dev.	Uncertainty (K x Std.Dev.)
[ ]F	[ ]D	[ ]D	[ ]D	[ ]D

Table 4-9 Statistics for SIMULATE-3 MOX Benchmark to CASMO-4

Theoretical Benchmark Normality Test Results

Cases	n	D' (P=0.005)	D' (calc.)	D' (P=0.995)	Evaluation
ALL	[ ]D	[ ]D	[ ]D	[ ]D	[ ]D

Uncertainty Calculation for Theoretical Benchmark

Cases	n	Normal Distribution (Y/N)	K	Std. Dev.	Normal Distribution Uncertainty (K x Std. Dev.)	Non-Parametric Uncertainty
All	[ ]D	[ ]D	[ ]D	[ ]D	N/A	[ ]D

For n = [ ]D , m = [ ]D for non-parametric 95/95 with no assumption of normality required. The error of the [ ]D worst comparison was [ ]D. The determination of normality for these distributions was based on the results of the D' test at a 1% level of significance as described in Section 4.4.2

Figure 4-1 B&W Critical Experiments – Core 1 General Layout

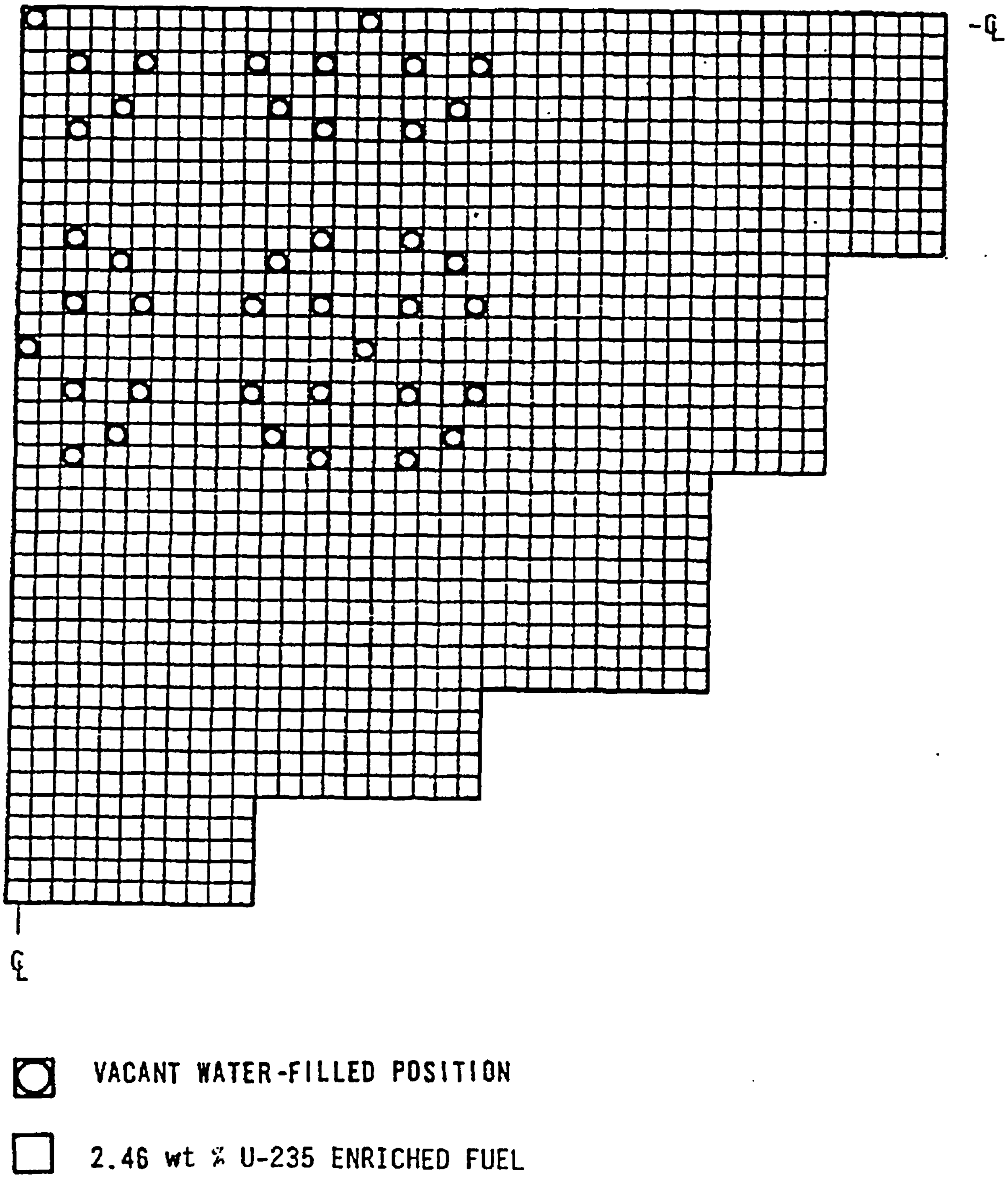


Figure 4-2 B&W Critical Experiments – Measured and Calculated Pin Power Distributions (Core 1)

D

Figure 4-3 B&W Critical Experiments – Measured and Calculated Pin Power Distributions (Core 12)

D

Figure 4-4 B&W Critical Experiments – Measured and Calculated Pin Power Distributions (Core 18)

D

Figure 4-5 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 2)

D

Figure 4-6 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 3)

D

Figure 4-7 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 4)

D

Figure 4-8 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 5)

D

Figure 4-9 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 21)

D

Figure 4-10 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 22)

D

Figure 4-11 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 24)

D

Figure 4-12 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 25)

D

Figure 4-13 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 26)

D

Figure 4-14 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 27)

D

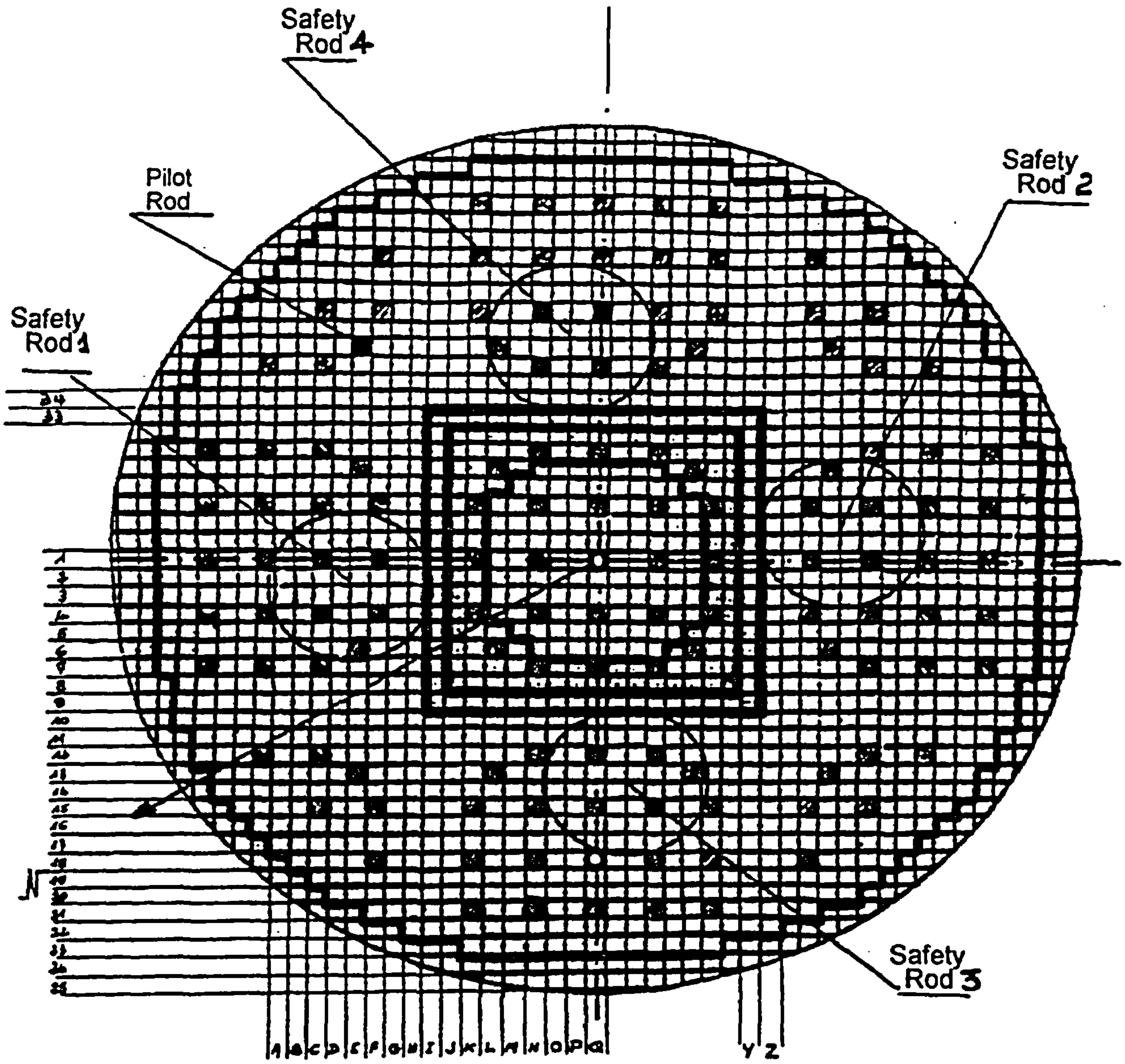
Figure 4-15 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 28)

D

Figure 4-16 Saxton Critical Experiments – Measured and Calculated Pin Power Distributions (Case 30)

D

Figure 4-17 EPICURE Critical Experiments – General Layout



Number of pins - UO_2	:	1264
Number of pins - UO_2-PuO_2	:	264
Number of guide tubes (including inst. tubes)	:	129
	Total :	1657

Pu% of the 264 UO_2-PuO_2 pins
100 @ 8.7 %
100 @ 7 %
64 @ 4.3 %

Figure 4-18 EPICURE Critical Experiments – Measured and Calculated Pin Power Distributions (UMZONE No BP)

No BP Rods

D, F

Figure 4-19 EPICURE Critical Experiments – Measured and Calculated Pin Power Distributions (UMZONE B_4C)

24 B_4C Rods in MOX Region

D, F

Figure 4-20 EPICURE Critical Experiments – Measured and Calculated Pin Power Distributions (UMZONE AIC)

24 AIC Rods in MOX Region

D, F

Figure 4-21 EPICURE Critical Experiments – Measured and Calculated Pin Power Distributions (MH1.2-93)

Central MOX Region with LEU Buffer

D, F

Figure 4-22 EPICURE Critical Experiments – Measured and Calculated Rod Power Distributions (UM 17x17/7%)

Central MOX Region with LEU Buffer

D, F

Figure 4-23 EPICURE Critical Experiments – Measured and Calculated Pin Power Distributions (UM 17x17/11%)

Central MOX Region with LEU Buffer

D, F

Figure 4-24 ERASME/L Critical Experiments – General Layout

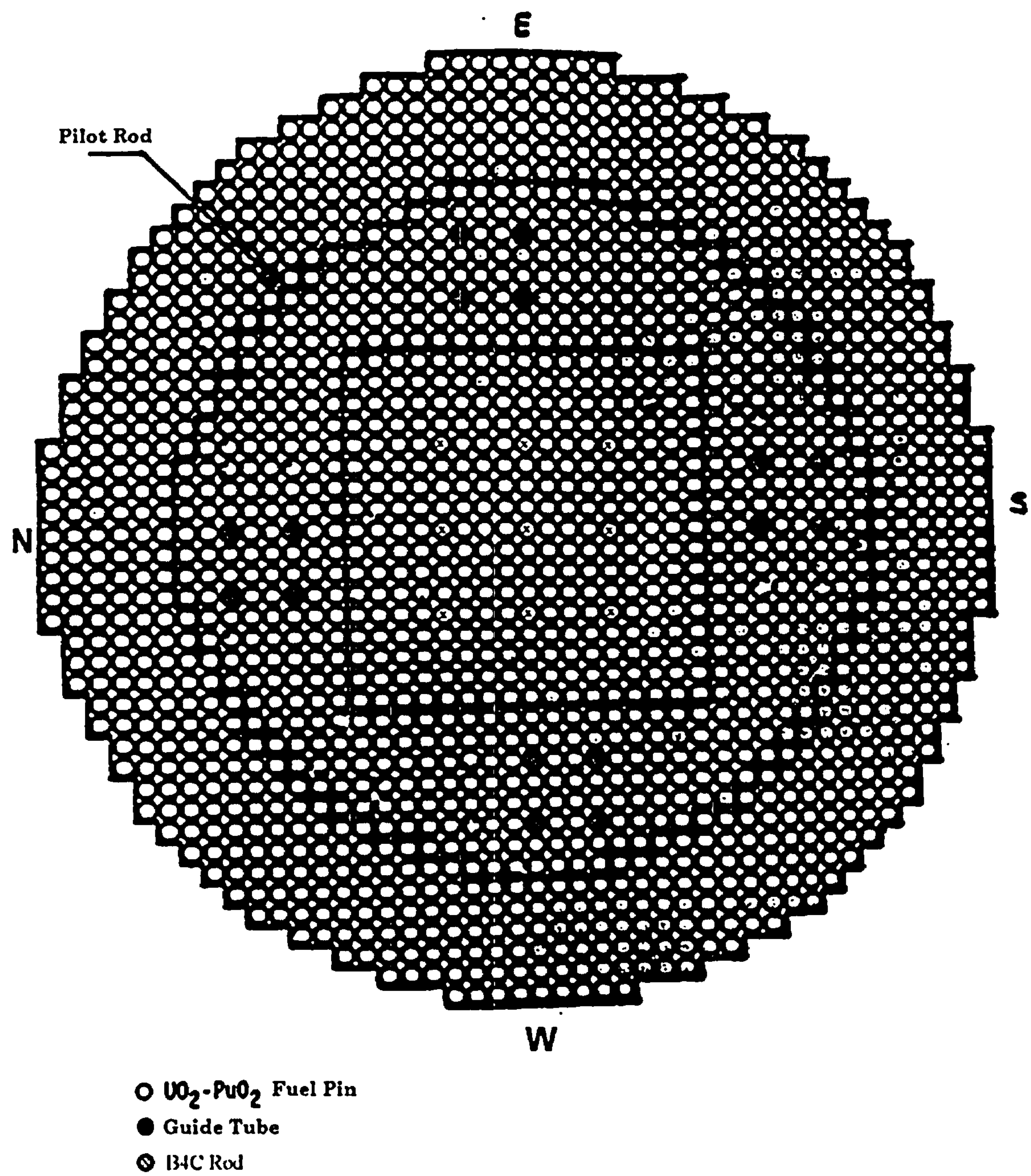


Figure 4-25 ERASME/L Critical Experiments – Measured and Calculated Power Distributions (One B_4C Rod)

D, F

Figure 4-26 ERASME/L Critical Experiments – Measured and Calculated Power Distributions (Nine B_4C Rods-Close Spacing)

D, F

Figure 4-27 ERASME/L Critical Experiments – Measured and Calculated Power Distributions (Nine B_4C Rods-Medium Spacing)

D, F

Figure 4-28 ERASME/L Critical Experiments – Measured and Calculated Power Distributions (Nine B_4C Rods-Large Spacing)

D, F

Figure 4-29 Theoretical Model Infinite Lattice (Colorset) Configurations

Case 1: Checkerboard MOX/LEU feed with two MOX reinserts

b* 4.37% MOX 24 BP rods pulled 15 GWd/Mthm	c 4.37% MOX 24 BP rods FEED
- 4.0% LEU No BP rods FEED	a 4.37% MOX No BP rods 25 GW/Mthm

Case 2: Face adjacent MOX feed with two LEU reinserts

- 4.0% LEU No BP rods 20 GWd/Mthm	b 4.37% MOX No BP rods FEED
- 4.0% LEU No BP rods 20 GWd/Mthm	a 4.37% MOX 24 BP rods FEED

Case 3: One MOX feed with one MOX & two LEU reinserts

- 4.0% LEU No BP rods 20 GWd/Mthm	b 4.37% MOX No BP rods FEED
- 4.0% LEU No BP 25 GWd/Mthm	a 4.37% MOX No BP rods 20 GWd/Mthm

Case 4: Face adjacent MOX/LEU feed, with two MOX reinserts

b 4.37% MOX No BP rods 15 GWd/Mthm	c 4.37% MOX 24 BP rods FEED
a 4.37% MOX 24 BP rods pulled 25 GWd/Mthm	- 4.0% LEU No BP rods FEED

Case 5: 2 MOX feed & 1 LEU feed (face adjacent) and 1 LEU reinsert

- 4.0% LEU No BP rods FEED	b 4.37% MOX 24 BPs rods FEED
a 4.37% MOX No BP rods FEED	- 4.0% LEU No BP rods 20 GWD/Mthm

*Note: Letters a, b, and c refer to case comparisons in Table 4-7.

[Page intentionally left blank]

5.0 MCGUIRE/CATAWBA STATISTICALLY COMBINED POWER DISTRIBUTION UNCERTAINTY FACTORS

5.1 General

Power distribution uncertainty factors are applied in both the design of reload cores and the surveillance of an operating fuel cycle. In each case the uncertainty factor is applied to power distribution peaking factors to insure a conservative comparison to thermal design limits on fuel pin performance. Because a direct measurement of individual pin power distribution is not available from power reactor operation, the complete uncertainty in the core model's ability to predict pin power distribution must be constructed from a synthesis of power reactor and critical experiment benchmark results. In its generic form, this synthesis can be expressed mathematically as follows:

$$\text{SCUF} = 1 - \frac{\sum_{i=1}^{n}\left(\frac{C_i - M_i}{M_i}\right)}{n} + \sqrt{\sum_{i=1}^{m}(K_i\sigma_i)^2}$$

where,

SCUF is the statistically combined uncertainty factor,

$\frac{\sum_{i=1}^{n}\left(\frac{C_i - M_i}{M_i}\right)}{n}$ is the bias, or average of n relative differences between calculated (C) and measured (M) values, and

$\sqrt{\sum_{i=1}^{m}(K_i\sigma_i)^2}$ is the combination by square root, sum of the squares of the individual 95/95 statistical deviations contributing to the total uncertainty factor.

For data sets that are shown to be normally distributed, $K\sigma$ is determined directly from the product of the one-sided upper tolerance limit K factor times the standard deviation, σ, of the data set. For data sets that do not pass a test for normality, $K\sigma$ is determined by the non-parametric evaluation described in Reference 16.

5.2 LEU Fuel Uncertainty Factor

For LEU fuel in McGuire and Catawba cores the SCUF equation is expressed as:

$$SCUF = 1 - bias + \sqrt{(K_a\sigma_a)^2 + (K_p\sigma_p)^2}$$

where $K_a\sigma_a$, represents the statistical deviation in the comparison between measured and calculated inter-assembly power distributions and $K_p\sigma_p$ is the equivalent term for intra-assembly pin power distribution deviation. The bias term and $K_a\sigma_a$ are derived from the McGuire/Catawba power distribution analyses results (Section 3.1.5) and $K_p\sigma_p$ is derived from SIMULATE-3 MOX modeling of the B&W critical experiments (Section 4.2).

5.3 MOX Fuel Uncertainty Factor

Benchmark results for St. Laurent B1 demonstrate that the CASMO-4/SIMULATE-3 MOX methodology produces statistical uncertainties on assembly power distribution that are similar for LEU and MOX fuel assemblies. Since the McGuire and Catawba reactors are fundamentally similar to St. Laurent B1, it is expected that the fidelity of fuel assembly power distribution predictions will also be similar for LEU and MOX fuel in McGuire and Catawba cores. Therefore, for MOX fuel in McGuire and Catawba cores, the bias and $K_a\sigma_a$ terms in the SCUF equation are derived from the McGuire/Catawba power distribution analysis.

The uncertainty on fuel pin power distribution calculations includes two components since the MOX fuel-bearing critical experiments could not be directly modeled with SIMULATE-3 MOX. The first component is derived from a comparison of CASMO-4 calculated pin power distributions to the Saxton, EPICURE, and ERASME/L critical experiments as described in Section 4.3. The second component is determined by comparing SIMULATE-3 MOX results to CASMO-4 pin power calculations for a set of theoretical problems that could be modeled with both codes as described in Section 4.4.

These two components are combined as described in Section 4.5.2 to obtain the fuel pin power uncertainty, $K_p\sigma_p$, for MOX fuel.

The calculated SCUFs for the CASMO-4/SIMULATE-3 MOX models of MNS/CNS LEU and MOX fuel assemblies are shown in Table 5-1.

Table 5-1 MOX and LEU Fuel Statistically Combined Uncertainty Factors

Parameter	Bias	Assembly Uncertainty ($K_a\sigma_a$)	Pin Uncertainty ($K_p\sigma_p$)	SCUF
LEU Fuel				
$F_{\Delta h}$	[ ]D	[ ]D	[ ]D	[ ]D
F_q	[ ]D	[ ]D	[ ]D	[ ]D
F_z	[ ]D	[ ]D	N/A	[ ]D
MOX Fuel				
$F_{\Delta h}$	[ ]D	[ ]D	[ ]D	[ ]D
F_q	[ ]D	[ ]D	[ ]D	[ ]D
F_z	[ ]D	[ ]D	N/A	[ ]D

6.0 DYNAMIC ROD WORTH MEASUREMENT

The reactivity worth of control rods is measured at the beginning of each fuel cycle. The purpose of the test is to compare the measured and predicted rod worths and confirm that the core is responding as expected. The Dynamic Rod Worth Measurement (DRWM) technique is a relatively new method for measuring the reactivity worth of individual control rod banks. It is accomplished by inserting and withdrawing an individual rod bank at maximum stepping speed without changing boron concentration. The excore detector signals are recorded during the rod insertion and then processed on a reactivity computer which solves the inverse point kinetics equation with proper analytical compensation for transient spatial effects.

6.1 Benchmark of CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX

The Westinghouse DRWM topical report, Reference 35, defines the DRWM technique used at McGuire and Catawba. Attachment 1 in Reference 35 establishes a set of criteria to be used by utilities that choose to perform their own calculations to support DRWM. Reference 3 addresses the DRWM technology transfer criteria for the CASMO-3/SIMULATE-3/SIMULATE-3K code package applied to McGuire and Catawba. This methodology was approved by the NRC and is currently used to support DRWM at McGuire and Catawba. Reference 3 documented an extensive benchmark of the Duke DRWM methodology by comparing against Westinghouse DRWM results for six separate startups.

This section documents results from CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX calculations for the same set of six DRWM tests included in Reference 3. Table 6-1 compares predicted rod worth results from the new core models to Westinghouse calculations. Table 6-2 makes a similar comparison for measured rod worths. Criterion 4 of the technology transfer criteria defined in Reference 35 states that the comparisons to Westinghouse calculated and measured results should agree within 2% or 25 pcm for individual banks, and 2% for total bank worth.

As shown in Tables 6-1 and 6-2, the 2% criterion on total bank worth was exceeded in only [ ]$_D$ of 12 comparisons. For Catawba 1 Cycle 11 the relative error between Duke and Westinghouse predictions of total bank worth was [ ]$_D$. The 25 pcm criterion on individual bank worth was exceeded in only [ ]$_D$ of 108 comparisons. The largest deviation was [ ]$_D$ pcm on predicted worth of control bank D for McGuire 2 Cycle 13. These results are slightly better than those produced by the CASMO-3/SIMULATE-3/ SIMULATE-3K code package where 6 of 108 individual bank worth deviations exceeded the 25 pcm criterion.

Westinghouse results are produced by the ALPHA/PHOENIX/ANC/SPNOVA core model package and are thus a completely independent methodology. Differences between Duke and Westinghouse methodologies are expected to produce different predicted and measured rod worths. The results in Tables 6-1 and 6-2 demonstrate exceptional consistency in the DRWM results for these two core modeling methodologies. These results also show that the CASMO-4/SIMULATE-3 MOX/ SIMULATE-3K MOX codes are suitable replacements for the Westinghouse codes in the DRWM methodology. In addition, the responses provided in Reference 35 for the other technology transfer criteria are still applicable with the new codes.

6.2 Impact of MOX Fuel on DRWM

Duke anticipates using MOX fuel in up to 40% of the fuel assemblies in a mixed core with traditional LEU fuel assemblies. Some of the nuclear characteristics of MOX fuel that could affect DRWM are a lower core average delayed neutron fraction and higher fast to thermal neutron flux ratio. Duke uses very low leakage core designs at McGuire and Catawba and routinely places high burnup LEU fuel assemblies with appreciable amounts of plutonium on the core periphery. Because most of the neutrons detected by excore detectors during DRWM originate from the core boundary, the excore detector response and DRWM results in partial MOX fuel cores are not significantly different from the existing experience with cores containing only LEU fuel.

To quantify the impact of MOX fuel, simulations have been performed using typical anticipated partial MOX fuel core designs. The excore detector signal is affected by two competing factors. The slightly harder neutron spectra of partial MOX fuel cores increase the excore signal due to more incident neutrons at the detectors, and the smaller core average delayed neutron fraction produces lower flux levels at the fully inserted configuration. These studies showed that the minimum excore detector signal during DRWM will be slightly smaller in partial MOX fuel cores, but the reduction will not be significant relative to the large reductions that occur when a control rod bank is inserted into the core.

The relative contribution of each fuel location to the excore detector signal has been recalculated with typical anticipated partial MOX fuel core designs. These detector response factor distributions were calculated with and without spatial/isotope dependent fission energy spectra. These studies showed that there is essentially no change in the excore weighting factors for partial MOX fuel cores. Therefore, the existing DRWM methodology can be used to accurately measure control rod bank worth in partial MOX fuel cores.

6.3 Sensitivity of DRWM Results to Inaccuracies in the Core Models

Strong space-time effects occur during the DRWM procedure. These effects must be properly accounted for in the DRWM analytical factors in order to produce an accurate measured static rod worth. The analytical factors correct for flux redistribution and delayed neutron effects and are derived from core models. The sensitivity of the measured rod worths to errors in the core model must therefore be addressed.

Sensitivity studies were performed using a McGuire core model to determine the impact of perturbations in predicted control rod cross sections and fission neutron density to the final measured bank worths. The sensitivities for McGuire cores containing all LEU fuel were determined to be essentially the same as cores which contain a mixture of MOX and

LEU fuel. Thus it is concluded that the quality of DRWM results is not impacted by the presence of MOX fuel.

Table 6-1 Predicted Rod Worth Comparisons

		West (pcm)	Duk-MOX (pcm)	(D-W) / W (%)	D - W (pcm)	
C1C11	CA	397.4				
	CB	610.3				
	CC	888.0				
	CD	631.2				
	SA	232.6				
	SB	890.0				
	SC	443.0				
	SD	440.1				
	SE	494.2				
	Total	5027				D
M2C12	CA	336.8				
	CB	644.2				
	CC	811.7				
	CD	613.5				
	SA	288.2				
	SB	1040.1				
	SC	489.8				
	SD	490.8				
	SE	506.4				
	Total	5222				D
M1C13	CA	304.6				
	CB	645.3				
	CC	725.4				
	CD	569.9				
	SA	268.4				
	SB	978.1				
	SC	455.8				
	SD	455.4				
	SE	513.2				
	Total	4916				D

		West (pcm)	Duk-MOX (pcm)	(D-W) / W (%)	D - W (pcm)	
C2C10	CA	422.1				
	CB	552.9				
	CC	851.9				
	CD	563.3				
	SA	240.1				
	SB	916.2				
	SC	393.5				
	SD	393.7				
	SE	477.1				
	Total	4811				D
C1C12	CA	288.2				
	CB	696.9				
	CC	766.1				
	CD	478.2				
	SA	326.5				
	SB	782.1				
	SC	457.5				
	SD	461.6				
	SE	498.3				
	Total	4755				D
M2C13	CA	352.8				
	CB	643.3				
	CC	780.9				
	CD	609.2				
	SA	295.7				
	SB	908.7				
	SC	463.2				
	SD	469.2				
	SE	487.4				
	Total	5010				D

Table 6-2 Measured Rod Worth Comparisons

Case	Rod	West (pcm)	Duk-MOX (pcm)	(D-W) / W (%)	D - W (pcm)	
	CA	374.6				
	CB	634.7				
	CC	889.5				
	CD	695.0				
	SA	235.6				
	SB	889.7				
	SC	468.3				
	SD	462.9				
	SE	460.8				
	Total	5111				D
	CA	293.9				
	CB	667.3				
	CC	763.0				
	CD	624.0				
	SA	305.5				
	SB	1067.4				
	SC	511.1				
	SD	513.1				
	SE	489.0				
	Total	5234				D
	CA	290.4				
	CB	670.4				
	CC	709.3				
	CD	569.0				
	SA	262.9				
	SB	994.7				
	SC	464.1				
	SD	455.7				
	SE	513.3				
	Total	4930				D
C2C10	CA	377.8				
	CB	601.1				
	CC	885.9				
	CD	558.9				
	SA	236.4				
	SB	1004.5				
	SC	402.5				
	SD	403.5				
	SE	477.1				
	Total	4948				D
C1C12	CA	275.3				
	CB	719.6				
	CC	780.6				
	CD	467.2				
	SA	317.0				
	SB	814.6				
	SC	449.1				
	SD	474.3				
	SE	511.2				
	Total	4809				D
M2C13	CA	340.6				
	CB	690.4				
	CC	815.0				
	CD	598.9				
	SA	277.6				
	SB	984.6				
	SC	466.4				
	SD	478.6				
	SE	502.9				
	Total	5155				D

7.0 CONCLUSION

This report justifies the use of CASMO-4 based SIMULATE-3 MOX models for reload design of Westinghouse 193-assembly plants. Nuclear uncertainty factors are provided for application to fuel assemblies (LEU fuel) in which the initial heavy metal loading is 100% uranium. Separate nuclear uncertainty factors are provided for application to fuel assemblies (MOX fuel) in which the initial heavy metal loading is a mixture of uranium and plutonium. This report further demonstrates Duke Power's competence in the application of CASMO-4 based SIMULATE-3 MOX models to reload design. The methodology presented herein supplements previous topical reports submitted by Duke Power (References 1 through 7 and Reference 36) describing models and methods for performing reload design calculations.

The report presents benchmarking of the CASMO-4/SIMULATE-3 MOX methodology to numerous McGuire and Catawba cycles of operation with LEU fuel. In addition, the report presents extensive benchmarking of this methodology to operating cycles of the St. Laurent B1 reactor in which a mixture of LEU and MOX fuel was used. Comparisons of calculated and measured data are presented from beginning of cycle hot zero power startup testing as well as operating data (power distributions and coolant boron concentrations). Peaking factor uncertainties for application of this methodology to calculation of core operating limits and Reactor Protection System limits are provided. These uncertainties are derived in a manner consistent with the previously approved methods of Reference 2 and are similar to the uncertainties for LEU cores.

The report also presents benchmarking of the CASMO-4/SIMULATE-3 MOX methodology to fuel rod power distributions measured in the B&W, Saxton, ERASME/L, and EPICURE critical experiments. The B&W critical experiments were comprised exclusively of LEU fuel pins, and the Saxton, ERASME/L, and EPICURE critical experiments that were benchmarked were comprised of either all MOX fuel pins or a combination of MOX and LEU fuel pins. These benchmarks form the basis for the fuel pin power uncertainties that are included as part of the overall nuclear uncertainty factors.

Finally, this report compares DRWM results from the CASMO-4/SIMULATE-3 MOX/ SIMULATE-3K MOX code package to Westinghouse results for the same McGuire/Catawba fuel cycles used to obtain approval for Duke's current DRWM methodology (Reference 3). These comparisons indicate that the CASMO-4/ SIMULATE-3 MOX/ SIMULATE-3K MOX methodology can be used to successfully support DRWM at McGuire and Catawba. This report also addresses the application of DRWM to cores comprised of a mixture of LEU and MOX fuel assemblies, and shows that the DRWM methodology is appropriate for control rod worth measurement in such cores.

8.0 REFERENCES

1. "Duke Power Company, McGuire/Catawba Nuclear Physics Methodology for Reload Design," DPC-NE-2010A, June, 1985.

2. "Duke Power Company, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," DPC-NE-1004A, Revision 1, April 1996.

3. "Duke Power Company, Dynamic Rod Worth Using CASMO/SIMULATE," DPC-NE-2012A, February 2000.

4. "Duke Power Company, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," DPC-NE-2011PA, March 1990.

5. "Duke Power Company, McGuire/Catawba Multi-dimensional Reactor Transients and Safety Analysis Physics Parameters," DPC-NE-3001PA, December 2000.

6. "Duke Power Company, Fuel Reconstitution Analysis Methodology," DPC-NE-2007PA, October 1995.

7. "Duke Power Company, Westinghouse Fuel Transition," DPC-NE-2009PA, September 1999.

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.

15. M.G. Natrella, "Experimental Statistics," National Bureau of Standards Handbook 91, October 1966.

16. "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification," U.S. Nuclear Regulatory Commission, Regulatory Guide 1.126, Revision 1, March 1978.

17. D.B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandia Corporation, pp. 46-54 (Table 2.4), March 1963.

18. "Assessment of the Assumption of Normality (Employing Individual Observed Values)," ANSI-N15.15-1974, October 1973.

19. "Operating Data for MOX Fuel Power Plant," Electricité de France, D4510/NT/BC/MET/N99.76 (Proprietary), July 1999.

20. "Operation Monitoring and Power Diagrams" (Proprietary), Electricité de France, June 1999.

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

22. Masaaki Mori, Mitsuru Kawamura, and Koichi Yamate, "CASMO-4/SIMULATE-3 Benchmarking Against High Plutonium Content Pressurized Water Reactor Mixed-Oxide Fuel Critical Experiment", Nuclear Science and Engineering, Volume 121, September 1995.

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

24. "Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," DOE/ET/34212-41, BAW-1810, April 1984.

25. E.G. Taylor, "Saxton Plutonium Program, Critical Experiments for the Saxton Partial Plutonium Core," EURAEC-1493, WCAP-3385-54, December 1965.

26. N.M. Abdurrahman, G. Radulescu, Igor Carron, "Benchmark Calculations of the Saxton Plutonium Program Critical Experiments," Nuclear Technology, Volume 127, September 1999.

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.]F

35. "Westinghouse Dynamic Rod Worth Measurement Technique," WCAP-13360-P-A, Revision 1, October 1998.

36. "Duke Power Company, Rod Swap Methodology Report for Startup Physics Testing," DPC-NE-1003A, May 1987.

37. E. Saji, Y. Inoue (Toden Software–Japan), M. Mori, T. Ushio (NEL–Japan), "Validation of the Nuclear Design Method for MOX Fuel Loaded LWR Cores," Transactions of the 2001 American Nuclear Society Annual Meeting, June 20, 2001.

[Page intentionally left blank]

APPENDIX A – MIXED OXIDE FUEL USE IN DUKE POWER'S MCGUIRE AND CATAWBA REACTORS

A.1 United States Mixed Oxide (MOX) Fuel Project

The United States MOX Fuel Project is part of an international nonproliferation program that has the goal of disposing of surplus weapons plutonium in the United States and Russia. MOX Fuel Project plans call for developing a MOX Fuel Fabrication Facility (MFFF) on the Department of Energy's Savannah River Site. The Duke Cogema Stone & Webster (DCS) company will operate the MFFF to produce MOX fuel for use in the McGuire and Catawba Nuclear Stations. DCS will manufacture the fuel using the Micronized Master Blend (MIMAS) process, which is the same process that was used to make the St. Laurent B1 MOX fuel. Consistent with the program goal of disposing of weapons-usable material, the MOX fuel will contain plutonium with weapons grade isotopics.

A MOX fuel lead test assembly program at either McGuire or Catawba will precede the use of significant quantities of MOX fuel. The eventual schedules for the use of MOX fuel at McGuire and Catawba are dependent on various factors, including Nuclear Regulatory Commission (NRC) reviews, U.S. Department of Energy (DOE) actions, international agreements, and plutonium disposition activities in Russia. Based on the number and type of external factors involved, the currently contemplated schedule is subject to change.

The purpose of this appendix is to provide information about the manner in which Duke currently intends to use MOX fuel in the McGuire and Catawba reactors. This information is based on preliminary MOX fuel assembly and partial MOX fuel core design information, as described in Reference A.1. The ultimate MOX fuel assembly design and core management approach may change from the concepts provided herein.

A.2 Planned MOX Fuel Assembly Design (Typical)

The McGuire and Catawba reactors use a 17x17 pressurized water (PWR) fuel assembly design. The Mark-BW/MOX1 fuel assembly is planned for deployment with MOX fuel in these reactors. The fuel assembly lattice is characterized by a central instrument tube, 24 control rod guide tubes, and 264 fuel pins. The Mark-BW/MOX1 fuel assembly mechanical design is based on the proven Mark-BW design that has been deployed at the McGuire and Catawba mission reactors for many years. The Mark-BW/MOX1 fuel assembly will contain the features of the current Mark-BW design, plus M5™ fuel pin cladding and, where necessary for compatibility with the resident fuel, mid-span mixing grids (MSMGs). Figure A-1 depicts the major features of the Mark-BW/MOX1 fuel assembly design. This fuel assembly for UO_2 applications, with the M5™ cladding and MSMGs, is designated the Advanced Mark-BW. The Mark-BW/MOX1 differs from the Advanced Mark-BW design primarily in that the fuel pellets are MOX instead of UO_2.

The planned MOX fuel assembly design will use multiple concentrations of plutonium in each assembly as shown in the radial fuel assembly zoning diagram (Figure A-2). In this context, the plutonium concentration refers to the mass ratio of plutonium to total heavy metal (plutonium plus uranium). Using multiple fuel pin concentration zones minimizes the intra-assembly power peaking that results from the sharp thermal neutron flux gradient between adjacent uranium and MOX fuel assemblies.

Key MOX fuel pin and assembly design parameters are summarized in Table A-1.

A.3 Planned Partial MOX Fuel Core Management (Typical)

Fuel management refers to the arrangement and characteristics of fuel assemblies and other components within the reactor core. Typical pressurized water reactor cores are a mixture of fuel assemblies that are in their first, second, or third cycle of irradiation. Duke Power currently employs a modified checkerboard feed pattern with face-adjacent feed assemblies. Figure A-3 illustrates this type of loading pattern. For core designs that utilize feed batch consisting of both MOX and low-enriched uranium (LEU) assemblies, the basic loading pattern is proposed to be very similar to that shown in Figure A-4. The basic core design remains a checkerboard with face-adjacent feed assemblies. Both all-LEU and MOX/LEU core designs are low-leakage designs, with once- or twice-burned fuel on the core periphery.

Major assumptions and constraints associated with partial MOX fuel core designs are as follows:

1. Maximum MOX fuel pin burnup is 50 gigawatt days per metric ton heavy metal (GWd/Mthm),
2. Maximum LEU fuel pin burnup is approximately 60 GWd/Mthm,
3. MOX fuel is discharged after two cycles,
4. MOX fuel peaking limits are similar to uranium fuel limits, and
5. MOX fuel core fractions are limited to approximately 40%.

For reactivity control, the current McGuire/Catawba equilibrium LEU cores make extensive use of integral fuel burnable absorber (IFBA) and discrete burnable poison (BP) rods in the uranium fuel. For partial MOX fuel cores, Duke plans to use IFBA and discrete BPs in the LEU fuel assemblies and discrete BPs only in the MOX fuel assemblies. Since BP rods occupy the same positions in the assembly as would control rods, the loading pattern must take into account the positions of these control rods. The control rod locations, by control group, are shown in Figure A-5.

A.4 References

(1) DCS-FQ-1999-001, Revision 2, Fuel Qualification Plan, Framatome ANP (US), April 2001.

Table A-1 Typical MOX Fuel Design Characteristics

Parameter	Value
Pellets	
Fuel Pellet Material	Ceramic PuO_2 and Depleted UO_2
Fuel Pellet Diameter	0.3225 in
Fuel Pellet Theoretical Density	~ 95%
Fuel Pellet Volume Reduction due to Chamfer and Dish	~ 1%
Pins	
Fuel Pin Length	152.4 in
Fuel Pin Cladding Material	M5™
Fuel Pin Inside Diameter	0.329 in
Fuel Pin Outside Diameter	0.374 in
Active Fuel Stack Height	144 in
Assemblies	
Fuel Assembly Length	159.8 in
Lattice Geometry	17x17
Fuel Pin Pitch	0.496 in
Number of Fuel Pins per Assembly	264
Heavy Metal Loading per Assembly	462.6 kg
Number of Grids	
Bottom End	1
Vaneless Intermediate	1
Vaned Intermediate	5
Mid-Span Mixing	3
Top End	1

Figure A-1 Mark BW/MOX1 Fuel Assembly

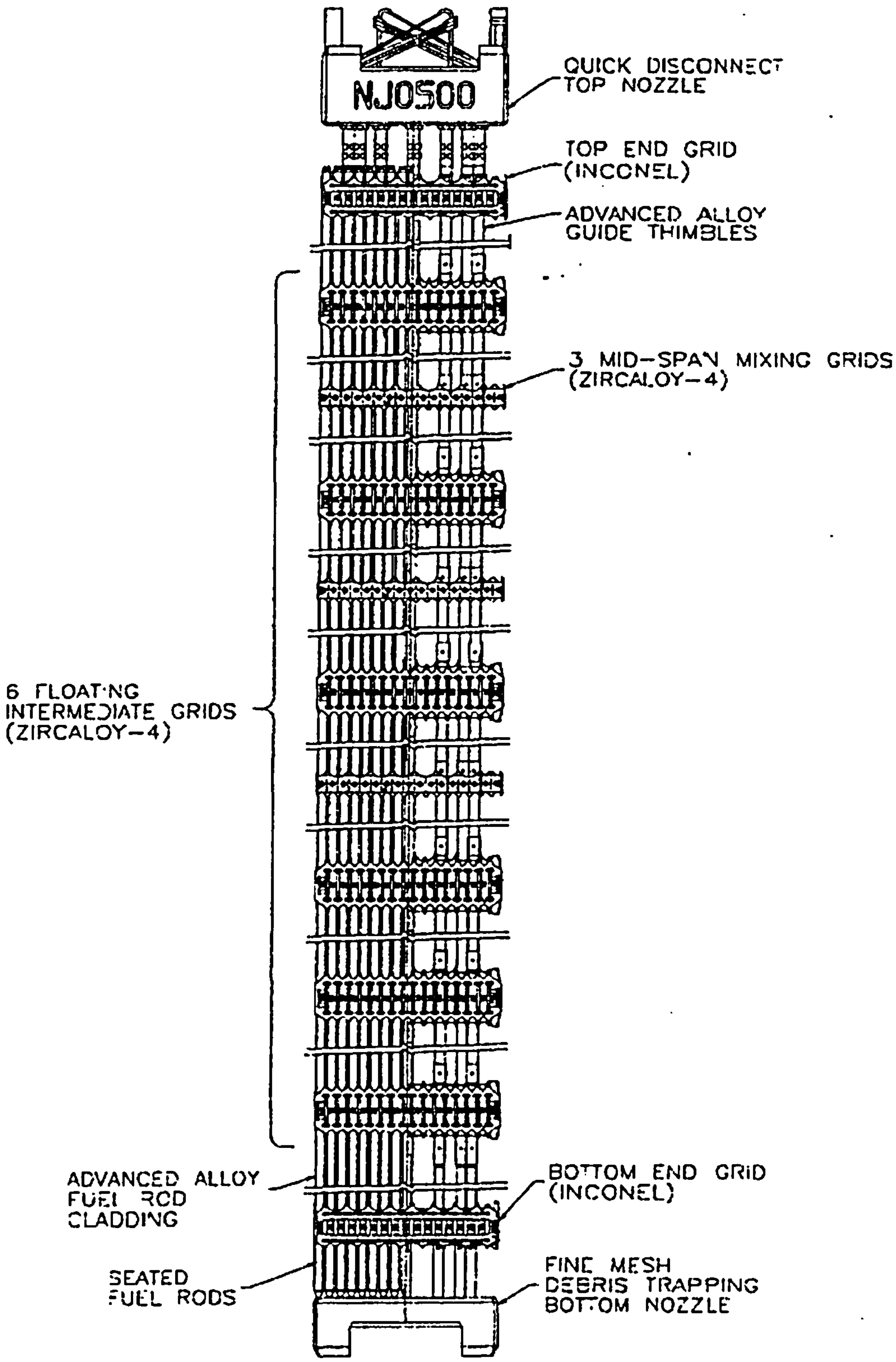


Figure A-2 Typical 17x17 Mark BW/MOX1 Fuel Assembly Configuration

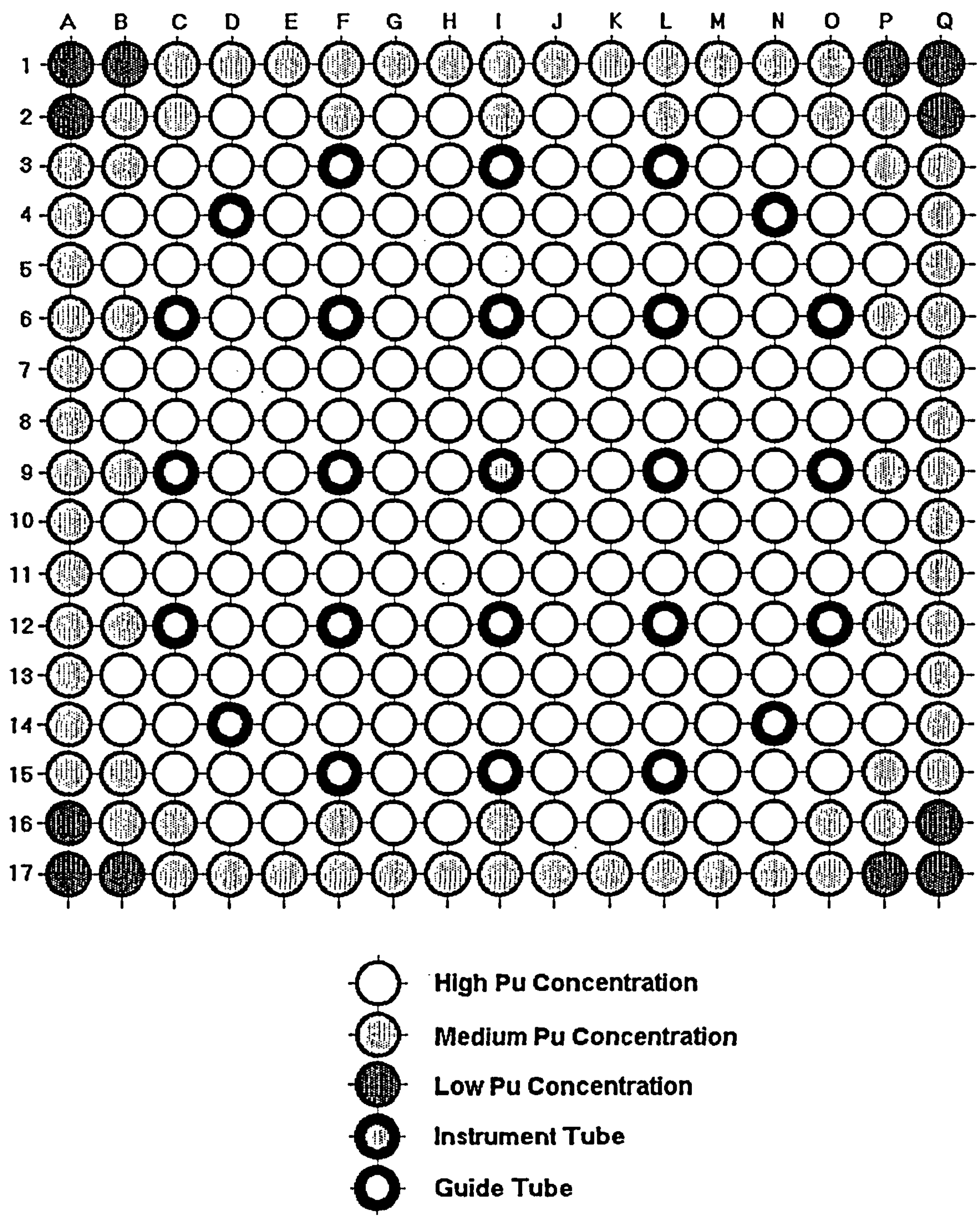


Figure A-3 Typical Loading Pattern for All LEU Core

	H	G	F	E	D	C	B	A
8	C-10 LEU 2	FEED LEU 0	G-08 LEU 1	C-12 LEU 1	P-08 LEU 1	N-13 LEU 1	FEED LEU 0	G-15 LEU 2
9	FEED LEU 0	C-08 LEU 1	FEED LEU 0	G-12 LEU 1	FEED LEU 0	E-06 LEU 1	FEED LEU 0	P-10 LEU 1
10	H-09 LEU 1	FEED LEU 0	C-04 LEU 1	FEED LEU 0	L-04 LEU 1	FEED LEU 0	FEED LEU 0	P-05 LEU 1
11	M-13 LEU 1	D-09 LEU 1	FEED LEU 0	J-15 LEU 2	FEED LEU 0	G-10 LEU 1	FEED LEU 0	R-06 LEU 2
12	H-02 LEU 1	FEED LEU 0	M-05 LEU 1	FEED LEU 0	E-08 LEU 1	FEED LEU 0	P-09 LEU 1	
13	N-03 LEU 1	K-11 LEU 1	FEED LEU 0	F-09 LEU 1	FEED LEU 0	FEED LEU 0	L-13 LEU 3	
14	FEED LEU 0	FEED LEU 0	FEED LEU 0	FEED LEU 0	G-02 LEU 1	C-05 LEU 3		
15	R-09 LEU 1	F-02 LEU 1	L-02 LEU 1	K-01 LEU 1	Previous Cycle Location (or Feed) Fuel Type (LEU or MOX) Cycles Previously in Core			

Figure A-4 Typical Loading Pattern for Equilibrium 40% MOX Fuel Core

	H	G	F	E	D	C	B	A
8	L-08 LEU 1	FEED LEU 0	C-06 LEU 1	[illegible] MOX 1	K-13 LEU 1	FEED MOX 0	B-09 MOX 1	N-08 LEU 1
9	FEED LEU 0	G-12 LEU 1	FEED LEU 0	[illegible] MOX 1	FEED LEU 0	F-02 LEU 1	FEED MOX 0	R-10 LEU 2
10	F-13 LEU 1	FEED LEU 0	G-08 LEU 1	FEED MOX 0	[illegible] MOX 1	FEED LEU 0	FEED LEU 0	E-04 LEU 1
11	[illegible] MOX 1	[illegible] MOX 1	FEED MOX 0	D-09 LEU 1	FEED LEU 0	[illegible] MOX 1	FEED MOX 0	P-04 LEU 2
12	N-06 LEU 1	FEED LEU 0	[illegible] MOX 1	FEED LEU 0	[illegible] MOX 1	FEED MOX 0	F-09 LEU 1	
13	FEED MOX 0	P-10 4.41 1	FEED LEU 0	[illegible] MOX 1	FEED MOX 0	FEED MOX 0	L-15 LEU 3	
14	[illegible] MOX 1	FEED MOX 0	FEED LEU 0	FEED MOX 0	G-10 LEU 1	A-05 LEU 3		
15	H-03 LEU 1	F-01 LEU 2	M-11 LEU 1	M-02 LEU 2	Previous Cycle Location (or Feed) Fuel Type (LEU or MOX) Cycles Previously in Core			

Figure A-5 Control Rod Location in McGuire/ Catawba Cores

	H	G	F	E	D	C	B	A
8	C_D		C_A		S_E		C_C	
9						S_B		
10	C_A		C_C				C_B	
11						S_C		
12	S_E				C_D		S_A	
13		S_B		S_D				
14	C_C		C_B		S_A			
15								

Control Rod Group

ERRATA

The following editorial errors were corrected in the final published version of this report.

1) p. 6-1, second paragraph, third sentence – Changed "Reference 36" to "Reference 3"

2) p. 6-1, third paragraph, first sentence – Changed "Reference 36" to Reference 3"

3) p. 6-2, first paragraph, first sentence – Changed "Table 6-1" to "Tables 6-1 and 6-2"

4) p. 6-2, first paragraph, second sentence – Changed "[ ]D" to "[ ]D"

5) p. 6-2, first paragraph, fourth sentence – Changed "[ ]D" to "[ ]D"

6) p. 6-2, third paragraph, second sentence – Changed "effect" to "affect"