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A Duke Energy Company

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February 27, 2003

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

Subject: Duke Energy Corporation
Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414
McGuire Nuclear Station Units 1 & 2, Docket Nos. 50-369, 50-370
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

Pursuant to 10 CFR 50.90, attached is a license amendment request for the McGuire and Catawba Nuclear Station Facility Operating Licenses and Technical Specifications. The changes proposed by Duke Energy Corporation (Duke) in this submittal revise the Technical Specifications to allow the use of four MOX fuel lead assemblies at McGuire or Catawba Nuclear Station. The current lead assembly fabrication schedule will support the insertion of MOX fuel lead assemblies into either McGuire Unit 2 or Catawba Unit 1 during the Spring 2005 refueling outage for the selected unit. No decision has yet been made as to which unit will irradiate the lead assemblies. Obtaining regulatory approval for MOX fuel lead assembly use at all four McGuire and Catawba units will better enable Duke to adjust to changes in the lead assembly fabrication schedule, should any such changes occur.

This submittal also includes a Request for Exemptions from selected regulations pursuant to 10 CFR 50.12 that are required in order to receive, handle, store and use MOX fuel lead assemblies.

The following attachments are included in this submittal.

Attachment 1 contains a marked copy of the current McGuire Technical Specifications and associated Bases showing the proposed changes to allow insertion of MOX fuel lead assemblies.

Attachment 2 contains a marked copy of the current Catawba Technical Specifications and associated Bases showing the proposed changes to allow insertion of MOX fuel lead assemblies.

Attachment 3 contains background information, a discussion of each of the proposed changes, and supporting technical information to justify the changes.

A001

Attachment 4 contains Duke's no significant hazards consideration analysis using the standards in 10 CFR 50.92.

Attachment 5 contains Duke's assessment of the environmental consequences of the proposed license and Technical Specification changes.

Attachment 6 contains a request for exemptions from selected NRC regulations in 10 CFR 50.

This license amendment request is being made as part of the ongoing United States – Russian Federation plutonium disposition program. The goal of this nuclear nonproliferation program is to dispose of surplus plutonium from nuclear weapons by converting the material into MOX fuel and using that fuel in nuclear reactors. Additional background information on the intended role of the McGuire and Catawba reactors in the plutonium disposition program is provided in Attachments 3 and 5.

Implementation of this amendment to the McGuire and Catawba Facility Operating Licenses and Technical Specifications is not expected to require changes to the plants' Updated Final Safety Analysis Reports (UFSARs). If, as a result of implementing this license amendment, Duke Energy determines that UFSAR changes are needed, appropriate changes will be made and submitted to the NRC in accordance with 10 CFR 50.71(e). Duke requests that the Nuclear Regulatory Commission issue these license amendments and exemptions by no later than August 2004. Regulatory approval in accordance with this schedule will facilitate orderly reload core design and fuel procurement for the Spring 2005 McGuire or Catawba refueling outage.

It should be noted that this license amendment request relies in part on two topical reports that are currently under NRC review. They are DPC-NE-1005P, *Nuclear Design Methodology Using CASMO-4/ SIMULATE-3 MOX* and BAW-10231P, *COPERNIC Fuel Rod Design Computer Code*. NRC approval of these reports is a prerequisite to implementation of the Technical Specification amendments identified in this amendment request.

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

Pursuant to 10 CFR 50.91, copies of these proposed amendments are being sent to the States of North Carolina and South Carolina.

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

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc:

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Oath and Affirmation

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

M. S Tuckman

M. S. Tuckman

Subscribed and sworn to before me on this 27TH day of February, 2003

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2006

SEAL

Attachment 1

Marked Copy of McGuire Technical Specifications

and

Associated BASES Changes

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment, burnup and number of Integral Fuel Burnable Absorber (IFBA) rods of each new or spent fuel assembly stored in the spent fuel pool storage racks shall be within the following configurations:

a. New or irradiated fuel may be stored in Region 1A of the spent fuel pool in accordance with these limits:

1. Unrestricted storage of new fuel meeting the criteria of Table 3.7.15-1; or

2. Unrestricted storage of fuel meeting the criteria of Table 3.7.15-2; or

3. Restricted storage in accordance with Figure 3.7.15-1, of fuel which does <u>not</u> meet the criteria of Table 3.7.15-2; or

4. Restricted storage, in accordance with Figure 3.7.15-1, of MOX fuel assemblies as Restricted Fuel.

b. New or irradiated fuel may be stored in Region 1B of the spent fuel pool in accordance with these limits:

1. Unrestricted storage of fuel meeting the criteria of Table 3.7.15-4; or

2. Restricted storage in accordance with Figure 3.7.15-2, of fuel which meets the criteria of Table 3.7.15-5; or

3. Checkerboard storage in accordance with Figure 3.7.15-3 of fuel which does <u>not</u> meet the criteria of Table 3.7.15-5; or

4. Checkerboard storage, in accordance with Figure 3.7.15-3, of MOX fuel assemblies as Checkerboard Fuel.

c. New or irradiated fuel which has decayed at least 16 days may be stored in Region 2A of the spent fuel pool in accordance with these limits:

1. Unrestricted storage of fuel meeting the criteria of Table 3.7.15-7; or

2. Restricted storage in accordance with Figure 3.7.15-4, of fuel which meets the criteria of Table 3.7.15-8; or

3. Checkerboard storage in accordance with Figure 3.7.15-5 of fuel which does <u>not</u> meet the criteria of Table 3.7.15-8; or

4. Checkerboard storage, in accordance with Figure 3.7.15-5, of MOX fuel assemblies as Checkerboard Fuel.

4. Checkerboard storage, in accordance with Figure 3.7.15-7, of MOX fuel assemblies as Checkerboard Fuel.

d. New or irradiated fuel which has decayed at least 16 days may be stored in Region 2B of the spent fuel pool in accordance with these limits:

1. Unrestricted storage of fuel meeting the criteria of Table 3.7.15-10; or

2. Restricted storage in accordance with Figure 3.7.15-6, of fuel which meets the criteria of Table 3.7.15-11; or

3. Checkerboard storage in accordance with Figure 3.7.15-7 of fuel which does not meet the criteria of Table 3.7.15-11; or

4.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

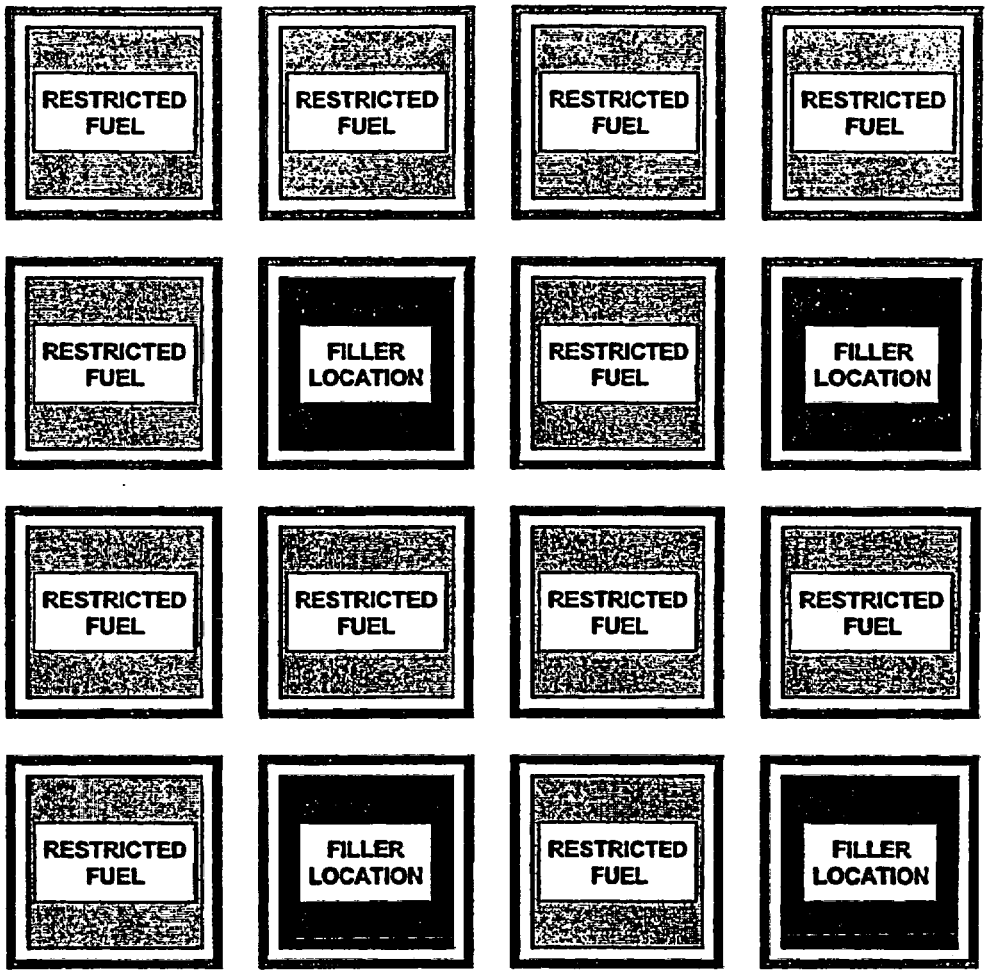
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to the correct location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the planned spent fuel pool location is acceptable for the fuel assembly being stored.	Prior to storing the fuel assembly in the spent fuel pool

Fuel which: a) does not meet the minimum burnup requirements of either Table 3.7.15-1 or Table 3.7.15-2; or b) is a mixed oxide fuel assembly with a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent.

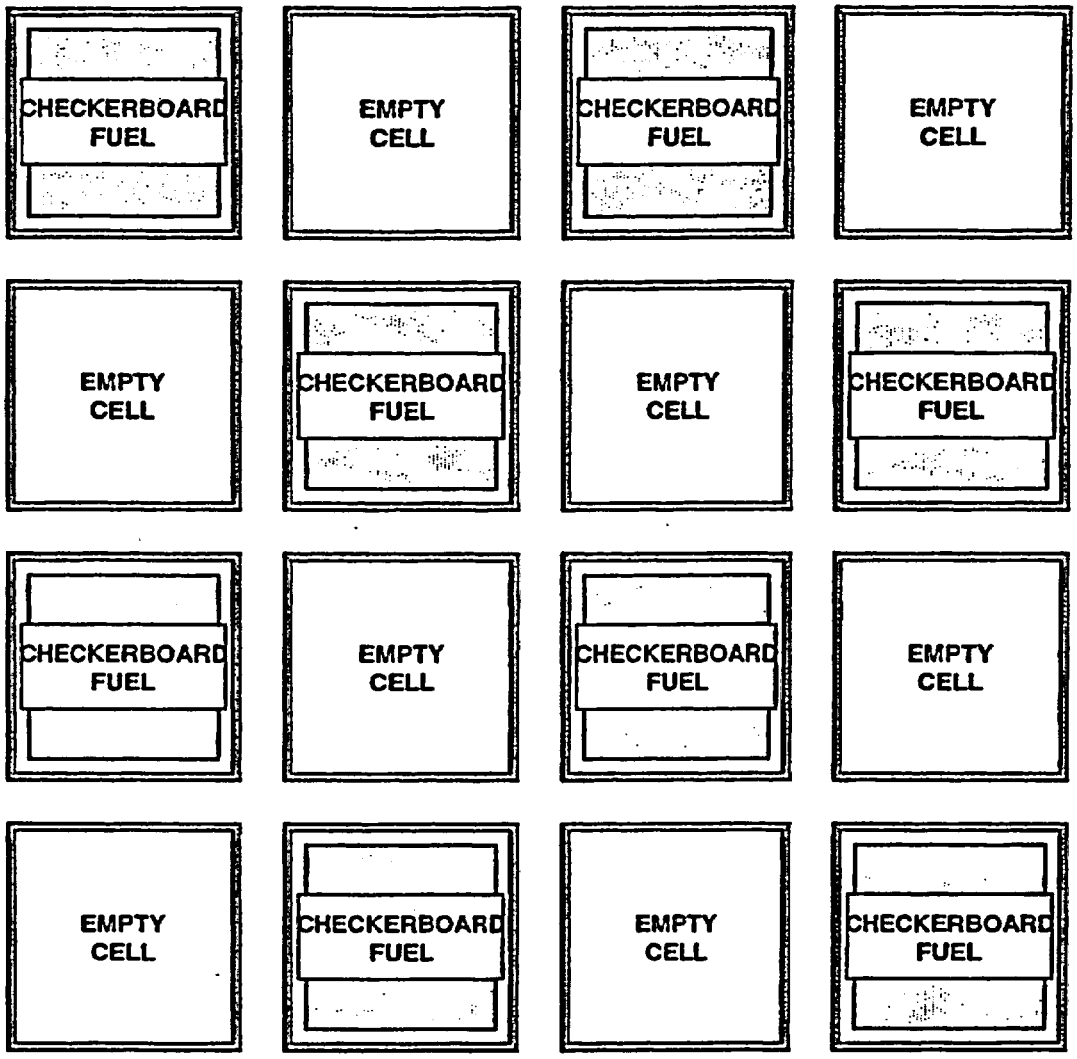


Restricted Fuel: ~~(Fuel which does not meet the minimum burnup requirements of either Table 3.7.15-1 or Table 3.7.15. 2.)~~ (Fuel which does meet the requirements of Table 3.7.15-1 or Table 3.7.15. 2, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed).

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.7.15-3, or an empty cell.

Boundary Condition: Any Restricted Region 1A Storage Area row bounded by any other storage area shall contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other. Example: In the figure above, row 1 or column 1 can not be adjacent to another storage area, but row 4 or column 4 can be.

Figure 3.7.15-1 (page 1 of 1)
Required 3 out of 4 Loading Pattern for Restricted Region 1A Storage

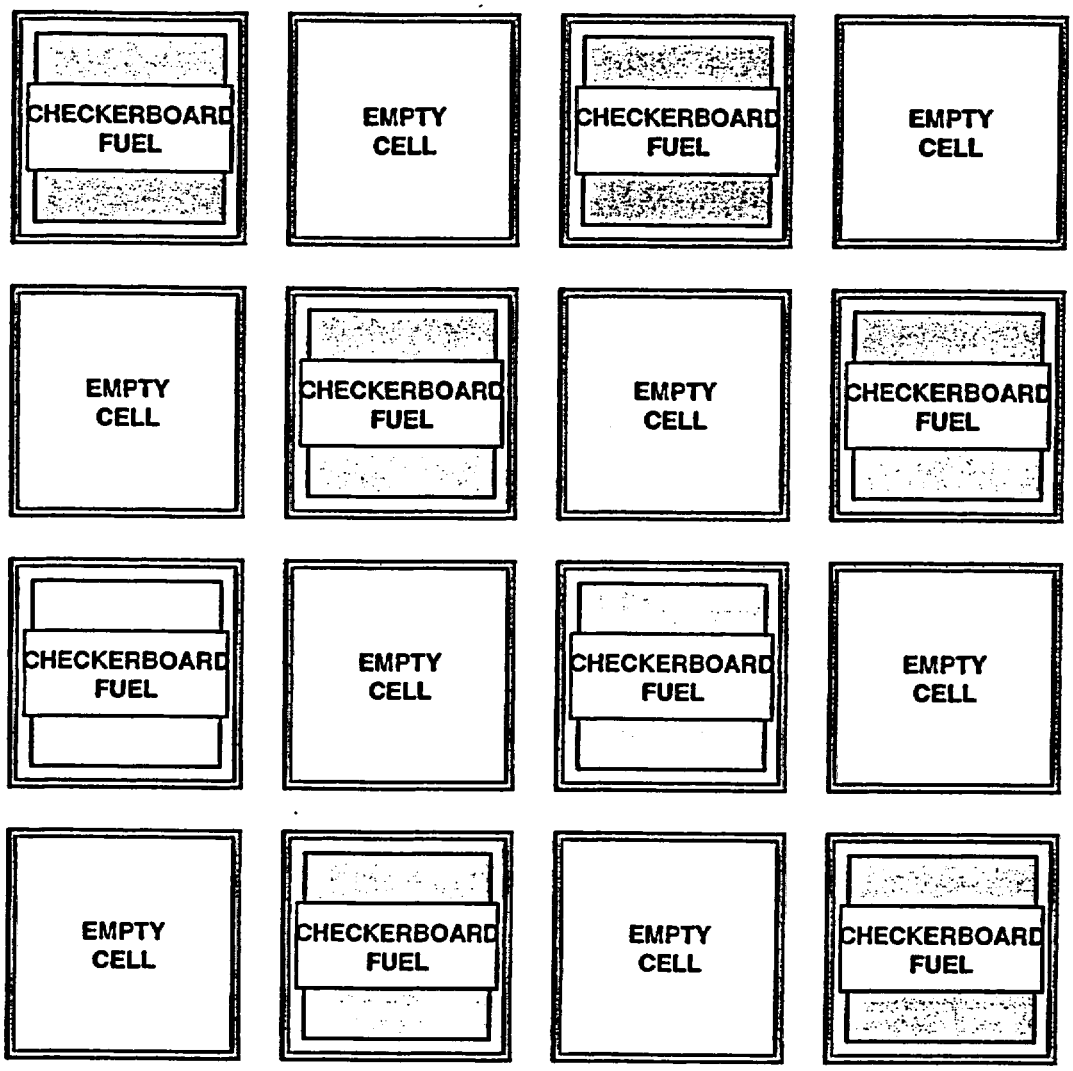


Checkerboard Fuel: ~~Fuel which does not meet the minimum burnup requirements of Table 3.7.15-5~~ (Fuel which does meet the requirements of Table 3.7.15-5, or non-fuel components, or an empty location may be placed in checkerboard fuel locations as needed)

Boundary Condition: Any Checkerboard Region 1B Storage Area must be separated from any other storage area by at least one row of empty cells, at all boundaries between storage regions.

Fuel which: a) does not meet the minimum burnup requirements of Table 3.7.15-5; or b) is a mixed oxide fuel assembly with a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent.

Figure 3.7.15-3 (page 1 of 1)
Required 2 out of 4 Loading Pattern for Checkerboard Region 1B Storage

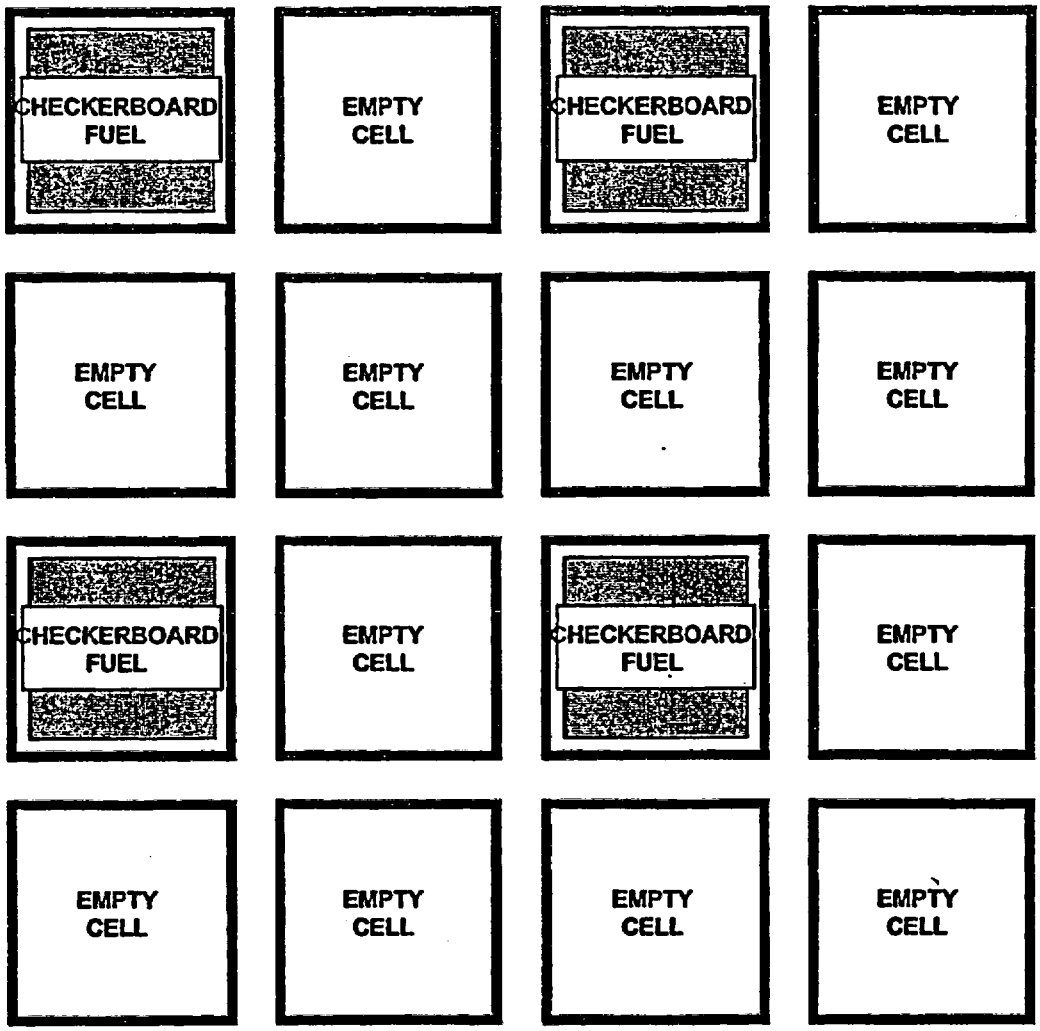


Checkerboard Fuel: ~~Fuel which does not meet the minimum burnup requirements of Table 3.7.15-8.~~ (Fuel which does meet the requirements of Table 3.7.15-8, or non-fuel components, or an empty location may be placed in checkerboard fuel locations as needed)

Boundary Condition: At least three of the four faces of each Checkerboard Fuel Assembly must be adjacent to an empty cell or the pool wall, at all boundaries between storage regions.

Fuel which: a) does not meet the minimum burnup requirements of Table 3.7.15-8; or b) is a mixed oxide fuel assembly with a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent.

Figure 3.7.15-5 (page 1 of 1)
Required 2 out of 4 Loading Pattern for Checkerboard Region 2A Storage



Checkerboard Fuel: ~~Fuel which does not meet the minimum burnup requirements of Table 3.7.15-11.~~ (Fuel which does meet the requirements of Table 3.7.15-11, or non-fuel components, or an empty location may be placed in checkerboard fuel locations as needed)

Boundary Condition: Any Checkerboard Region 2B Storage Area row bounded by any other storage area shall contain only empty cells arranged such that no Checkerboard Fuel assemblies are adjacent to any fuel. Example: In the figure above, row 1 or column 1 can not be adjacent to another storage area, but row 4 or column 4 can be.

Figure 3.7.15-7 (page 1 of 1)
Required 1 out of 4 Loading Pattern for Checkerboard Region 2B Storage

Fuel which: a) does not meet the minimum burnup requirements of Table 3.7.15-11; or b) is a mixed oxide fuel assembly with a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent.

4.0 DESIGN FEATURES

4.1 Site Location

The McGuire Nuclear Station site is located at latitude 35 degrees, 25 minutes, 59 seconds north and longitude 80 degrees, 56 minutes, 55 seconds west. The Universal Transverse Mercator Grid Coordinates are E 504, 669, 256, and N 3, 920, 870, 471. The site is in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina.

4.2 Reactor Core

Insert

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material.* Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Insert →

* A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into either the Unit 1 or Unit 2 reactor core.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium (Unit 1) silver indium cadmium and boron carbide (Unit 2) as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 4.75 weight percent;

or mixed oxide fuel assemblies having a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 850 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

5. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," (DPC Proprietary).

6. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design".

7. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology".

8. DPC-NE-3000PA, "Thermal-Hydraulic Transient Analysis Methodology ," (DPC Proprietary).

9. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P".

10. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," (DPC Proprietary).

11. DPC-NE-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," (DPC Proprietary).

12. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," (DPC Proprietary).

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," (W Proprietary).

14. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report," (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," (W Proprietary).

Insert 1.

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

(continued)

INSERT 1 to McGuire TS 5.6.5 b.

16. DPC-NE-1005P-A, "Duke Power Nuclear Design Methodology Using CASMO-4/ SIMULATE-3 MOX," (Duke Proprietary).

17. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," (Framatome ANP Proprietary).

BASES

BACKGROUND (continued)

configuration with lower reactivity fuel. A third loading pattern, Checkerboard storage, was defined for Regions 1B, 2A and 2B. Checkerboard storage allows storage of the highest reactivity fuel in each region when checkerboarded with empty storage cells.

or mixed oxide fuel assemblies with a maximum nominal fissile plutonium concentration up to 4.15 weight percent (maximum tolerance of +/- 0.075 weight percent fissile Pu) and a maximum nominal Uranium-235 enrichment of 0.35 weight percent;

The mixed oxide fuel assembly design is radially zoned with fuel rods at three different plutonium concentrations. The nominal fissile plutonium concentration limit is the weighted average for the entire fuel assembly.

The McGuire spent fuel storage racks have been analyzed taking credit for soluble boron as allowed in Reference 3. The methodology ensures that the spent fuel rack multiplication factor, k_{eff}, is less than or equal to 0.95 as recommended in ANSI/ANS-57.2-1983 (Ref. 4) and NRC guidance (Ref. 5). The spent fuel storage racks are analyzed to allow storage of fuel assemblies with enrichments up to a maximum nominal enrichment of 4.75 weight percent Uranium-235 while maintaining $k_{eff} \leq$ 0.95 including uncertainties, tolerances, bias, and credit for soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions and to provide subcritical margin such that the spent fuel pool k_{eff} is maintained less than or equal to 0.95. The soluble boron concentration required to maintain k_{eff} less than or equal to 0.95 under normal conditions is 850 ppm. In addition, sub-criticality of the pool ($k_{eff} < 1.0$) is assured on a 95/95 basis, without the presence of the soluble boron in the pool. The criticality analysis performed shows that the acceptance criteria for criticality is met for the storage of fuel assemblies when credit is taken for reactivity depletion due to fuel burnup, the presence of Integral Fuel Burnable Absorber (IFBA) rods, reduced credit for the Boraflex neutron absorber panels and storage configurations and enrichment limits Specified by LCO 3.7.15.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in reactivity of the racks in the spent fuel pool. Examples of these accident conditions are the drop of a fuel assembly on top of a rack, the drop of a fuel assembly between rack modules (rack design precludes this condition), and the drop of a fuel assembly between rack modules and the pool wall. However, three accidents can be postulated which could result in an increase in reactivity in the spent fuel storage pools. The first is a drop or placement of a fuel assembly into the cask loading area. The second is a significant change in the spent fuel pool water temperature (either the loss of normal cooling to the spent fuel pool water which causes an increase in the pool water temperature or a large makeup to the pool with cold water which causes a decrease in the pool water temperature) and the third is the misloading of a fuel assembly into a location which the restrictions on location, enrichment, burnup and number of IFBA rods is not satisfied.

For an occurrence of these postulated accidents, the double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter

Attachment 2

Marked Copy of Catawba Technical Specifications

and

Associated BASES Changes

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 The combination of initial enrichment and burnup of each new or spent fuel assembly stored in the spent fuel pool storage racks shall be within the following configurations:

a. Unrestricted storage of fuel meeting the criteria of Table 3.7.16-1; or

b. Restricted storage in accordance with Figure 3.7.16-1, of fuel which does not meet the criteria of Table 3.7.16-1; or

Add c. Restricted storage, in accordance with Figure 3.7.16-1, of MOX fuel assemblies as Restricted Fuel.

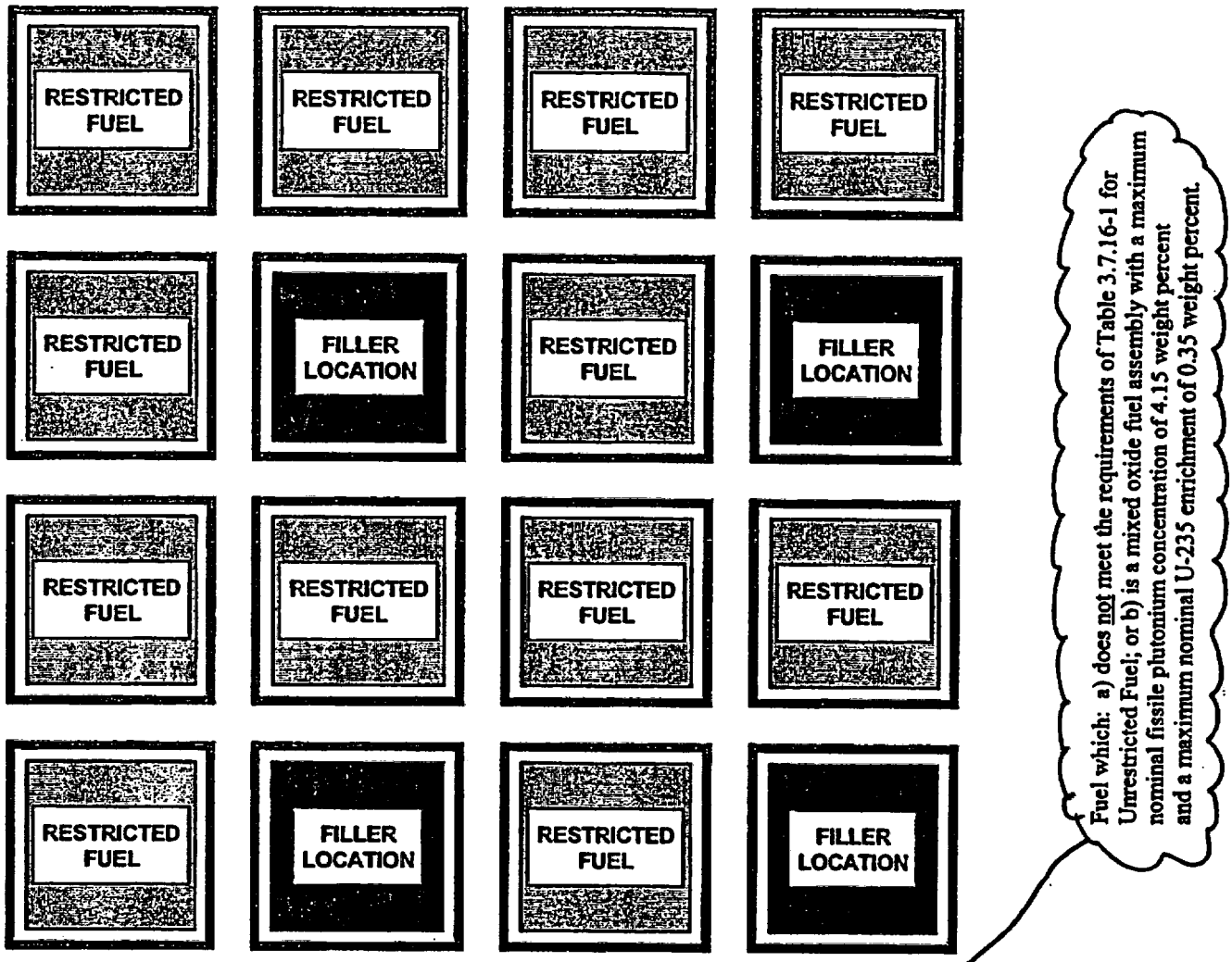
APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ——NOTE—— LCO 3.0.3 is not applicable. ——— Initiate action to move the noncomplying fuel assembly to the correct location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 ~~Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with the specified configurations.~~ Verify by administrative means the planned spent fuel pool location is acceptable for the fuel assembly being stored.	Prior to storing the fuel assembly in the spent fuel pool



Restricted Fuel: ~~Fuel defined for Restricted Storage in Table 3.7.16-1.~~ (Fuel defined for Unrestricted Storage in Table 3.7.16-1, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.7.16-2, or an empty cell.

Boundary Condition: Any row bounded by an Unrestricted Storage Area shall contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other. Example: In the figure above, row 1 or column 1 can not be adjacent to an Unrestricted Storage Area, but row 4 or column 4 can be.

Figure 3.7.16-1
Required 3 out of 4 Loading Pattern for Restricted Storage

4.0 DESIGN FEATURES

4.1 Site Location

Catawba Nuclear Station is located in the north central portion of South Carolina approximately six miles north of Rock Hill and adjacent to Lake Wylie. The station center is located at latitude 35 degrees, 3 minutes, 5 seconds north and longitude 81 degrees, 4 minutes, 10 seconds west. The corresponding Universal Transverse Mercator Coordinates are E 493, 660 and N 3, 878, 558, zone 17.

4.2 Reactor Core

4.2.1 Fuel Assemblies

Insert

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either ZIRLO™ or Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material.* Limited substitutions of ZIRLO™, zirconium alloy, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Insert →

* A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium and boron carbide as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent or mixed oxide fuel assemblies having a maximum nominal fissile plutonium concentration up to 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

c. A nominal 13.5 inch center to center distance between fuel assemblies placed in the fuel storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 596 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1418 fuel assemblies.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report" (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis" (W Proprietary).

Insert 1.

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

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Ventilation Systems Heater Report

When a report is required by LCO 3.6.10, "Annulus Ventilation System (AVS)," LCO 3.7.10, "Control Room Area Ventilation System (CRAVS)," LCO 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)," LCO 3.7.13, "Fuel Handling Ventilation Exhaust System (FHVES)," or LCO 3.9.3, "Containment Penetrations," a report shall be submitted within the following 30 days. The report shall outline the reason for the inoperability and the planned actions to return the systems to OPERABLE status.

5.6.7 PAM Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Steam Generator Tube Inspection Report

a. The number of tubes plugged in each steam generator shall be reported to the NRC within 15 days following completion of the program;

(continued)

INSERT 1 to Catawba TS 5.6.5 b.

16. DPC-NE-1005P-A, "Duke Power Nuclear Design Methodology Using CASMO-4/ SIMULATE-3 MOX," (DPC Proprietary).

17. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," (Framatome ANP Proprietary).

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage rack (Ref. 1) is limited to a capacity of 1418 fuel assemblies. The spent fuel storage rack is designed to accommodate fuel with a maximum nominal enrichment of 5.0 wt% U-235 (maximum tolerance of ± 0.05 wt%) which have accumulated minimum burnups greater than or equal to the minimum qualifying burnups in Table 3.7.16-1. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Figure 3.7.16-1.

The storage rack can also accommodate mixed oxide fuel assemblies with a maximum nominal fissile plutonium concentration up to 4.15 weight percent (maximum tolerance of +/- 0.075 weight percent fissile Pu) and a maximum nominal Uranium-235 enrichment of 0.35 weight percent. The mixed oxide fuel assembly design is radially zoned with fuel rods at three different plutonium concentrations. The nominal fissile plutonium concentration limit is the weighted average for the entire fuel assembly.

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel storage racks is based on the use of unborated water, which maintains the spent fuel pool in a subcritical condition during normal operation when fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 2) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of a fuel assembly. This could potentially increase the reactivity of the spent fuel pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel pool storage rack with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1.

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Ref. 3). For these accident occurrences, the presence of soluble boron in the spent fuel pool (controlled by LCO 3.7.15, "Spent Fuel Pool Boron Concentration") prevents criticality in the spent fuel pool storage racks. By closely controlling the movement of each assembly and by checking the location

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BASES

APPLICABLE SAFETY ANALYSES (continued)

of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Table 3.7.16-1, in the accompanying LCO, ensures the k_{eff} of the spent fuel pool will always remain < 0.95, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Figure 3.7.16-1 and Table 3.7.16-2.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with the LCO, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the fuel assembly is in accordance with the configurations specified in the accompanying LCO.

~~This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with the configurations specified in the accompanying LCO.~~

Attachment 3

Description and Technical Justification

Attachment 3
Description and Technical Justification

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3. DESCRIPTION AND TECHNICAL JUSTIFICATION

3.1 INTRODUCTION

Duke Energy Corporation (Duke) is part of a consortium that has contracted with the Department of Energy to dispose of surplus weapon grade (WG) plutonium. Under this contract and options the consortium, Duke Cogema Stone & Webster (DCS), will provide for the design, construction, operation, and deactivation of a Mixed Oxide[1] (MOX) Fuel Fabrication Facility (MFFF). DCS will process PuO_2 powder supplied by the Department of Energy (DOE), blend it with depleted UO_2 powder, and fabricate it into MOX fuel pellets. These pellets will be loaded into MOX fuel assemblies. Following NRC approval of required license amendments, the fuel assemblies will be used in the McGuire and Catawba Nuclear Stations with core fractions up to 40% MOX fuel.

MOX fuel has been used extensively in European reactors for over 20 years. As a result, a large database of performance characteristics has been amassed by various European utilities. Currently, there are over 30 reactors that are using MOX fuel in Europe. The MOX fuel used in Europe is manufactured using the plutonium extracted from reprocessed low enriched uranium (LEU) fuel that has been discharged from European reactors. This fuel is typically referred to as reactor grade (RG) MOX fuel since the plutonium in the fuel is derived from the irradiation of uranium in a commercial reactor. RG MOX fuel is similar to the WG MOX fuel that Duke proposes to use. The primary differences between RG and WG MOX fuel are in the isotopic content of the plutonium and the marginally higher impurities contained in the fuel pellets. These differences are discussed in detail in Section 3.5.1 and in Reference 1.

In preparation for the utilization of batch quantities of MOX fuel, Duke currently plans to insert four MOX fuel lead assemblies into McGuire Unit 2 or Catawba Unit 1 during the Spring 2005 refueling outage. However, Duke is requesting and providing justification for insertion of lead assemblies into any of the Catawba or McGuire reactors. These reactors are similar enough that supporting safety analyses to use MOX fuel, in most instances, apply to any reactor at either station. Any differences are noted and described in more detail. By obtaining license amendments for Catawba and McGuire, Duke will have the flexibility to accommodate changes in lead assembly manufacturing schedules or other program changes without the need for additional license amendments.

The MOX fuel lead assemblies will be manufactured under the direction of Framatome ANP. Plans call for four lead assemblies to be irradiated for a minimum of two cycles to confirm acceptability of the planned MOX fuel assembly design, verify the validity of Duke's models to predict fuel assembly performance, and confirm the applicability of the European database to Duke's use of MOX fuel. Poolside post-irradiation examination (PIE) is planned to verify selected mechanical properties of the lead assemblies. In addition, some or all of the lead assemblies will undergo a third cycle of irradiation to assure that the lead assembly burnup

[1] The term "mixed oxide" refers to reactor fuel containing a mixture of plutonium and uranium oxides (PuO_2 and UO_2) with plutonium providing the primary fissile isotopes.

bounds the planned batch fuel burnup. Examination of one or more fuel rods in a hot cell is planned at the completion of the lead assembly irradiation program.

3.2 DESCRIPTION OF REQUESTED ACTION

Specific license amendments to the Catawba Nuclear Station, Unit 1 & 2 Facility Operating Licenses and Technical Specifications and to the McGuire Nuclear Station, Units 1 & 2 Facility Operating Licenses and Technical Specifications are requested to allow the insertion of a maximum of four MOX fuel lead assemblies. Typically, the insertion of one or more lead assemblies into a licensed reactor does not require Technical Specification (TS) changes or NRC approval provided the number of lead assemblies is limited and they are loaded into non-limiting core locations. However, due to several technical differences and specific wording in the Technical Specifications, the use of MOX fuel lead assemblies requires regulatory approval through the license amendment process under 10 CFR 50.90. Given the limited number of lead assemblies and the similarity of the MOX fuel assembly design to the current LEU fuel assemblies, Duke has concluded that there is no significant impact on plant safety and no discernible effect on reactor operation.

The MOX fuel lead assemblies each have a nominal average total plutonium concentration of 4.37 weight percent plutonium (w/o Pu). This concentration is attained through the use of a radially zoned fuel assembly made up of three different fuel rods containing pellets with nominally either 4.94, 3.35, or 2.40 w/o Pu. The weight percent refers to the total weight of plutonium (all isotopes) relative to the total weight of heavy metal (plutonium + uranium). The nominal average total plutonium concentration of 4.37 w/o corresponds to a nominal average fissile plutonium concentration of 4.06 w/o, assuming the plutonium isotopic vector shown in Table A3-2. This fuel design concept is described in more detail in Section 3.5.1 and in Reference 1.

3.3 DESCRIPTION OF McGUIRE TECHNICAL SPECIFICATION CHANGES

This section describes each of the proposed changes to the Technical Specifications for McGuire Nuclear Station. The use of MOX fuel lead assemblies necessitates revising Technical Specifications on spent fuel storage (including Bases), design features, and administrative controls. The proposed changes and technical justification for the changes are discussed in detail below.

3.3.1 McGuire Technical Specification 3.7.15 – Spent Fuel Assembly Storage

McGuire has two independent spent fuel pools (SFPs) for Units 1 and 2. The SFPs contain storage racks, which are located in two separate and distinct regions within the SFPs.

Region 1 is designed and generally reserved for temporary storage of new or partially irradiated fuel since the storage cell configuration represents a less reactive array than that in Region 2. The cells in Region 1 are spaced at 10.4 inches on center and were constructed

with a neutron absorbing material (Boraflex) attached to the exterior cell wall wrapper plate. The Boraflex material contains the isotope B^{10} as the primary neutron absorber. This region has the capacity to accommodate storage of a complete off load of the reactor core (193 fuel assemblies) coincident with a reload fuel batch.

Region 2 is designed and is generally used for normal long term storage of permanently discharged fuel that meets specified burnup and initial enrichment criteria. The storage cell configuration in this region represents a more reactive array than that of Region 1. This region has a closer center-to-center cell spacing of 9.125 inches. These cells also use the same Boraflex material, with somewhat less B^{10}, resulting in reduced neutron absorbing capability relative to the material in Region 1.

The Boraflex material in both Region 1 and Region 2 has experienced unexpected degradation. In order to maintain acceptable spent fuel storage limits in the face of this degradation, each region has been subdivided into 'A' and 'B' regions. In the 'A' regions partial credit is allowed for Boraflex, while in the 'B' regions no credit is taken for Boraflex. The regions are defined as follows:

Region 1A - Credit for 25% of the original Boraflex material is allowed
Region 1B – No credit for Boraflex is assumed
Region 2A – Credit for 40% of the original Boraflex material is allowed
Region 2B – No credit for Boraflex is assumed

Currently within each of these regions, low-enriched uranium (LEU) fuel assemblies are qualified as "Restricted," "Unrestricted," "Filler," or "Checkerboard," based on initial enrichment and burnup criteria. Using the same subcriticality requirements, an evaluation was performed to determine the acceptable storage configurations for MOX fuel assemblies in the McGuire SFPs. "Restricted" or "Checkerboard" storage configurations were defined for MOX fuel storage in each of the four regions. This evaluation shows that "Restricted" storage is allowed for higher reactivity MOX fuel when limited to a specified storage configuration with lower reactivity LEU fuel. "Checkerboard" storage is allowed for MOX fuel assemblies in a particular region, by surrounding or "checkerboarding" the MOX fuel with empty storage cells.

3.3.1.1 Description of McGuire Technical Specification 3.7.15 Changes

Fresh MOX fuel assemblies will be received, inspected, and loaded directly into the SFP for storage prior to insertion into the reactor. Currently, Limiting Condition for Operation (LCO) 3.7.15 specifies criteria for fuel storage by reference to tables and figures to determine allowable LEU fuel storage configurations. Revisions to this LCO and associated figures are proposed in this license amendment request (LAR) to also allow storage of MOX fuel assemblies in the McGuire SFPs. Fresh or irradiated MOX fuel may be stored as Restricted Fuel in Region 1A, and as Checkerboard Fuel in Regions 1B, 2A, and 2B. The descriptions of these fuel classifications in the pertinent TS 3.7.15 figures are revised to include MOX assemblies as qualifying fuel.

Marked up pages showing the proposed changes to the McGuire Technical Specifications and associated Bases are included in Attachment 1.

3.3.1.2 Justification for McGuire Technical Specification 3.7.15 Changes

An evaluation was performed that demonstrates the MOX fuel lead assemblies can be safely stored in the McGuire SFPs with no modifications to the existing storage racks. This evaluation confirmed that pertinent subcriticality criteria were met for storage of MOX fuel assemblies as "Restricted" or "Checkerboard" fuel in the different storage regions of the McGuire SFPs. In addition to normal storage conditions, several accident conditions were also analyzed, including a fuel assembly misload event, a dropped fuel assembly, abnormal SFP temperature changes, and a postulated heavy load drop (weir gate) onto the storage racks. The detailed criticality safety evaluation that supports the allowable MOX fuel storage configurations is contained in Appendix 3-1, "Criticality Evaluation of MOX Fuel Storage in the McGuire and Catawba Spent Fuel Pools." Additionally, the dose consequences of postulated MOX fuel handling accidents were evaluated and are presented in Section 3.7.3.

3.3.2 McGuire Technical Specification 4.2.1 - Fuel Assemblies

McGuire TS 4.2.1 currently allows fuel assemblies with "a matrix of Zircalloy (sic) fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material." The MOX fuel lead assemblies utilize fuel rods clad with M5™ which is a zirconium alloy with a different material specification than Zircaloy. Therefore, the McGuire Technical Specifications need to be revised to allow the insertion of fuel assemblies containing MOX fuel rods with M5™ cladding. Also, NRC regulations in 10 CFR 50.46 implicitly require the use of either Zircaloy or ZIRLO™ cladding, so that the use of M5™ clad fuel rods requires an exemption from the NRC regulations in addition to a license amendment. An exemption request pursuant to 10 CFR 50.12 to allow the use of M5™ clad fuel rods is included in this submittal as Attachment 6.

3.3.2.1 Description of McGuire Technical Specification 4.2.1 Changes

The proposed change to allow up to four MOX fuel lead assemblies to be inserted into either McGuire reactor consists of adding an asterisk to the second sentence in TS 4.2.1. To allow the use of MOX as a fuel material and M5™ as an alternative cladding type in the lead assemblies, a footnote is included that references this same sentence with an asterisk. The proposed footnote associated with the asterisk reads:

> *A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core.

The revised Technical Specification would permit the introduction of up to four MOX fuel lead assemblies with M5™ cladding into either of the McGuire reactors. As noted earlier, Duke currently plans to begin irradiation of four lead assemblies in either McGuire Unit 2 or Catawba Unit 1 in Spring 2005. However, the lead assembly manufacturing schedule could be revised to change the number of lead assemblies, the irradiation schedule, or the target reactor. Adding the proposed footnote to both the McGuire and Catawba Technical Specifications provides sufficient flexibility to accommodate a variety of contingencies.

3.3.2.2 Justification for McGuire Technical Specification 4.2.1 Changes

Section 3.5 contains a detailed description of the design of the MOX fuel rod and the mechanical design of the fuel assembly. Section 3.6 describes the operational implications of MOX fuel lead assemblies including core design and plant impacts. Section 3.7 contains the safety analysis for operating with MOX fuel lead assemblies, which includes evaluations of their behavior during transient and accident conditions and evaluations of dose consequences. All of these sections collectively provide the requisite justification for revising these technical specifications to include reference to the use of MOX fuel lead assemblies.

3.3.3 McGuire Technical Specification 4.3.1 - Criticality

This Technical Specification defines the criteria used to assure that the fuel storage racks in the spent fuel pool are designed to prevent criticality. These criteria include a limit on the enrichment of LEU fuel that can be stored in the fuel racks. However, there are no criteria related to the storage of MOX fuel. Therefore, to allow the storage of new and spent MOX fuel assemblies in these storage racks, this Technical Specification must be revised to include criteria for MOX fuel.

3.3.3.1 Description of McGuire TS 4.3.1 Change

This Technical Specification currently limits fuel storage in the spent fuel pool to LEU fuel assemblies with a "maximum nominal U-235 enrichment" of 4.75%. The proposed change

adds a MOX fuel limit to McGuire TS 4.3.1.1 a. The proposed Technical Specification is as follows (additional wording shown in italics):

"The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 4.75 weight percent, *or mixed oxide fuel assemblies with a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35% weight percent.*;"

3.3.3.2 Justification for McGuire Technical Specification 4.3.1 Change

The justification for changing McGuire TS 4.3.1.1 a. is the same as that for revising TS 3.7.15, which is provided in Section 3.3.1.2. This section includes reference to an evaluation (Appendix 3-1) that demonstrates the ability to safely store MOX fuel assemblies in the McGuire spent fuel pool. The evaluation includes criticality analyses for specific MOX fuel storage patterns, for a fuel assembly misload event, for a dropped fuel assembly, for abnormal SFP temperature changes, and for a postulated heavy load drop (weir gate) onto the storage racks. Since these analyses consider unirradiated MOX fuel assemblies with a maximum fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent, they also provide the necessary justification for the proposed changes to McGuire TS 4.3.1.1 a.

3.3.4 McGuire Technical Specification 5.6.5 – Core Operating Limits Report

Section 5.6.5 a. of the McGuire Technical Specifications contains a list of core operating limits that must be established for each reload cycle. These limits are calculated and documented in a cycle-specific Core Operating Limits Report (COLR). The methodologies used to develop the core operating limits listed in the COLR section of the Technical Specifications are described in either Duke or vendor topical reports that are approved by the NRC. Once these reports are approved, NRC requires that the reports be listed in Section 5.6.5 b. of the Technical Specifications for the plants to which the reports are applicable.

3.3.4.1 Description of TS 5.6.5 Change

The proposed change to TS 5.6.5 b. is to add the additional topical reports that Duke and Framatome ANP have submitted to NRC for approval that will be used to evaluate McGuire core designs containing MOX fuel lead assemblies. These topical reports are:

1) DPC-NE-1005P-A, "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," (DPC Proprietary).

2) BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," (Framatome ANP Proprietary).

3.3.4.2 Justification for McGuire Technical Specification 5.6.5 Changes

Both of these reports have been submitted to NRC for review and approval. NRC issuance of safety evaluation reports approving the use of the methodologies described in the topical reports will provide the necessary justification for changing this Technical Specification. Once NRC has completed their review and issued Safety Evaluation Reports approving the use of the methodologies described in these reports, it is an administrative change to include the approved reports in TS 5.6.5 b. The reports listed above and in the proposed Technical Specifications are identified as "approved" reports; *i.e.*, with an '-A' suffix even though the NRC review is not complete as of the submittal date of this license amendment request. Identifying these reports as approved is done anticipating that these reports will be approved as part of the overall process to allow insertion of MOX fuel lead assemblies into either of the McGuire reactors.

3.4 DESCRIPTION OF CATAWBA TECHNICAL SPECIFICATION CHANGES

This section describes each of the proposed changes to the Technical Specifications for Catawba Nuclear Station. The use of MOX fuel lead assemblies necessitates revising Technical Specifications on spent fuel storage, design features, and administrative controls. The proposed changes and technical justification for the changes are discussed in detail below.

3.4.1 Catawba Technical Specification 3.7.16 - Spent Fuel Assembly Storage

The Catawba SFPs are different from the McGuire pools in that they contain a single storage region with one storage rack design. All of the storage racks have the same cell center-to-center spacing (13.5 inches) and have no Boraflex panels. Currently, LEU fuel assemblies are qualified as "Restricted," "Unrestricted," or "Filler," based on initial enrichment and burnup criteria. "Restricted" storage allows storage of higher reactivity fuel when limited to a specified storage configuration with lower reactivity fuel (filler assemblies). Using the same subcriticality requirements, the criticality evaluation has determined an acceptable "Restricted" storage configuration for MOX fuel assemblies in the Catawba SFPs. In this evaluation "Restricted" storage is allowed for higher reactivity MOX fuel assemblies when limited to a specified storage configuration with lower reactivity LEU fuel.

3.4.1.1 Description of Catawba Technical Specification 3.7.16 Changes

Fresh MOX fuel assemblies will be received, inspected, and loaded directly into the SFP for storage prior to insertion into the reactor. Currently, LCO 3.7.16 specifies allowable LEU fuel storage configurations by reference to Table 3.7.16-1 and Figure 3.7.16-1. A revision to this LCO is proposed in this LAR to also allow storage of MOX fuel assemblies as Restricted

Fuel in the Catawba SFPs. The description of the Restricted Fuel classification is in Figure 3.7.16-1 which is revised to include MOX assemblies as qualifying fuel.

In addition, Surveillance Requirement (SR) 3.16.1 is revised since the current language refers to initial enrichment and burnup criteria, neither of which applies to MOX fuel storage. SR 3.7.16-1 currently reads:

> "Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with the specified configurations."

The intent of SR 3.7.16.1 is to verify that a fuel assembly meets the necessary criteria for storage in the spent fuel pool. The proposed change is to delete the current wording and insert the same language as contained in McGuire SR 3.7.15-1, which reads:

> "Verify by administrative means the planned spent fuel pool location is acceptable for the fuel assembly being stored."

The proposed change applies equally to a LEU or MOX fuel assembly and still requires verification that any fuel assembly meet the appropriate storage requirements identified in the associated LCO prior to moving it into the spent fuel.

Marked-up pages showing the proposed changes to the Catawba Technical Specifications and associated Bases are included in Attachment 2.

3.4.1.2 Justification for Catawba Technical Specification 3.7.16 Changes

An evaluation was performed that demonstrates the MOX fuel lead assemblies can be safely stored in the Catawba SFPs with no modifications to the existing storage racks. This evaluation confirmed that pertinent subcriticality criteria were met for storage of MOX fuel assemblies as Restricted Fuel in the Catawba SFPs. In addition to normal storage conditions, several accident conditions were also analyzed, including a fuel assembly misload event, a dropped fuel assembly, abnormal SFP temperature changes, and a postulated heavy load drop (weir gate) onto the storage racks. The detailed criticality evaluation that supports the allowable MOX fuel storage configurations is contained in Appendix 3-1, "Criticality Evaluation of MOX Fuel Storage in the McGuire and Catawba Spent Fuel Pools." Additionally, the dose consequences of postulated MOX fuel handling accidents were evaluated and are presented in Section 3.7.3.

3.4.2 Catawba Technical Specification 4.2.1 - Fuel Assemblies

Catawba TS 4.2.1 specifically allows fuel assemblies with "a matrix of either ZIRLO™ or Zircalloy (sic) fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material." The MOX fuel lead assemblies utilize fuel rods clad with M5™ which is a zirconium alloy with a different material specification than either Zircaloy or

ZIRLO™. Therefore, the Catawba Technical Specifications need to be revised to allow the insertion of fuel assemblies containing MOX fuel rods with M5™ cladding. Also, NRC regulations in 10 CFR 50.46 implicitly require the use of Zircaloy or ZIRLO™ cladding, so that the use of M5™ clad fuel rods requires an exemption from the NRC regulations in addition to a license amendment. An exemption request pursuant to 10 CFR 50.12 to allow the use of M5™ clad fuel rods is included in this submittal as Attachment 6.

3.4.2.1 Description of Catawba TS 4.2.1 Changes

The proposed change to allow up to four MOX fuel lead assemblies to be inserted into either Catawba reactor consists of adding an asterisk to the second sentence in TS 4.2.1. To allow the use of MOX as a fuel material and M5™ as an alternative cladding type in the lead assemblies, a footnote is included that references this same sentence with an asterisk. The proposed footnote associated with the asterisk reads:

> *A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core.

The revised Technical Specification and footnote would permit the introduction of MOX fuel lead assemblies with M5™ cladding into either of the Catawba reactors. As noted earlier, Duke currently plans to begin irradiation of four lead assemblies in either McGuire Unit 2 or Catawba Unit 1 in Spring 2005. However, the lead assembly manufacturing schedule could be revised to change the number of lead assemblies, the irradiation schedule, or the target reactor. Adding the proposed footnote to both the Catawba and McGuire Technical Specifications provides sufficient flexibility to accommodate a variety of contingencies.

3.4.2.2 Justification for Catawba Technical Specification 4.2.1 Changes

Section 3.5 contains a detailed description of the design of the MOX fuel rod and the mechanical design of the fuel assembly. Section 3.6 describes the operational implications of MOX fuel lead assemblies including core design and plant impacts. Section 3.7 contains the safety analysis for operating with MOX fuel lead assemblies, which includes evaluations of their behavior during transient and accident conditions and evaluations of dose consequences. All of these sections collectively provide the requisite justification for revising these technical specifications to include reference to the use of MOX fuel lead assemblies.

3.4.3 Catawba Technical Specification 4.3.1 – Criticality

Catawba Technical Specification 4.3.1.1 is similar to the McGuire Technical Specification and defines the criteria used to assure that the fuel storage racks in the Catawba spent fuel pool are designed to prevent criticality. TS 4.3.1.1 a. includes a limit on the enrichment of LEU fuel that can be stored in the fuel racks. However, there are no criteria related to the

storage of MOX fuel. Therefore, to allow the storage of new and spent MOX fuel lead assemblies in these storage racks, this Technical Specification must be revised to include criteria for MOX fuel.

3.4.3.1 Description of Catawba Technical Specification 4.3.1 Change

This Technical Specification currently limits fuel storage in the spent fuel pool to LEU fuel assemblies with a "maximum nominal U-235 enrichment" of 5.0 %. The proposed change adds a MOX fuel limit to Catawba TS 4.3.1.1 a. The proposed Technical Specification is as follows (new wording shown in italics):

> "The spent fuel storage racks are designed and shall be maintained with:
>
> a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent, *or MOX fuel assemblies having a maximum nominal fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent.*;"

3.4.3.2 Justification for Catawba Technical Specification 4.3.1 Change

The justification for revising Catawba TS 4.3.1.1 a. is the same as that for revising TS 3.7.16, which is provided in Section 3.4.1.2. This section includes reference to an evaluation (Appendix 3-1) that demonstrates the ability to safely store MOX fuel assemblies in the Catawba spent fuel pool. The evaluation includes criticality analyses for specific MOX fuel storage patterns, for a fuel assembly misload event, for a dropped fuel assembly, for abnormal SFP temperature changes, and for a postulated heavy load drop (weir gate) onto the storage racks. Since these analyses consider unirradiated MOX fuel with a maximum fissile plutonium concentration of 4.15 weight percent and a maximum nominal U-235 enrichment of 0.35 weight percent, they also provide the necessary justification for the proposed changes to Catawba TS 4.3.1.1 a.

3.4.4 Catawba Technical Specification 5.6.5 – Core Operating Limits Report

Section 5.6.5 a. of the Catawba Technical Specifications contains a list of core operating limits that must be established for each reload cycle. These limits are calculated and documented in a cycle-specific Core Operating Limits Report (COLR). The methodologies used to develop the core operating limits listed in the COLR section of the Technical Specifications are described in either Duke or vendor topical reports that are approved by the NRC. Once these reports are approved, NRC requires that the reports be listed in Section 5.6.5 b. of the Technical Specifications for the plants to which the reports are applicable.

3.4.4.1 Description of Catawba Technical Specification 5.6.5 Change

The proposed change to TS 5.6.5 b. is to add the additional topical reports that Duke and Framatome ANP have submitted to NRC for approval that will be used to evaluate Catawba core designs containing MOX fuel lead assemblies. These topical reports are:

1) DPC-NE-1005P-A, "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," (DPC Proprietary).
2) BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," (Framatome ANP Proprietary).

3.4.4.2 Justification for Catawba Technical Specification 5.6.5 Change

Both of these reports have been submitted to NRC for review and approval. NRC issuance of safety evaluation reports approving the use of the methodologies described in the topical reports will provide the necessary justification for changing this Technical Specification. Once NRC has completed their review and issued Safety Evaluation Reports approving the use of the methodologies described in these reports, it is an administrative change to include the approved reports in TS 5.6.5 b. The reports listed above and in the proposed Technical Specifications are identified as "approved" reports; *i.e.*, with an '-A' suffix even though the NRC review is not complete as of the submittal date of this license amendment request. Identifying these reports as approved is done anticipating that these reports will be approved as part of the overall process to allow insertion of MOX fuel lead assemblies into either of the Catawba reactors.

3.5 DESCRIPTION OF MOX FUEL LEAD ASSEMBLIES

The MOX fuel program will utilize the Framatome ANP Advanced Mark-BW fuel assembly, a fully qualified fuel assembly design that will be adapted for MOX application through changes to the fuel rod design. The Advanced Mark-BW fuel assembly is a standard lattice 17x17 fuel assembly specifically designed for use in Westinghouse-designed reactors. The Advanced Mark-BW adaptation for MOX applications, the Mark-BW/MOX1, is dimensionally and structurally identical to the Advanced Mark-BW with the only change appearing in the fuel rod internal design. The Advanced Mark-BW and the Mark-BW/MOX1 share the following base design features:

- Seated fuel rods,
- Floating intermediate spacer grids,
- Quick Disconnect removable top nozzle,
- High thermal performance spacer grids,
- TRAPPER ™ bottom nozzle[2],
- M5™ alloy[3] fuel rod cladding, guide thimbles and spacer grids,

[2] TRAPPER is a trademark of Framatome ANP, Inc.

- Mid-span mixing grids for enhanced thermal performance.

The MOX fuel lead assemblies will have these design features.

3.5.1 MOX Fuel and Fuel Rod Design Features

The fuel rod design consists of UO_2-PuO_2 (MOX) pellets contained in a seamless M5™ tube with M5™ end caps welded at each end. The design typically utilizes a 144.0 inch fuel stack height. The fuel pellets have a diameter of 0.3225 inches. The fuel rod cladding has a 0.374 inch outside diameter and a 0.0225 inch wall thickness. This configuration leaves a small clearance (approximately 0.003 inches radial clearance) between the inside diameter of the cladding and the outside diameter of the fuel pellets.

The fuel rod utilizes one stainless steel spring in the upper plenum to prevent the formation of fuel stack axial gaps during shipping and handling, while also allowing for the expansion of the fuel stack during operation. The fuel stack rests on the lower end cap. The lower end cap is made from M5™ and has a bullet nose shape to provide a smooth flow transition in addition to facilitating reinsertion of the rods into the assembly if any rods are removed after the assemblies have been irradiated (*e.g.*, during fuel examination programs). The upper end cap is also made of M5™ and has a grippable top hat shape that allows for the removal of the fuel rods from the fuel assembly if necessary. The upper end cap has a hole to permit evacuation and back-filling of the fuel rod with helium gas prior to re-sealing.

The fuel pellets are a sintered ceramic of high density UO_2-PuO_2. The UO_2 matrix is derived from depleted uranium enriched to nominally 0.25% U-235. The fuel pellets are cylindrically shaped with a dish at each end. The corners of the pellets have a chamfer that eases the loading of the pellets into the cladding. The dish and chamfer geometry also reduces the tendency for the pellets to assume an hourglass shape during operation. The design density of the pellets is 95% Theoretical Density (TD) with a maximum plutonium content of 6 weight percent. However, the maximum expected plutonium concentration in MOX fuel pellets for the lead assemblies is 4.94 weight percent.

The schematic diagram of Figure 3-1 shows an axial cross section of the MOX fuel rod for the Mark-BW/MOX1.

3.5.1.1 MOX/LEU Design Comparison

A comparison of typical fuel rod design details for the MOX and LEU fuel rod designs is summarized in Table 3-1. In addition to a different fuel pellet type, the MOX fuel rod design differs from the LEU fuel rod design in the areas of fuel rod length, design density, and maximum fuel rod burnup.

- Fuel Rod Length – The additional fission gas release from the MOX fuel is accommodated by increasing the fuel rod length, and thus the plenum volume. This

[3] M5 is a trademark of Framatome ANP, Inc.

increase in rod length can be incorporated in the MOX design while maintaining the required shoulder gap due to the lower burnup limit for the MOX design.

- Design Density – The design density for the MOX design is 95% TD whereas the LEU fuel rod design utilizes 96% TD pellets. The selection of 95% TD for the MOX was made to be consistent with previous European experience with reactor grade MOX fuel. Future increases in the design density will be evaluated for the Material Disposition Program when and if the European MOX designs evolve to higher densities.
- Maximum Fuel Rod Burnup – The objectives of the Material Disposition Program can be accommodated with a maximum fuel rod burnup that is less than that currently being used for LEU fuels. However, for the most efficient use of the MOX fuel, the design burnup may be increased as European experience at higher burnups provides the operational experience and data to justify the increase.

3.5.1.2 Pellet Microstructure

The plutonium fissile content – Pu-239 plus Pu-241 – of the WG MOX fuel is expected to be about 94%, whereas the RG MOX fuel is about 70%. Further, the RG material contains significantly higher concentrations of Pu-240, which acts as an absorber, reducing the reactivity of the RG material relative to the WG material. Thus, the plutonium concentrations for MOX fuel from the WG material must be reduced approximately 40% to maintain the same total reactivity as the MOX fuel made from RG material. This reduction in total plutonium concentration ensures that the macroscopic plutonium effects on fuel performance are bounded by the data from MOX fuel made from RG plutonium.

On a microscopic scale, the distribution of fissile material within the PuO_2-UO_2 matrix is controlled by the manufacturing process. In the MOX fuel fabrication process using RG material, a primary blend and micronization is performed with a UO_2/PuO_2 ratio of 70/30. This process step establishes the fissile content of the plutonium rich agglomerates. The micronized master blend is then diluted with UO_2 to reach the final plutonium concentration. Thus, the microstructure of the pellet from RG material consists of a uniform UO_2 matrix with uniformly distributed PuO_2-UO_2 agglomerates containing 30% PuO_2.

For the WG material the primary blend will be performed with a UO_2/PuO_2 ratio of 80/20. Using the same process as used with the RG material, this master mix is diluted with UO_2 to reach the final plutonium concentration. However, since the WG material has a relative 35% higher fissile content and significantly less Pu-240 parasitic material, the 80/20 master mix will produce plutonium rich agglomerates from the WG material that are equivalent in fissile content with the plutonium rich agglomerates produced from RG material using the 70/30 ratio. The resulting pellet microstructure for the MOX pellet from WG plutonium will be equivalent to the pellet microstructure of the MOX pellet made from RG material because:

- The UO_2 matrix that establishes the overall pellet microstructure is the same since the same process and the same feed UO_2 is used in both cases.

- The grain size, particle size, and particle distribution will be the same since the process is the same in terms of blender operation, size of sieves, pressing conditions, and sintering conditions.
- The distribution of fissile material will be the same since the particle size and distribution are the same, and the master mix adjustment has maintained the same fissile content of the plutonium rich agglomerates.

Thus, the fission density and the fission product inventory will be the same in both WG and RG MOX fuels. Since the two fuels are equivalent in fissile content and distribution of the fissile material, it can also be concluded that WG MOX fuel will behave the same as RG MOX fuel for considerations involving pellet thermal-mechanical behavior – fission gas release, transient response, and swelling.

The thermal conductivity of the WG MOX fuel will be lower than that of LEU fuel but bounded by that of the RG MOX fuel. Since the two materials have equivalent distributions of fissile material, and the WG material has lower total plutonium concentrations, the thermal conductivity of the WG MOX fuel will be less affected.

The fuel pellet radial power profile for WG MOX fuel will likewise be bounded by the RG MOX fuel performance. The distribution of fissile material is equivalent for the two materials, while the total plutonium concentrations are reduced for the WG MOX fuel.

3.5.2 MOX Fuel Assembly Mechanical Design Features

The Mark-BW/MOX1 fuel assembly is based on the Advanced Mark-BW, an improved 17x17 fuel assembly designed specifically for Westinghouse-designed PWRs and utilizing many proven features of the base Mark-BW design. The advanced design features have been demonstrated through a lead test assembly program at the North Anna reactors.

The Mark-BW/MOX1 and Advanced Mark-BW fuel assembly utilizes 11 spacer grids that, with the 24 guide thimbles, instrument tube, and top and bottom nozzles, provide the structural cage for the 264 fuel rods. The top and bottom end grids are made from Inconel 718 strip material. The six intermediate grids and three mid-span mixing grids are constructed from M5™ strip material. The intermediate grids are those between the end grids, not including the mid-span mixing grids. The M5™ clad fuel rods rest on the bottom nozzle and are laterally supported by the top and bottom end spacer grids and six intermediate spacer grids.

The Mark-BW/MOX1 and Advanced Mark-BW intermediate spacer grids are not mechanically attached to the guide thimbles, which allow the grids to "float" a very limited axial distance and thereby accommodate any axial differential growth between the fuel rods and guide thimbles. Ferrules around 8 of the 24 guide thimbles are designed to limit the axial displacement of the intermediate grids. The axial location of the spacer grids remains unchanged from previous Mark-BW designs. This arrangement reduces the axial forces on the guide thimbles and fuel rods, and the resultant forces on the spacer grids. In addition,

guide thimble axial loads are reduced, given that the weight of the fuel rods passes directly to the bottom nozzle.

The Mark-BW/MOX1 and Advanced Mark-BW spacer grid designs utilize hard/soft stops in the cells to support the fuel rod. Mark-BW/MOX1 and Advanced Mark-BW end and intermediate grids maintain the same grid periphery lead-in features as used in the Mark-BW design to ensure good fuel assembly-handling performance.

Features on the guide thimble assemblies constrain axial motion of the end grids. The bottom end grid is restrained through stainless steel sleeves that are welded to the bottom end grid. The grid sleeves are mechanically crimped to the guide thimble lower end plugs that are fixed to the bottom nozzle. Top end grid motion is restrained by stainless steel spacer sleeves that are welded to the top end grid and located on the guide thimbles between the bottom of the top nozzle and the top of the top end grid.

A quick disconnect mechanism is utilized on the Mark-BW/MOX1 and Advanced Mark-BW fuel assemblies. The attachments at the guide thimble/top nozzle interface allow the top nozzle to be removed for fuel assembly reconstitution. The Mark-BW leaf spring design, consisting of four sets of leaf springs made of Inconel 718 material, is also utilized on the Mark-BW/MOX1 and Advanced Mark-BW assemblies. Located in the top nozzle, the spring maintains positive fuel assembly contact with the core support structure under all normal operating conditions; it also maintains positive holddown margin for the fuel assembly hydraulic forces.

The bottom nozzle is the TRAPPER debris filter bottom nozzle, which is also used on the Mark-BW fuel assembly.

All key dimensions are maintained to ensure compatibility with existing interfaces. The dimensions presented are current values, which are subject to change for optimization as additional operating data are acquired while ensuring that all design bases are met.

3.5.3 Design Evaluation

The Mark-BW/MOX1 fuel assembly design meets all applicable criteria to maintain safe plant operation. The mechanical analysis demonstrates that the fuel assembly satisfies the requirements outlined in Section 4.2 in the SRP, NUREG-0800.

The design of the Mark-BW/MOX1 fuel assembly is such that it preserves the interface with resident fuel assemblies and all reactor internals and all equipment for normal handling. The Mark-BW/MOX1 is designed to preserve the original plant licensing bases for all reactor internal components.

The analyses performed in Framatome ANP topical report BAW-10239, *Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report* (Reference 2), are applicable to the Mark-BW/MOX1 fuel assembly except for those evaluations impacted by pellet characteristics. The fuel rod analyses follow the previously approved methods except that the fuel performance code COPERNIC (Reference 3) is used with MOX specific models. COPERNIC is used to provide pressures, oxide thickness and strains used in fuel rod mechanical analyses.

Mechanical and thermal analyses on the fuel rod design have been completed using preliminary fuel cycle information provided by Duke, and are summarized in Framatome topical report BAW-10238, *MOX Fuel Design Report* (Reference 1). Methods used are as specified in the COPERNIC fuel performance code topical report (Reference 3). Analyses will be redone if necessary when final fuel cycle design information is available. If rod design changes are necessary, revisions will meet the same criteria presented herein. This preliminary design is presented in Table 3-1, with a comparison to the Advanced Mark-BW (UO_2 design).

The use of M5™ has been evaluated by Framatome ANP as an advanced cladding and structural material and the results are documented in BAW-10227P-A (Reference 4).

3.5.4 Quality Assurance

Framatome ANP has the responsibility for the overall Quality Assurance (QA) oversight of the entire fuel assembly fabrication process, from the supply of government furnished PuO_2 powder to the delivery of the finished fuel assemblies to the reactor site. Every sub-vendor who operates under the technical requirements provided by Framatome ANP will be qualified by Framatome ANP as an approved supplier. Framatome ANP will verify that each of these vendors/facilities meets the requirements of 10 CFR 50 Appendix B. This verification may include Quality system audits by Framatome ANP, review of audits performed by other Framatome ANP facilities from other regions, and/or surveillance audits by other approved Framatome ANP quality auditors. Also as fuel assembly designer, Framatome ANP ultimately has the responsibility for certification of the finished fuel assemblies to Duke Power, through DCS.

3.6 EFFECTS OF MOX FUEL LEAD ASSEMBLIES ON PLANT OPERATION

The effects of four MOX fuel lead assemblies on various aspects of plant operation were evaluated. Areas specifically evaluated were reactor vessel irradiation, nuclear and thermal-hydraulic design effects, fuel handling, and plant security. The effects on operational transients were also evaluated as part of the safety analysis in Section 3.7.2.

3.6.1 Reactor Vessel

The primary plant component that is potentially impacted by the presence of MOX fuel lead assemblies is the reactor vessel. The specific concern is the aging effect due to increased neutron fluence on the beltline region of the reactor vessel; i.e. the reduction of fracture toughness of the reactor vessel due to neutron embrittlement. The fast neutron flux in a MOX fuel assembly is about 5% higher than a comparable LEU fuel assembly due to the fission yield characteristics of plutonium. However, this higher fast neutron flux in the MOX fuel lead assemblies does not translate to increased neutron embrittlement of the reactor vessel. There are two reasons for this. First, there are only four MOX fuel lead assemblies out of a total of 193 fuel assemblies in the reactor core. The overall neutron flux is dominated by the

189 LEU fuel assemblies. As a result, the core-wide fast neutron flux increase is less than 1%. Second, reactor vessel fluence is controlled primarily by the power in the fuel assemblies on the periphery of the core. Since, fresh MOX fuel lead assemblies will not be located on the core periphery, the fast flux impacting the reactor vessel will be virtually identical to that for an all-LEU core. In any event, the Reactor Vessel Integrity Program will manage reduction in fracture toughness of the reactor vessel beltline region so that the function of the vessel is maintained. The existing pressure-temperature curves in McGuire and Catawba Technical Specification 3.4.3 remain valid with four MOX fuel lead assemblies.

3.6.2 Nuclear Design

The primary active fuel material in MOX fuel is plutonium, which has different nuclear properties than conventional LEU fuel. However, even with these different nuclear properties, four MOX fuel lead assemblies have an insignificant effect on core wide behavior. Core performance is dominated by the nuclear properties of the remaining 189 assemblies in the core. A comparison of several key core wide physics parameters (critical boron concentration, control rod worths, moderator and fuel temperature coefficients) in a typical LEU core model with four MOX fuel assemblies showed that these physics parameters are very similar to those in a typical all-LEU core with no MOX fuel assemblies (see Tables 3-7 through 3-10).

The reload design process for a core with MOX fuel assemblies differs from the currently employed methods only in the use of the CASMO-4/SIMULATE-3 MOX code system, which is an update to the current CASMO-3/SIMULATE-3 code system. The CASMO-4/SIMULATE-3 MOX codes will be used to perform the required analyses of cycle-specific nuclear physics parameters and core transient behavior for both mixed LEU/MOX fuel cores and all-LEU fuel cores that are being performed with the current code system. Likewise, power distribution uncertainty factors, used to evaluate predicted fuel performance with respect to established peaking limits, were developed by benchmarking the CASMO-4/SIMULATE-3 MOX code system against partial MOX fuel cores, all-LEU cores, and critical experiments. Uncertainties were developed for both LEU and MOX fuel assemblies. The detailed nuclear design methodology is described in Reference 5 which was submitted in August 2001 for NRC review and approval.

3.6.3 Thermal-Hydraulic and Mechanical Design

The majority of the fuel assemblies (189 of 193) in the mixed core containing the MOX fuel lead assemblies will be the resident Westinghouse Robust Fuel Assemblies (RFAs). The basic thermal-hydraulic and mechanical design of the Mark-BW/MOX1 fuel assembly is similar to the RFA design and is well within the range of designs previously evaluated and employed in Duke's reactors. Duke and Framatome ANP will use current NRC-approved methodologies to analyze the thermal-hydraulic and mechanical design performance of these assemblies in order to determine appropriate limits for operation of the Mark-BW/MOX1 lead assemblies in the mixed core environment. Analyses will be performed using the VIPRE-01

computer code on McGuire and Catawba cores as described in Reference 6 with changes incorporated to model the Westinghouse RFA fuel in a mixed core as discussed in Reference 7. Mark-BW/MOX1 model information and statistical core design techniques used are described in Reference 8.

3.6.4 Fuel Handling

The fresh MOX fuel lead assemblies will arrive at McGuire or Catawba via the Department of Energy Safeguards Transportation System. The vehicles used in this system are enclosed tractor-trailer trucks with built-in safeguards. Unloading the fuel assemblies from an enclosed truck necessitates some differences in fuel handling procedures since current LEU fuel arrives on open flatbed trucks. Also, the shipping package for a fresh MOX fuel assembly will be a Type B(U) container, in compliance with 10 CFR Part 71 requirements for fresh fuel containing plutonium. The shipping package is expected to be a Cogema FS-65 package which is an end-loading design that holds one fuel assembly, as opposed to the usual clamshell arrangement used with LEU fresh fuel shipping packages, which hold two fuel assemblies.

The different package design requires some changes to the fresh fuel receiving procedures. An air pallet is used to remove the shipping package from the truck trailer. Special handling fixtures are used for lifting and uprighting the FS-65 package in order to position and restrain the package prior to opening. Special tooling is used to remove the end closure lid on the package. Existing plant cranes will be used to lift the package and handling fixtures. In addition, the existing new fuel handling tool used to grapple fresh LEU fuel assemblies will be used on the fresh MOX fuel lead assemblies. Once the fuel assembly is removed from the shipping package and inspected, it is placed in the Spent Fuel Pool where it remains until loaded into the reactor.

The presence of plutonium and americium in the fresh MOX fuel matrix results in a neutron and gamma dose on the order of two mrem/hr at one meter from any face of the fuel assembly. This increased dose rate will result in some revision to the new fuel receipt and inspection procedures in order to assure that these operations remains "as low as reasonably achievable" (ALARA); *i.e.*, minimize personnel radiation exposure consistent with the needs of the receipt and inspection procedures. Conservative estimates of the total dose associated with the receipt and inspection of one MOX fuel assembly range from .020 to .042 person-rem.

The consequences from dropping a fresh MOX fuel assembly are also greater than for current LEU fuel. Evaluation of the dose consequences for a postulated drop of a fresh MOX fuel assembly is contained in Section 3.7.3.

3.6.5 Security

The security program requirements for all Duke nuclear stations are contained in the Duke Power Company Nuclear Security and Contingency Plan. This plan will be revised to include

enhanced security requirements during receipt, handling, and storage of unirradiated MOX fuel assemblies. The specific changes to this plan will be submitted separately with the intent of having additional security measures and associated plan changes approved in the same time frame as the license amendments.

3.7 SAFETY ANALYSIS OF MOX FUEL LEAD ASSEMBLIES

The MOX fuel lead assemblies have slightly different nuclear and thermal-hydraulic characteristics from the resident Westinghouse LEU fuel assemblies. The effect of these differences on the design basis transients and accidents described in the UFSAR were evaluated to verify that acceptance criteria continued to be met for the MOX fuel lead assemblies.

3.7.1 Impact of MOX Fuel Lead Assemblies on Loss of Coolant Accident Analyses

The effects of MOX fuel lead assemblies on core operating and safety limits with respect to loss of coolant analyses (LOCA) were evaluated. With the conservative calculation approach described herein, there were no significant differences in the predicted performance of MOX fuel relative to LEU fuel for LOCA. This conclusion is based on an evaluation of MOX fuel with respect to isotopic content, decay heat, fuel material properties, and on representative LOCA calculations.

MOX fuel phenomena that have the potential to affect LOCA results are addressed in Section 3.7.1.1. Some adjustments to the Framatome ANP large break LOCA evaluation model are required to model MOX fuel. These adjustments are discussed Section 3.7.1.2. A limited set of large break LOCA calculations comparing MOX fuel lead assemblies to LEU fuel assemblies are summarized in Section 3.7.1.3. Section 3.7.1.4 contains a description of the set of MOX fuel lead assembly large break LOCA calculations that will be performed prior to operation with the lead assemblies. Section 3.7.1.5 addresses potential MOX fuel impacts on small break LOCA evaluations. Section 3.7.1.6 discusses potential mixed core loading effects for the MOX fuel lead assemblies.

3.7.1.1 MOX Fuel Phenomena and Lead Assembly Design Features that Potentially Affect LOCA

This section addresses the effects of the MOX fuel isotopics on LOCA performance. It is concluded that the changes in delayed neutron fraction and void reactivity feedback are not significant for the lead assemblies and the use of the LEU decay heat standard is shown to be conservative for application to MOX fuel.

3.7.1.1.1 Fissionable Isotope

The key difference between MOX fuel and LEU fuel is that Pu-239 is the predominant fissionable isotope in the MOX fuel. The substitution of a MOX fuel assembly for a LEU fuel assembly affects the assembly neutronic behavior, its neutronic interaction with the rest of the core, and the fission product concentrations. Neutronic interaction between MOX and LEU fuel assemblies occurs through the energy spectrum of the neutron flux. It is primarily embodied in a change of the delayed neutron fraction (β_{eff}), the void reactivity effect, and the prompt neutron lifetime. The Doppler reactivity effect between MOX and LEU fuel is similar and not of consequence in predicting the peak cladding temperature during a LOCA. The differing concentrations of fission products and nuclei activation alter the decay heat rate between MOX and LEU fuel pins. However, as discussed in Section 3.7.1.1.2, LEU fuel decay heat modeling required by current NRC regulations remains conservative for application to MOX fuel.

Delayed Neutron Fraction (β_{eff})
The fraction of delayed neutrons (β_{eff}) is lower in MOX fuel than in LEU fuel. As an example, the delayed neutron fraction for a 40 percent MOX fuel batch application will be reduced from around 0.0063 to about 0.0050 at beginning-of-life (BOL) conditions. This difference has two effects: (1) reactivity changes imposed on the core will produce a larger change in fission power, and (2) the neutron source for shutdown fission power will decrease. Both effects act to lower the power of the MOX fuel assembly relative to the LEU assembly during the transient.

Change in Void Reactivity Feedback
During LOCA, the void effect is responsible for achieving reactor shutdown and maintaining low fission powers in the unquenched regions of the core. Figure 3-2 provides a comparison of a void reactivity curve (effect on assembly k_∞) for a reference Framatome ANP designed LEU fuel assembly with a void reactivity curve calculated for a weapons grade MOX fuel assembly at the same conditions. A larger negative reactivity insertion occurs for the MOX fuel assembly than for the LEU assembly for all void fractions. This effectively suppresses the MOX fuel assembly power relative to the LEU assembly throughout a LOCA.

Prompt Neutron Lifetime
The prompt neutron lifetime decreases for MOX fuel cores. For a 40 percent MOX fuel batch application the lifetime can decrease by approximately 25 percent. This change will not affect LOCA calculations because the prompt neutron lifetime only becomes important for positive reactivity insertions greater than β_{eff}.

Use of Pre-LOCA Peaking throughout LOCA Simulation
The LEU fuel LOCA evaluation model assumes constant local peaking factors throughout the accident simulation. If k_∞ of any assembly does not monotonically decrease with increasing voiding, then local assembly peaking (assembly power relative to core average power) can increase during portions of the accident. This could increase the hot pin peaking factor for the fission component of the pin power and bring the assumption of constant peaking into

question. However, an examination of the void reactivity function for the plutonium concentrations anticipated for the lead assemblies, Figure 3-2, shows that the local k_∞ for both the MOX fuel and the LEU fuel assemblies is monotonically decreasing with increasing void fraction. Thus, the hot assembly (highest void fraction) power levels are continuously suppressed during the evolution of the accident and the application of the initial peaking factors is justified and conservative for MOX fuel as well as for LEU fuel.

Combined Effects on LOCA

Each of the neutronic effects identified as significantly differing between MOX fuel and LEU fuel results in a potential benefit in the MOX fuel parameter value over the corresponding LEU fuel value. Taken together these changes assure that the heat load within the MOX fuel lead assembly during LOCA will be lower than that in the resident LEU assembly. Thus, with all other processes being equal, core cooling mechanisms will more effectively control the cladding temperatures in the MOX fuel assembly than in the LEU fuel assemblies. The actual changes for the lead assemblies will not be significant because the effect of four assemblies on the core neutronic behavior will be limited and the MOX fuel assemblies will be substantially driven by the surrounding LEU fuel assemblies. Because the trend of the neutronic parameters is to the benefit of the MOX fuel assembly, it is conservative, as is done herein, to use LEU fuel neutronic parameter values in MOX fuel LOCA calculations.

3.7.1.1.2 Decay Heat

The fission product decay heat rate for MOX fuel assemblies, representative of the lead assembly design, was determined using the 1994 ANSI/ANS 5.1, "Decay Heat in Light Water Reactors." The actinide heat rate was determined using ORIGEN-S with the SAS2H procedures in the SCALE code system (Reference 9). The result, including the appropriate uncertainties, is that the sum of the decay heat and actinide heat for the lead assemblies, for fully saturated decay chains, falls substantially below that used for LEU fuel cores. Figure 3-3 shows a comparison of decay heat plus actinide heat for MOX fuel, the curve fit applied in the Framatome ANP evaluation model for LEU fuel, and the 1971 proposed ANS 5.1 Standard required by 10 CFR 50.46 Appendix K. The MOX fuel curve includes uncertainty factors sufficient to provide a 95 percent level of confidence that there is a 95 percent probability that the decay heat and the actinide heat are over-predicted. The Framatome ANP curve is a conservative fit to the 1971 proposed decay heat standard required by Appendix K. Both the Framatome ANP curve and the 1971 standard curve include a 20 percent increase in the decay heat and best-estimate actinide heat prediction.

The MOX fuel decay heat curve is consistently below the Framatome ANP LOCA evaluation model curve for the first 36,000 seconds (10 hours) and, except for times less than 0.1 seconds, consistently below the 1971 proposed ANS standard to 1,000 seconds. Beyond 1,000 seconds, there is no significant difference between the MOX fuel curve and the 1971 proposed standard. Integrating the decay and actinide powers, the total energy represented by the Framatome ANP curve up to the approximate time of peak cladding temperature, 150 to 400 seconds, averages more than 12 percent higher than the MOX fuel curve. Therefore, it is conservative to use the same decay and actinide heat rate for MOX fuel of the lead assembly

design as was approved for LEU fuel. No change to the evaluation model is required for MOX fuel decay heat effects.

3.7.1.1.3 Thermal and Mechanical Properties

The MOX fuel thermal-mechanical properties are very similar to those for LEU fuel. Six primary fuel properties are used in LOCA evaluations: thermal expansion, thermal conductivity, specific heat, emissivity, elastic modulus, and Poisson's ratio. The COPERNIC fuel rod performance code (Reference 3) differentiates a MOX fuel correlation only for thermal conductivity.[4] For each of these physical properties, the MOX LOCA evaluations will be conducted with close approximations over the LOCA temperature range to the appropriate COPERNIC correlation (MOX or LEU).

3.7.1.1.4 Steady State Fuel Temperature Prediction

The Framatome ANP LOCA evaluation model requires that the initial fuel temperature for a LOCA simulation be determined by a NRC-approved fuel performance code. For LEU fuel Framatome ANP has typically used the TACO3 code as discussed in References 10 and 11. However, COPERNIC, a more recent Framatome ANP code has models capable of predicting MOX and LEU fuel performance. Accordingly, Framatome ANP has applied COPERNIC for the determination of the steady state performance of the MOX fuel lead assemblies and for the initialization of comparison LEU fuel calculations. The following subsections discuss the changes to the LOCA evaluation model necessitated by the adoption of COPERNIC for LOCA initialization.

Transient Initialization

The main effect on LOCA evaluations due to the change from TACO3 to COPERNIC is that the improved fuel conductivity model alters the RELAP5 fuel-to-clad gap initialization. With TACO3, the RELAP5 gap model was initialized at steady state. Agreement with TACO3 initial volume-averaged fuel temperature predictions was achieved by adjusting the multipliers on the gaseous conductance term coefficient. Multiplier values varied from 0.8 to 2.0. Although the multipliers were retained throughout the transient, they did not impose a significant change in the gap coefficient. With COPERNIC, an adjustment to only the gaseous conductance would require larger multipliers than are deemed appropriate for application throughout the LOCA transient. An alternative approach was chosen for the MOX fuel lead assembly analyses, specifically to initialize RELAP5 with the COPERNIC fuel temperatures and gaseous conductance multipliers of 1.0. The core model will not be in steady state at transient initiation but the gap coefficient will be appropriate for use during the transient. The lack of a time zero steady state is not consequential because the cladding response to a LOCA is a rapid heatup during the first one or two seconds of the transient. This causes the cladding to pull away from the pellet. Under this condition, the gaseous

[4] COPERNIC has been approved by NRC for use with UO_2 fuel. NRC review of COPERNIC for application to MOX fuel is underway with approval expected by January 2003.

conductance is the only significant contributor to the gap coefficient. Thus, the approach improves the gap modeling for the LOCA transient relative to the current EM. A sensitivity study documented in Section 3.7.1.3 shows that the effect on peak cladding temperature of changing the gaseous conductance by a factor of 2.0 is small.

Initial Fuel Temperature Uncertainty
The use of COPERNIC for LOCA initialization necessitates a determination of the initial fuel temperature uncertainties to be applied to the average core, the hot assembly, and the hot pin. The measured-to-predicted distribution for COPERNIC, Reference 12, demonstrates that a constant temperature increment should be added to COPERNIC predictions to assure that 95 percent of the data are bounded with 95 percent confidence at high temperatures. Thus, the LOCA simulation for the hot pin should be initialized at the COPERNIC prediction plus the incremental adjustment. Assuming that the uncertainty distribution for COPERNIC is approximately normal, the relationships between the hot pin, the hot bundle, and the average core initial temperature predictions developed for TACO3 in Reference 13 (and approved by the NRC in Reference 14) remain reasonable for application to COPERNIC predictions. TACO3 applications required that 11.5 percent be added to the hot pin initial temperature to assure a 95/95 prediction and that 3.0 percent be added to the hot assembly to assure a 95/95 confidence. The corresponding temperature adjustments for core initialization with COPERNIC are: 1) no adjustment of the COPERNIC prediction for the average core, 2) the hot assembly predicted temperature is increased by 26 percent of the COPERNIC incremental adjustment, and 3) the hot pin temperature is increased by the full COPERNIC incremental adjustment.

3.7.1.1.5 Plutonium Concentration in Fuel Pins

A MOX fuel lead assembly contains three regions or zones of fuel pins, with each region having a different plutonium concentration. The differing plutonium concentrations will have an effect on the material properties of the pin, as described in Section 3.7.1.1.3. This effect is explicitly modeled in the analyses described in Section 3.7.1.3, and the results indicate that the effect is negligible.

3.7.1.2 Evaluation Model Adjustments Required for Lead Assembly LBLOCA Calculations

This section describes the changes made to the approved Framatome ANP LBLOCA evaluation model (References 4 and 15) for use in MOX lead assembly calculations. The changes described are directly related to MOX fuel effects.

3.7.1.2.1 Adjustments for COPERNIC

The technique for the lead assembly LBLOCA calculations is altered as a result of the use of COPERNIC to specify initial fuel conditions. The alteration involves the initialization of RELAP5 with COPERNIC initial fuel temperatures without adjusting the fuel-to-clad gap

coefficient to produce a thermal steady state condition. The fuel is in a transient condition at the start of the LOCA simulation. As discussed in Section 3.7.1.1.4, this approach offers the benefit of preserving the gaseous conductance term of the fuel-to-clad gap coefficient throughout the transient. Additionally, the initial fuel temperature uncertainty adjustments were altered as described in Section 3.7.1.1.4 to reflect the measured-to-predicted distribution from the COPERNIC benchmarks.

3.7.1.2.2 Adjustments for MOX Fuel Physical Properties

The approved evaluation model uses fuel materials properties characteristic of LEU fuel. The evaluation for the MOX fuel lead assemblies uses fuel materials properties based upon the COPERNIC code, which is under review for application to MOX fuel. Although these properties do not differ substantially between MOX and LEU fuel, the thermal conductivity correlation within COPERNIC (for LEU fuel or MOX fuel) is improved over the conductivity modeling previously incorporated in Framatome ANP evaluation models.

3.7.1.2.3 Rupture Modeling for Mid-Span Mixing Grids

This section describes how the approved fuel pin rupture model will be applied to fuel assemblies incorporating mid-span mixing grids (MSMGs - non-structural grids centered between structural grids). For the purpose of determining bundle blockage characteristics following cladding rupture, the Framatome ANP LOCA evaluation model assumes that the incidence of rupture is distributed throughout the upper two-thirds of the structural grid span within which rupture is calculated. For cores containing fuel assemblies with MSMGs, the modeling assumption is that the rupture density at the location of maximum blockage is not altered from that of a core containing no fuel assemblies with MSMGs. Rupture cooling is modeled in the hot assembly at only one elevation for cores with either type of grid configuration.

3.7.1.3 Representative LBLOCA Calculations

To provide validation of the expected LOCA results for the MOX fuel lead assemblies, a set of large break LOCA comparison cases for LEU and MOX fuel assemblies, both of the lead assembly design, were run. All cases simulated a full double-ended guillotine break at the cold leg pump discharge with a C_D of 1.0 and an initial power distribution peaked toward the core outlet (10.3-ft elevation). All cases incorporated the evaluation model adjustments described in Section 3.7.1.2, except as noted below for Case 2. The three cases are described below.

Case 1: MOX fuel base case with nominal gap conductance (See Section 3.7.1.2.1).
Case 2: MOX fuel case with 2.0 multiplier on nominal gap conductance.
Case 3: LEU fuel case otherwise identical to Case 1.

These calculations demonstrated that no significant difference exists between the two fuel types.

Table 3-2 lists the plant parameters and their values used in the calculations. As indicated in this table, the MOX fuel lead assemblies were held to a total peaking limit (F_Q) of 2.4, four percent lower than the limit for the resident LEU fuel. A sequence of events for Case 1, the base MOX fuel lead assembly calculation, is provided in Table 3-3. Table 3-4 shows the results for fuel pins of three differing plutonium concentrations representative of the MOX fuel lead assemblies.

Table 3-5 compares the base MOX fuel evaluation case (Case 1) with the same MOX fuel assembly initialized with a fuel-to-clad gaseous conductance coefficient multiplier of 2.0 (Case 2). Increasing the clad-pellet gaseous conductance coefficient to twice its value approximates the type of core initialization that is used when the initial fuel temperature is obtained from TACO3. The peak cladding temperature changes by about 13 degrees F. The comparison of the MOX fuel (Case 1) and the LEU fuel (Case 3) results show a difference of 37 degrees F. This is expected, given the relatively minor differences in the modeling of the two fuel types.

Figures 3-4 and 3-5 provide information about the evaluation model and the input models for these calculations. Figures 3-6 through 3-11 provide the time dependence for important LOCA parameters based on Case 1. Note that there are no essential differences in calculation results between LEU fuel and MOX fuel with the modeling assumptions and conservatisms used.

The conclusion from these comparison calculations is that:

1) The calculated LOCA performance of MOX fuel and LEU fuel is substantially unaffected by the difference in the fissionable isotope even when no credit is taken for the expected reduction in decay heat in MOX fuel,

2) The impact of the EM core initialization technique, removal of the forced thermal steady state requirement, is small, and

3) The effect of different plutonium concentrations on peak cladding temperature (PCT) is insignificant and need not be specifically modeled.

3.7.1.4 LBLOCA Analytical Basis for Operation

The LOCA analytical basis for operation of the lead assemblies will be developed during 2002 and early 2003. It is expected that the results will validate the allowed peaking employed in the sample calculations as shown in Table 3-2. The following calculations will be performed to validate lead assembly operability.

1) Time-in-Life (Burnup) Sensitivity Study to 60 GWd/MThm (assembly burnup)

2) Steam Generator Design Effects Study (Three of the four McGuire/Catawba units have replacement steam generators of slightly altered design and lower tube plugging.)

3) Power Distribution (LOCA Limits) Study to Validate K_Z

These calculations will employ the model adjustments as described in Section 3.7.1.2.

3.7.1.5 Small Break LOCA (SBLOCA) Evaluation

The primary SBLOCA issue is determining the core mixture level as a function of time. After such a determination is made, steam production below the mixture level is used with convection-to-steam and radiation heat transfer models to determine cladding temperatures above the mixture level. For the MOX fuel lead assembly core, the resident fuel assemblies dominate the core mixture level prediction, and the existing licensing calculations are applicable to the lead assemblies. Steam is rapidly diverted from the hot assembly to the average core to achieve a relatively uniform steam velocity across the core. Hence, the steam flow in the hot assembly at the location of the hot spot is characteristic of the average core flow and is essentially independent of the hot bundle power or configuration. Therefore, so long as the surface area for heat transfer or other local film coefficient effects are not altered, there will be no effect on the predicted cladding temperature between the lead assemblies and the resident LEU fuel assemblies. The lead assemblies have the same heat transfer surface area as the resident assemblies. The allowed local power of each MOX fuel lead assembly will not exceed that allowed for the resident fuel assemblies. Therefore, the calculated peak cladding temperatures for the lead assemblies will be less than those calculated for the resident fuel assemblies and it is appropriate for the lead assemblies to use the existing SBLOCA evaluation as their licensing basis.

3.7.1.6 Mixed Core Loading Effects

The MOX fuel lead assemblies will reside within a core of Westinghouse LEU fuel assemblies. The lead assemblies will be surrounded by resident LEU fuel assemblies having the same physical dimensions and very similar hydraulic characteristics. The MOX fuel lead assembly design employs MSMGs and the resident fuel design uses intermediate flow mixing grids (IFMs). The design of these mixing grids is such that the MOX fuel lead assembly pressure drop is less than four percent lower than the pressure drop for a resident Westinghouse fuel assembly at design flow rates. Hence, flow diversion favoring one fuel assembly at the expense of the other design is expected to be inconsequential. Therefore, there will be no mixed core impact on the LOCA performance of the resident Westinghouse assemblies. The complete set of lead assembly LOCA calculations will be done with the average core modeled to simulate the hydraulic performance of the resident assemblies, providing a direct evaluation of the resident fuel effects on the MOX fuel lead assemblies.

3.7.1.7 Conclusions

There are no significant differences in calculated LOCA performance between LEU and MOX fuel with the modeling assumptions and conservatisms selected. No adverse consequences due to the presence of four MOX fuel lead assemblies in the resident core of LEU fuel assemblies are expected. Therefore, during a postulated LOCA, the MOX fuel lead assemblies behave essentially the same as the resident LEU fuel assemblies and the calculations for the resident assemblies can be applied to the lead assemblies. However, the resident LEU fuel assemblies rely on a best estimate LOCA model as the licensing basis, and the calculations described herein were performed with a deterministic model. To reconcile this difference, the 95/95 bounding LOCA results for the resident assemblies are compared to the lead assembly representative results in Table 3-6. This table will be reconstructed when the final licensing basis calculations are performed. The differences between the calculation approaches and the assembly designs are identified within the table. These differences can, if necessary, be applied to future resident assembly calculations to establish the expected impact on the lead assemblies. This eliminates the need to perform calculations on both resident LEU fuel assemblies and the MOX fuel lead assemblies in the event that revised LOCA calculations are needed. If the need for recalculation specifically concerns the performance of the lead assemblies, specific lead assembly calculations will be made with the models described herein and the relationship between the resident fuel and MOX fuel lead assembly LOCA results reestablished.

3.7.2 Impacts of MOX Fuel Lead Assemblies on Non-LOCA Analyses

All of the non-LOCA transients and accident analyses described in Chapter 15 of the McGuire and Catawba UFSARs were reviewed to determine the impact of MOX fuel lead assemblies on the results and to verify that acceptance criteria continue to be met. In addition, the mass and energy release analyses in Chapter 6 of the UFSAR were also reviewed for any effect due to MOX fuel. Potential effects due to fuel assembly design differences are addressed in Section 3.7.2.2. The evaluation of MOX fuel effects resulting from changes in core average physics parameters is provided in Section 3.7.2.3. Some design bases transients and accidents are potentially sensitive to local physics parameters, and those are evaluated in Section 3.7.2.4. Potential decay heat effects are addressed in Section 3.7.2.5.

3.7.2.1 Transients and Accidents Evaluated

The transients and accidents evaluated and the associated UFSAR sections are listed below.

1) Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (6.2.1.3)
2) Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures inside Containment (6.2.1.4)

3) Feedwater System Malfunctions that result in a Reduction in Feedwater Temperature (15.1.1)
4) Feedwater System Malfunction Causing an Increase in Feedwater Flow (15.1.2)
5) Excessive Increase in Secondary Steam Flow (15.1.3)
6) Inadvertent Opening of a Steam Generator Relief or Safety Valve (15.1.4)
7) Steam System Piping Failure (15.1.5)
8) Steam Pressure Regulator Malfunction or Failure that Results in
9) Decreasing Steam Flow (15.2.1)
10) Loss of External Load (15.2.2)
11) Turbine Trip (15.2.3)
12) Inadvertent Closure of Main Steam Isolation Valves (15.2.4)
13) Loss of Condenser Vacuum and Other Events Causing a Turbine Trip (15.2.5)
14) Loss of Non-Emergency AC Power to the Station Auxiliaries (15.2.6)
15) Loss of Normal Feedwater Flow (15.2.7)
16) Feedwater System Pipe Break (15.2.8)
17) Partial Loss of Forced Reactor Coolant Flow (15.3.1)
18) Complete Loss of Forced Reactor Coolant Flow (15.3.2)
19) Reactor Coolant Pump Shaft Seizure (Locked Rotor) (15.3.3)
20) Reactor Coolant Pump Shaft Break (15.3.4)
21) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition (15.4.1)
22) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)
23) Rod Cluster Control Assembly (RCCA) Misoperation (System Malfunction or Operator Error) (15.4.3)
24) Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (15.4.4)
25) Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (15.4.6)
26) Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7)
27) Spectrum of Rod Cluster Control Assembly Ejection Accidents (15.4.8)
28) Inadvertent Operation of Emergency Core Cooling System during Power Operation (15.5.1)
29) Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (15.5.2)
30) Inadvertent Opening of a Pressurizer Safety or Relief Valve (15.6.1)
31) Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment (15.6.2)
32) Steam Generator Tube Failure (15.6.3)
33) Anticipated Transients without Trip (15.8)

3.7.2.2 Thermal-Hydraulic Differences

The MOX fuel lead assemblies will be co-located with Westinghouse Robust Fuel Assembly (RFA) design resident fuel, which has Intermediate Flow Mixing grids (IFMs). The MOX fuel lead assembly design (Mark-BW/MOX1) is a Mark-BW design fuel assembly with Mid-

Span Mixing Grids (MSMGs). The MSMGs are included in the design to improve thermal performance and make the thermal-hydraulic design compatible with the resident RFA design. As a result, overall fuel assembly pressure drop for each of the fuel designs is within about 4%.

For a McGuire or Catawba core with both Mark-BW/MOX1 lead assemblies and RFA fuel assemblies (referred to as a mixed core) the impacts of differences in hydraulic design and mixing vane grid performance of the two fuel assembly types are explicitly analyzed. As a result, each fuel type has specific limits that include the effects of flow variations as well as fuel assembly feature performance.

The behavior of the minimum departure from nucleate boiling ratio (DNBR) is calculated for a limiting mixed core of Mark-BW/MOX1 and RFA fuel assemblies. The limits derived from this calculation are applied to the Mark-BW/MOX1 lead assemblies to ensure DNBR criteria are met.

3.7.2.3 Comparison of Core Average Physics Parameters

The addition of four MOX fuel lead assemblies to an otherwise all-LEU core has no significant impact on the core average physics parameters shown in Tables 3-7 through 3-10 for a typical McGuire/Catawba reactor core. These tables summarize the differences in various core physics parameters between two representative core models. One core model (designated MOX in the tables) had four MOX fuel assemblies in locations typical of the planned lead assembly core. The second core model (designated LEU in the tables) had all LEU fuel assemblies. In the second core model the four MOX fuel assembly locations were replaced with four LEU fuel assemblies that were chosen so that the boron letdown and assembly powers were as close as possible to the first core model with the four MOX fuel assemblies. Depletion simulations were then run on both core models and the core physics parameters calculated at various effective full power days during the simulation runs. The comparisons in Tables 3-7 through 3-10 demonstrate that the presence of four MOX fuel assemblies in an otherwise all-LEU core does not produce a significant change in any of these core physics parameters.

In the first cycle of operation the four MOX fuel assemblies will be placed in unrodded symmetric core locations. The planned core design is an In-In-Out checkerboard reload pattern similar to that used in previous cycles. The reload value for each physics parameter used in the safety analysis and maneuvering analysis will be confirmed to be within the reference values previously calculated as described in References 16 and 17. If any of the reload values fall outside the reference values, the core design or safety limits will be modified and/or changes made to the core operating limits as allowed in the Core Operating Limits Report.

3.7.2.4 Comparison of Local Core Physics Parameters

The transients and accidents that are sensitive to local physics parameters include:

1) Control rod ejection,
2) Rod Cluster Control Assembly (RCCA) misoperation (withdrawal/drop),
3) Steam system piping failure, and
4) Fuel assembly misloading.

In the first cycle of operation MOX fuel assemblies will be unrodded and located away from fuel assemblies having significant ejected control rod worth. Thus the power increase that would occur in a MOX fuel lead assembly location during a rod ejection accident is substantially less than the power increase that would occur in LEU fuel located in the vicinity of the highest worth ejected control rod. Maintaining key core parameters within present reference values insures that both core wide and localized responses to a rod ejection in a core with MOX fuel lead assemblies are no more limiting than for a core containing only LEU fuel assemblies.

A control rod ejection simulation was performed with four MOX fuel assemblies placed in their most likely locations in a representative core. This analysis was performed with SIMULATE-3K MOX (Reference 5) and included appropriate conservatisms on ejected control rod worth, delayed neutron fraction, fuel temperature coefficient, moderator temperature coefficient, control rod trip worth, and trip delay time. The calculated peak enthalpy in the core under EOC HZP conditions was 54 calories per gram and occurred in a LEU fuel assembly located face adjacent to the ejected control rod location. The peak enthalpy predicted in a MOX fuel lead assembly was 30 calories per gram. Therefore, for the core design contemplated for the MOX fuel lead assemblies, the control rod ejection accident calculation results are benign relative to current regulatory acceptance criteria for LEU fuel. Furthermore, the conservatively-calculated MOX fuel energy deposition values are well below values at which cladding failure has been observed in CABRI reactivity insertion accident tests involving MOX fuel. It can be concluded that four MOX fuel lead assemblies can be used without presenting an undue risk to the health and safety of the public due to postulated reactivity insertion events.

Single control rod withdrawal and control rod drop events are not expected to be impacted by the introduction of four MOX fuel lead assemblies. As previously noted, the MOX fuel lead assemblies will not be placed under control rods in the first cycle of operation. For later cycles the assembly reactivity and rod worth for any control rod inserted in a MOX fuel assembly will be insignificant. Therefore, the MOX fuel assemblies will not be in the limiting core locations for a single withdrawal or drop. The reload values for the control rod worths will be within the reference values contained in the safety analysis. Results are expected to be no different than for any typical core reload.

Steam system piping failure with the most reactive rod stuck will not be impacted. The introduction of the four MOX fuel lead assemblies in unrodded locations will not significantly alter the rod worth of the most reactive rod. The core design will control the worth of the

most reactive rod and the target value for the reload will be less than the reference value contained in safety analysis such that the current acceptance criteria for this accident will be met.

The analysis of operation with a misloaded fuel assembly will not be significantly impacted by the introduction of four MOX fuel lead assemblies. Administrative measures are used to ensure against misloaded fuel assemblies, and additional assurance of detecting misloading is provided by core power distribution measurements during plant startup. The administrative measures are equally effective for MOX fuel. MOX fuel assemblies have much lower thermal neutron flux than LEU fuel assemblies for the same power level. Therefore, a MOX fuel assembly misloaded into a LEU location (or vice versa) would be even more apparent from a core flux map than a misloaded LEU assembly in a LEU location. Finally, the planned reactivity for the MOX fuel assemblies was chosen to be similar to the reactivity of the co-resident LEU assemblies. Accordingly, the equally reactive MOX assemblies would have no more of an impact if misloaded than a similar misloaded LEU fuel assembly. In addition MOX fuel assemblies have about one half of the thermal flux of a LEU assembly. As a result, a misloaded fuel assembly would be readily detected, given that the incore detector signal for a LEU assembly loaded in a MOX fuel location would be much higher than the expected signal for the MOX fuel assembly. Therefore, given that MOX and LEU fuel assemblies are equally reactive and misloading of a MOX fuel assembly is readily detectable, the analysis of a misloaded fuel assembly will not be adversely impacted by the use of MOX fuel lead assemblies in the core.

3.7.2.5 Decay Heat

MOX fuel and LEU fuel differ in their decay heat levels. Analyses of MOX fuel and LEU fuel decay heat levels have shown that LEU fuel has a slightly higher decay heat level immediately after shutdown and for several days after shutdown as shown in Figure 3-12. Therefore, assuming LEU fuel decay heat levels for all fuel in the core is conservative for those analyses where higher decay heat levels are more limiting for several days after shutdown. Furthermore, the four MOX fuel lead assemblies will have no significant impact on the post-shutdown decay heat levels in the core because (i) the difference in decay heat level is small and (ii) four MOX fuel assemblies comprise only about 2% of the 193 assemblies in the core. Therefore, the differing decay heat level of the MOX fuel lead assemblies will have no adverse impact on analysis results of UFSAR transients and accidents.

3.7.2.6 Conclusions

All of the non-LOCA transients and accidents contained in Chapters 6 and 15 of the McGuire and Catawba UFSARs were evaluated in this section to determine the impact of the MOX fuel lead assemblies. Conclusions are as follows:

1) There are no significant differences in the thermal-hydraulic parameters for the MOX fuel lead assembly fuel design. The impacts of the differences in hydraulic design and mixing vane grid performance of the two fuel assembly types are explicitly analyzed. No adverse impact is expected on any of the Chapter 6 or 15 analyses listed in Section 3.7.2.1 due to thermal-hydraulic differences in the MOX fuel design.

2) There is no significant impact expected on the core average physics parameters. Therefore, there will be no impact on Chapter 6 and 15 analyses listed in Section 3.7.2.1 due to changes in core average physics parameters by the introduction of MOX fuel lead assemblies. This will be confirmed as a part of the normal reload design and safety review process for the core in which the MOX fuel lead assemblies are ultimately used.

3) Changes in the local physics parameters due to the presence of four MOX fuel lead assemblies are small and will have no adverse impact on any of the analyzed UFSAR transients or accidents.

4) The short term decay heat level of the MOX fuel lead assemblies is less than comparable LEU fuel assemblies and will have no adverse impact on the UFSAR transients and accidents.

The overall conclusion of this evaluation is that the introduction of MOX fuel lead assemblies into any of the McGuire or Catawba reactor cores will have no significant impact on the non-LOCA UFSAR transients and accidents listed in Section 3.7.2.1.

3.7.3 Radiological Consequences of Postulated Accidents

The differences in radiological consequences for selected design basis accidents (DBAs) for an all-LEU core and a core containing four MOX fuel lead assemblies were evaluated for Catawba Nuclear Station. Radiation doses for a number of scenarios associated with these DBAs were calculated to determine the difference in results between MOX and LEU fuel. The relative differences in radiation doses for the same accidents would be the same for McGuire Nuclear Station.

Primarily, the use of MOX fuel lead assemblies has the potential to affect the dose consequences for the following DBAs:

1) Fuel handling accidents (FHA) in containment,
2) FHA in the fuel building, and
3) Weir gate drop.

These DBAs are important since the dose consequences are independent of the number of MOX fuel assemblies; *i.e.*, the results are applicable whether there are only a few MOX fuel assemblies or many fuel assemblies. Source terms and radioactive releases to the environment are postulated for several other DBAs (rod ejection accident, locked rotor accident, loss of coolant accident). However, the relative contribution of the four MOX fuel

lead assemblies to the source term will be significantly smaller for these DBA's compared to the FHAs and weir gate drop accident. The FHAs and weir gate drop accident are the limiting DBAs in terms of evaluating the relative difference in post-accident radiation doses between MOX and LEU fuel. Therefore, radiation doses were specifically calculated for FHAs and weir gate drop accidents and the results compared for LEU fuel and MOX fuel. The analyses were conducted in conformance with the regulatory positions of Regulatory Guide 1.25 and the guidelines in Standard Review Plans (SRPs) 15.7.4, 9.4.1, and 9.4.2.

The fission product isotopic content of a MOX fuel assembly differs somewhat from the fission product isotopic content of a comparable LEU fuel assembly. Fission product isotopic contents were calculated for (i) a MOX fuel assembly with a nominal plutonium concentration of 4.4 w/o and (ii) a LEU fuel assembly with an enrichment of 4.25%. Calculated thyroid radiation doses using this MOX fuel source term were compared to thyroid doses calculated with the LEU fuel source term. The calculated doses were about 3% higher using the MOX fuel source term, primarily due to the increase in the I-131 inventory. Specifics of the calculations for each accident are described in the following paragraphs.

3.7.3.1 Fuel Handling Accident

Radiation doses were calculated for the following FHA scenarios:

1) FHA in containment, offsite power available, failure of the operating train of the Control Room Area Ventilation System (CRAVS).
2) FHA in the fuel building, failure of the operating train of the Fuel Handling Ventilation Exhaust System (FHVES).
3) FHA in the fuel building, offsite power available, failure of the operating CRAVS train.

Two sets of radiation doses were calculated for each scenario. One calculation assumed that damage occurred in LEU fuel assemblies. The second calculation assumed that damage occurred in MOX fuel assemblies. The limiting difference in radiation doses was determined to be due to the difference in thyroid radiation doses. The thyroid radiation doses for the MOX fuel cases were calculated to be about 3% greater than the LEU fuel cases for these FHA scenarios. Radiation doses for all MOX and LEU scenarios were within the guideline values in SRPs 15.7.4 and 6.4.

3.7.3.2 Weir Gate Drop Accident

For the weir gate drop accident comparison, seven MOX fuel assemblies of the same type were assumed to be damaged. This conservative assumption was made even though only four MOX fuel lead assemblies are present. This assumption also allows for a direct comparison to the weir gate drop accident involving only LEU fuel assemblies since this analysis assumes that seven fuel assemblies are damaged. Radiation doses were calculated for the following scenarios:

1) Weir gate drop, failure of the operating FHVES train.
2) Weir gate drop, offsite power available, failure of the operating CRAVS train.

As for the FHAs, two sets of radiation doses were calculated for each weir gate drop scenario; one assuming the damage occurred in LEU fuel assemblies and the other assuming the damage occurred to fuel assemblies containing MOX fuel. As was the case for the FHA scenarios, the limiting difference in radiation doses between MOX fuel and LEU fuel for the weir gate drop scenarios was due to the difference in thyroid radiation doses. The thyroid radiation doses for the MOX fuel cases were calculated to be about 3% higher than the LEU fuel cases for the weir gate drop scenarios. The calculated radiation doses for all scenarios were within the guideline values of SRPs 15.7.4 and 6.4.

3.7.3.3 Other Design Basis Accidents

The effects of MOX fuel on thyroid radiation doses for other DBAs will be significantly less than the effect on FHAs and weir gate drop accidents. The relative contribution of the MOX fuel assemblies to the source term for other DBAs is much smaller since the thyroid radiation dose increase is generally proportional to the increase in I-131 inventory and to the number of MOX fuel assemblies in the core. Since the MOX fuel lead assemblies will constitute only four out of a total of 193 fuel assemblies in the core, the effect of the small increase in the source term for a particular isotope in a MOX fuel assembly would have a negligible impact on DBAs that use a core-wide source term.

Insertion of MOX fuel lead assemblies would have a small effect on thyroid radiation doses for DNB-limited accidents, which involve core-wide fuel failure assumptions. These include the locked rotor accident, single rod withdrawal accident, and rod ejection accident. Assuming the most conservative scenario in which the MOX fuel assemblies fail preferentially, the limiting thyroid radiation doses calculated for the locked rotor accident are about 0.4% greater for a core with four MOX fuel assemblies than for an all-LEU core. Similarly, the limiting thyroid radiation doses for the rod ejection accident would increase by at most 0.2% and the calculated LOCA doses would increase by less than 0.1%. The analyses and evaluations described in Sections 3.7.1 and 3.7.2 conclude that the MOX fuel assemblies should not fail preferentially in these events.

3.7.3.4 Whole Body and Skin Doses

Whole body radiation doses showed essentially no change for any of the FHAs or weir gate drop accidents. Skin radiation dose to the control room operators decreased for MOX fuel relative to LEU fuel. Whole body and skin radiation doses are influenced primarily by the activity of other radioisotopes, in particular, the noble gases. In general, the activities of the noble gas radioisotopes were shown to either remain essentially unchanged or decrease in MOX fuel relative to LEU fuel.

3.7.3.5 Fresh MOX Fuel Assembly Drop

Both plutonium and uranium are alpha emitters, but the plutonium isotopes in MOX fuel have a much shorter half-life than the uranium isotopes in LEU fuel. Therefore, plutonium has a much higher specific activity level than uranium and can present a more severe radiological hazard if inhaled. Protection against inhalation hazards is provided by the physical form of plutonium in insoluble ceramic pellets that are contained within welded metal rods. However, if a fresh MOX fuel assembly is severely damaged prior to being placed in the spent fuel pool, it is theoretically possible that some plutonium might become airborne and therefore pose the possibility of an inhalation risk.

Accordingly, a bounding analysis of the offsite and control room radiological consequences was performed for a drop of a fresh MOX fuel assembly in air. Due to the limited amount of data on the expected damage and release from such an occurrence, very conservative assumptions and modeling were employed. Because of the bounding nature of the analyses, especially in predicting the amount of fuel assembly damage, subsequent airborne release and filtration, the expected amount of fuel assembly damage, and subsequent release would be bounded by this model. Where applicable, more conventional and conservative assumptions were made including licensing basis dispersion factors and Federal Guidance Report (Reference 18) dose conversion factors. Using these extremely conservative assumptions, the resulting calculated dose for both the exclusion area boundary (EAB) and the control room from a 30 foot drop was about 0.3 Rem TEDE. More realistic assumptions resulted in calculated doses of less than 0.1 Rem TEDE for the EAB and control room. These results are much lower than the EAB dose limits of 25 Rem TEDE or control room dose limits of 5 Rem TEDE.

3.8 RISK IMPACT OF MOX FUEL LEAD ASSEMBLIES

The use of four MOX fuel lead assemblies (out of a total of 193 fuel assemblies in the core) will not significantly change the risk to public health and safety that is posed by operation of McGuire and Catawba.

Duke uses probabilistic risk assessment (PRA) analyses to evaluate the risk to public health and safety due to operation of its nuclear plants. PRA analyses quantify the probability and consequences of severe accidents that involve core melt and containment failure events. Key considerations in PRA analyses are equipment requirements to prevent core melt (success criteria); ice melt times, containment pressurization rates, and potential containment failures (containment performance); and doses to the public (offsite consequences). The attributes of MOX fuel that impact these areas are fundamentally similar to uranium fuel, as discussed below.

- Plant configuration: The plant equipment, including passive and active safety systems, is unaffected by the type of fuel in the core.
- Fuel characteristics: As discussed in Reference 1, MOX fuel is fundamentally similar to conventional LEU fuel. Both MOX and LEU fuel consist of sintered ceramic

pellets with very similar material properties. Both MOX and LEU fuel are clad with zirconium alloy metal. The MOX fuel assembly design is very similar to the resident LEU fuel assembly design.

- Decay heat: Decay heat from MOX fuel is slightly lower than decay heat from an equivalent LEU fuel assembly during the time frame of concern for core melt accidents.
- Radionuclide inventory: Irradiated MOX fuel has a somewhat different radionuclide inventory than LEU fuel. For fission products, the same radionuclides are present, in generally similar amounts, as discussed in Section 3.7.3 (Radiological Consequences of Design Basis Accidents). Irradiated MOX fuel contains significantly more actinides (plutonium, neptunium, americium, and cerium) than equivalent uranium fuel. However, these elements do not transport to the environment nearly as effectively as lighter radionuclides. Therefore, the dose consequences of the actinides are small relative to radionuclides like iodine and cesium.

Due to the fundamental similarity between MOX fuel assemblies and uranium fuel assemblies, and the identical plant configuration, four MOX fuel lead assemblies will have no appreciable impact on the thermal-hydraulic response of the core. Success criteria are dominated by the plant configuration and core decay heat, and four MOX fuel lead assemblies will therefore have no adverse impact in this area. Containment performance is also dominated by the plant configuration and core decay heat; again, no adverse impact on severe accident consequences would be expected.

The largest impact of MOX fuel use will derive from differences in the original radionuclide inventory. The most important radionuclides from an offsite dose perspective are volatile fission products like cesium and iodine. Those radionuclide inventories are generally similar for MOX and LEU fuel. Higher actinide concentrations in MOX fuel have the potential to cause higher offsite doses, but this is a smaller effect. In Reference 19 the Department of Energy evaluated several severe accident sequences at McGuire and Catawba for cores containing all-uranium fuel and cores containing approximately 40% MOX fuel. In the DOE analyses, offsite consequences from severe accidents ranged from minus 4% to plus 14% compared to LEU fuel. For cores containing four MOX fuel lead assemblies (2% of the total number of fuel assemblies), the potential impact on offsite consequences from severe accidents would range from about minus 0.1% to plus 0.3% compared to LEU fuel. Accordingly, it is concluded that operation with up to four MOX fuel lead assemblies will have no significant impact on public health risk at either McGuire or Catawba.

3.9 REFERENCES

1. *MOX Fuel Design Report*, BAW-10238(NP), Revision 0, Framatome ANP, March 2002.

2. *Advanced Mark-BW Fuel Assembly Mechanical Design*, BAW-10239(P), Revision 0, Framatome ANP, March 2002.

3. *COPERNIC Fuel Rod Design Computer Code,* BAW-10231P, Revision 0, September 1999.

4. David B. Mitchell and Bert M. Dunn, *Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel*, BAW-10227P-A, February 2000.

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

6. Duke Power Company *McGuire/Catawba Core Thermal-Hydraulic Methodology Using VIPRE-01*, DPC-NE-2004P-A, Revision 1, February 1997.

7. Duke Power Company *Westinghouse Fuel Transition Report*, DPC-NE-2009P-A, September 1999.

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

9. NUREG/CR-0200, CCC-545, *Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation*, Oak Ridge National Laboratory, November 1993.

10. D.A. Wesley and K. J. Firth, *TACO3 – Fuel Pin Thermal Analysis Code*, BAW-10162PA, October 1989.

11. *Extended Burnup Evaluation,* BAW-10186PA, Revisions 1and 2, and "Supplement 1 to BAW-10186P Revision 1 Mark-BW Extended Burnup."

12. Letter Framatome ANP to U. S. Nuclear Regulatory Commission, Response to Request for Additional Information on BAW-10231P "COPERNIC Fuel Rod Design Code," Response to Question 3, February 5, 2001.

13. Letter Framatome Technologies to U.S. Nuclear Regulatory Commission, "Modeling Refinements to Framatome Technologies RELAP5-Based, Large Break LOCA Evaluation Models – BAW-10168 for Non-B&W-Designed, Recirculating Steam Generator Plants and BAW-10192 for B&W-Designed, Once-Through Steam Generator Plants," FTI-00-551, February 29, 2000.

14. Letter U. S. Nuclear Regulatory Commission to Framatome ANP, "Safety Evaluation of Framatome Technologies Topical Report BAW-10164P, Revision 4, "RELAP5/MOD2-B&W, An Advanced Computer Transient Analysis (TAC Nos. MA8465 and MA8568)," April 9, 2002.

15. *RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants*, BAW-10168PA, Revision 3, December 1996.

16. Duke Power Company *Thermal-Hydraulic Transient Analysis Methodology*, DPC-NE-3000-PA, Revision 2, December 2000.

17. Duke Power Company *Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology*, DPC-NE-3001-PA, December 2000.

18. Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, DC, 1988.

19. *Surplus Plutonium Disposition Final Environmental Impact Statement,* DOE/EIS-0283, U.S. Department of Energy, Office of Fissile Materials Disposition, November 1999.

Table 3-1 Mark-BW/MOX1 Preliminary Design Summary

Parameter	Value	
	Advanced Mark-BW	Mark-BW/MOX1
Pellets		
Fuel Pellet Material	Enriched UO_2	PuO_2 and Depleted UO_2
Fuel Pellet Diameter, in	0.3225	0.3225
Fuel Pellet Theoretical Density, %TD	96	95
Fuel Pellet Volume Reduction due to Chamfer and Dish, %	1.24	1.0
Rods		
Fuel Rod Length, in	152.16	152.40
Fuel Rod Cladding Material	M5™	M5™
Fuel Rod Inside Diameter, in	0.329	0.329
Fuel Rod Outside Diameter, in	0.374	0.374
Active Fuel Stack Height, in	144	144
Maximum Fuel Rod Burnup, MWd/MThm	60,000	50,000
Assemblies		
Fuel Assembly Length, in	159.8	159.8
Lattice Geometry	17x17	17x17
Fuel Rod Pitch, in	0.496	0.496
Number of Fuel Rods per Assembly	264	264
Heavy Metal Loading per Assembly, kg	466.1	462.6
Total Fuel Assembly Weight, kg	674.9	670.9
Number of Grids		
Bottom End	1	1
Vaneless Intermediate	1	1
Vaned Intermediate	5	5
Mid-Span Mixing	3	3
Top End	1	1

Table 3-2

Plant Parameters and Operating Conditions Used In LOCA Evaluation

Parameter	Value
Reactor Power (MWt)	3411
Pressurizer Operating Pressure (psia)	2310
System Flow (gpm)	382,000
Hot Leg Temperature (degrees F)	616
Cold Leg Temperature (degrees F)	555
Core Average Linear Power Generation Rate* (kW/ft)	5.69
Highest Allowable Total Peaking for MOX Fuel Assembly (F_Q)	2.4
Hot Pin and Hot Assembly Radial Peaking Factors	1.60
Core Axial Peaking Factor	1.50

* Increased to include 102 percent of rated power

Table 3-3

Case 1 - Sequence of Events for MOX Fuel Lead Assembly Calculation

Event	Time (seconds)
Leak Initiation	0
Accumulator Injection Begins	12.8
End of Blowdown	25.3
Bottom of Core Recovery	39.7
Rupture in Hot Assembly	73
Peak Cladding Temperature (unruptured node)	130

Table 3-4

Plutonium Loading LOCA Results Comparison

Calculation Results	2.3 % Pu Pin	3.6 % Pu Pin	4.4 % Pu Pin
Peak Cladding Temperature (degrees F)	2018	2017	2017
PCT Location (ft)	8.8	8.8	8.8
Peak Cladding Temperature at Rupture Location (degrees F)	1841	1841	1841
Hot Pin Rupture Location (ft)	9.7	9.7	9.7
Hot Pin Rupture Time (sec)	73	73	73

Table 3-5

LBLOCA Sample Calculations Comparison

Results	Case 1 MOX Fuel	Case 2 MOX Fuel 2x Gap Factor	Case 3 LEU Fuel
Peak Cladding Temperature Data (Peak Pin Data)			
Peak Cladding Temp. (degrees F)	2018	2005	1981
PCT Location (ft)	8.8	8.8	8.8
Rupture Node Data			
Peak Temperature at Rupture Location (degrees F)	1841	1783	1753
Hot Pin Rupture Location (ft)	9.7	9.7	9.7
Hot Pin Rupture Time (sec)	73	73	71
Oxidation Data			
Max. Local Oxidation* (percent)	4.5	4.6	4.0
Location of Max. Oxidation (ft)	8.8	8.8	8.8

* Local Oxidation at the end of 400 second simulation.

Table 3-6

Comparison of Resident Fuel LOCA Calculation to MOX Fuel Calculation

	MOX Fuel	Resident Fuel (95 percentile)	Difference
Highest Allowable Total Peaking (F_Q)	2.4	2.5	-0.1
Peak Cladding Temperature (degrees F)	2018	2056	-38
Maximum Local Oxidation* (percent)	4.5	10	-5.5

*After 400 seconds

Table 3-7

Boron Letdown, Assembly Power, and Pin Power Comparisons between MOX Fuel Lead Assembly and LEU Cores

EFPD	POWER (percent)	BORON (ppm)			MAX ASSY POWER 2RPF			2-D PEAK PIN POWER 2PIN		
		MOX	LEU	DELTA	MOX	LEU	DELTA	MOX	LEU	DELTA
0	0	1832	1815	17	1.407	1.334	0.073	1.557	1.498	0.059
4	100	1242	1235	7	1.291	1.284	0.007	1.426	1.423	0.003
12	100	1224	1218	6	1.272	1.277	-0.005	1.411	1.418	-0.007
25	100	1234	1230	4	1.272	1.275	-0.003	1.416	1.420	-0.004
50	100	1260	1258	2	1.270	1.270	0.000	1.421	1.421	0.000
100	100	1249	1250	-1	1.321	1.317	0.004	1.401	1.397	0.004
150	100	1170	1173	-3	1.345	1.340	0.005	1.414	1.409	0.005
200	100	1046	1051	-5	1.357	1.353	0.004	1.430	1.425	0.005
250	100	892	898	-6	1.373	1.365	0.008	1.437	1.431	0.006
300	100	720	728	-8	1.375	1.366	0.009	1.435	1.425	0.010
350	100	537	545	-8	1.361	1.354	0.007	1.420	1.413	0.007
400	100	350	359	-9	1.339	1.332	0.007	1.395	1.388	0.007
450	100	164	173	-9	1.313	1.307	0.006	1.368	1.362	0.006
470	100	91	100	-9	1.302	1.297	0.005	1.357	1.351	0.006
490	100	19	28	-9	1.293	1.289	0.004	1.347	1.342	0.005
495	100	1	10	-9	1.291	1.287	0.004	1.344	1.340	0.004

Table 3-8

Beta-Effective and Prompt Neutron Lifetime Comparisons between MOX Fuel Lead Assembly and LEU Cores

	EFPD	POWER (percent)	BETA-EFFECTIVE	DELTA	PROMPT NEUTRON LIFETIME	DELTA
MOX	4	100	0.00609		15.74	
LEU	4	100	0.00622	-0.00013	16.03	-0.29
MOX	495	100	0.00504		19.57	
LEU	495	100	0.00509	-0.00005	19.76	-0.19

Table 3-9

Equilibrium Xenon Worth Comparisons between MOX Fuel Lead Assembly and LEU Cores

EFPD	POWER (percent)	EQUIL. XENON WORTH (pcm)		
		MOX	LEU	DELTA
4	100	-2389	-2415	26
200	100	-2625	-2640	15
495	100	-2836	-2851	15

Table 3-10

Comparisons of Isothermal Temperature Coefficient, Moderator Temperature Coefficient, Doppler Coefficient, and Differential Boron Worth between MOX Fuel Lead Assembly and LEU Cores

EFPD	POWER (percent)	BORON (ppmb)	ITC (pcm/°F)			MTC (pcm/°F)		
			MOX	LEU	DELTA	MOX	LEU	DELTA
0	100	1832	-8.48	-8.05	-0.43	-7.03	-6.60	-0.43
0	0	1832	-3.47	-3.10	-0.37	-1.76	-1.40	-0.36
4	100	1242	-13.84	-13.47	-0.37	-12.40	-12.04	-0.36
4	0	1242	-8.15	-7.85	-0.30	-6.46	-6.18	-0.28
200	100	1046	-18.34	-17.95	-0.39	-16.85	-16.47	-0.38
200	0	1046	-10.90	-10.60	-0.30	-9.18	-8.89	-0.29
495	100	1	-37.56	-37.25	-0.31	-35.92	-35.61	-0.31
495	0	1	-26.47	-26.25	-0.22	-24.66	-24.43	-0.23
EFPD	POWER (percent)	BORON (ppmb)	DOPPLER (pcm/°F)			DIFF BORON WORTH (pcm/ppm)		
			MOX	LEU	DELTA	MOX	LEU	DELTA
0	100	1832	-1.45	-1.45	0.00	-6.19	-6.30	0.11
0	0	1832	-1.71	-1.70	-0.01	-6.54	-6.68	0.14
4	100	1242	-1.44	-1.43	-0.01	-6.30	-6.40	0.10
4	0	1242	-1.69	-1.67	-0.02	-6.66	-6.78	0.12
200	100	1046	-1.49	-1.48	-0.01	-6.49	-6.56	0.07
200	0	1046	-1.72	-1.71	-0.01	-6.82	-6.89	0.07
495	100	1	-1.64	-1.64	0.00	-7.94	-8.01	0.07
495	0	1	-1.81	-1.82	0.01	-8.28	-8.35	0.07

Note: Boron concentrations in this table are for a representative core with MOX fuel lead assemblies. Table 3-7 has the corresponding boron concentrations for an all-LEU core.

Figure 3-1 Mark-BW/MOX1 Fuel Rod Design

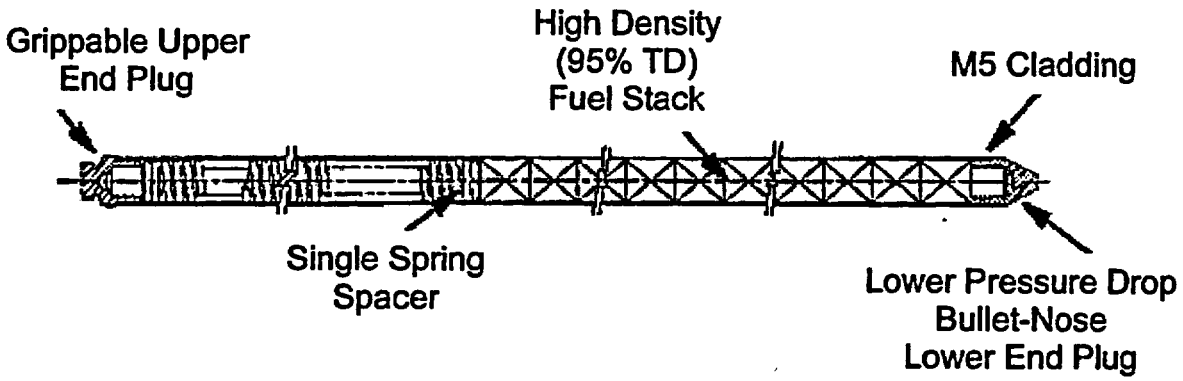


Figure 3-2
Reactivity Insertion versus Void Fraction
Infinite Lattice UO_2 at MTC = 0.0 pcm/degree F

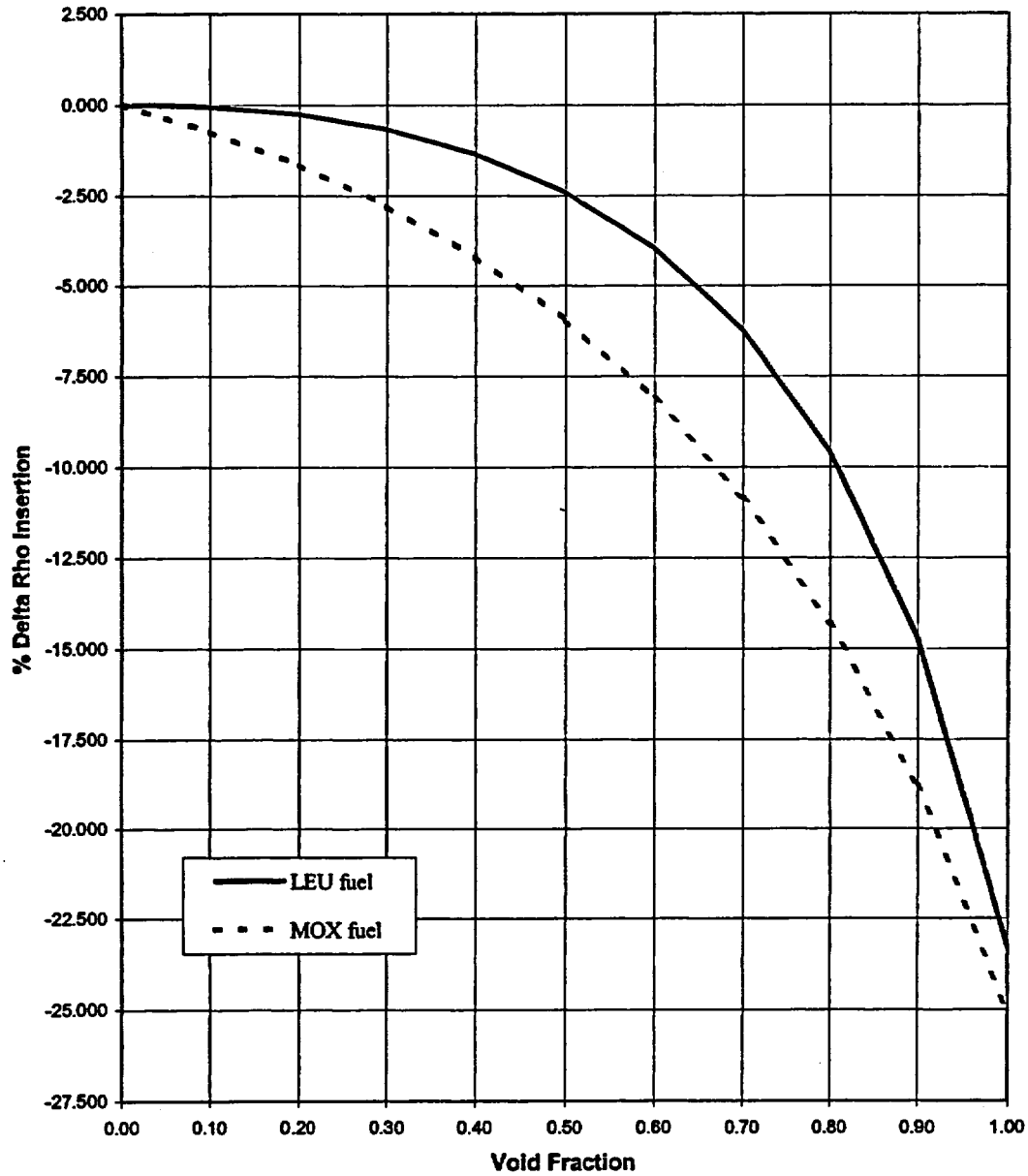


Figure 3-3
Decay Heat Rate Comparisons
MOX and LEU Fuel Fission Products plus Actinides

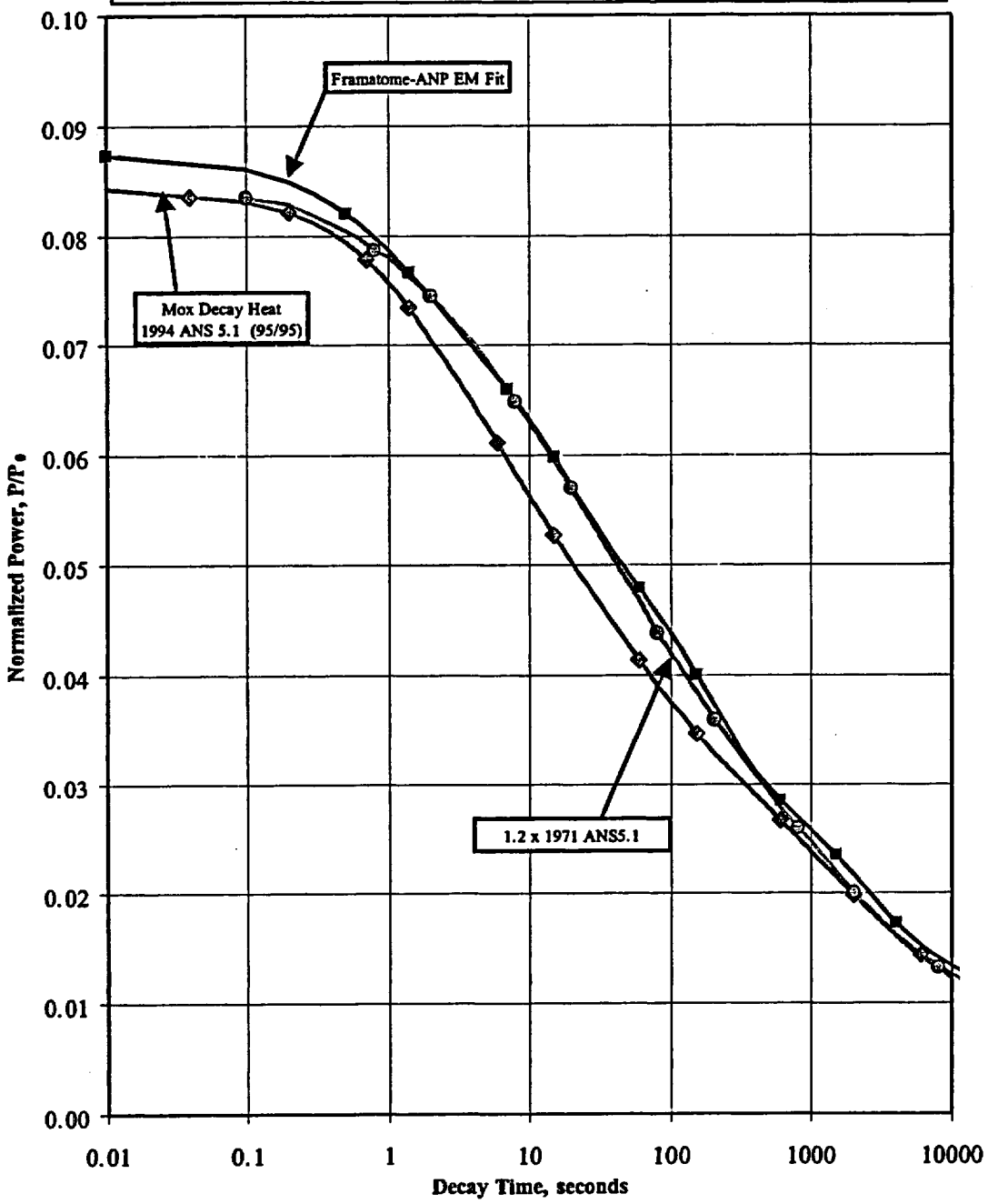
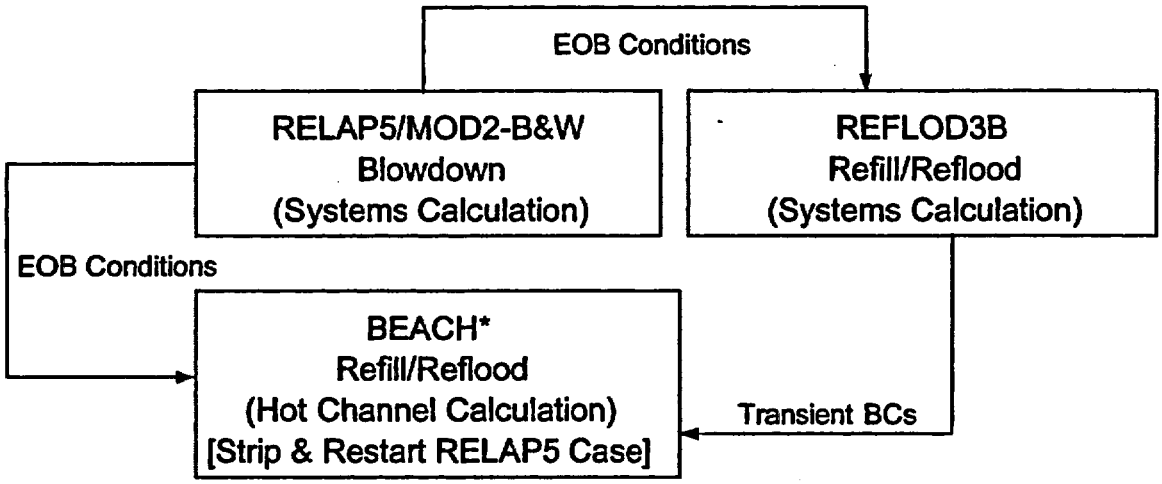


Figure 3-4
Framatome ANP Recirculating Steam Generator
LOCA Evaluation Model Codes



*BEACH is a set of reflood heat transfer subroutines in RELAP5.

Figure 3-5
RELAP5/BEACH Core Noding with Mid Span Mixing Grids

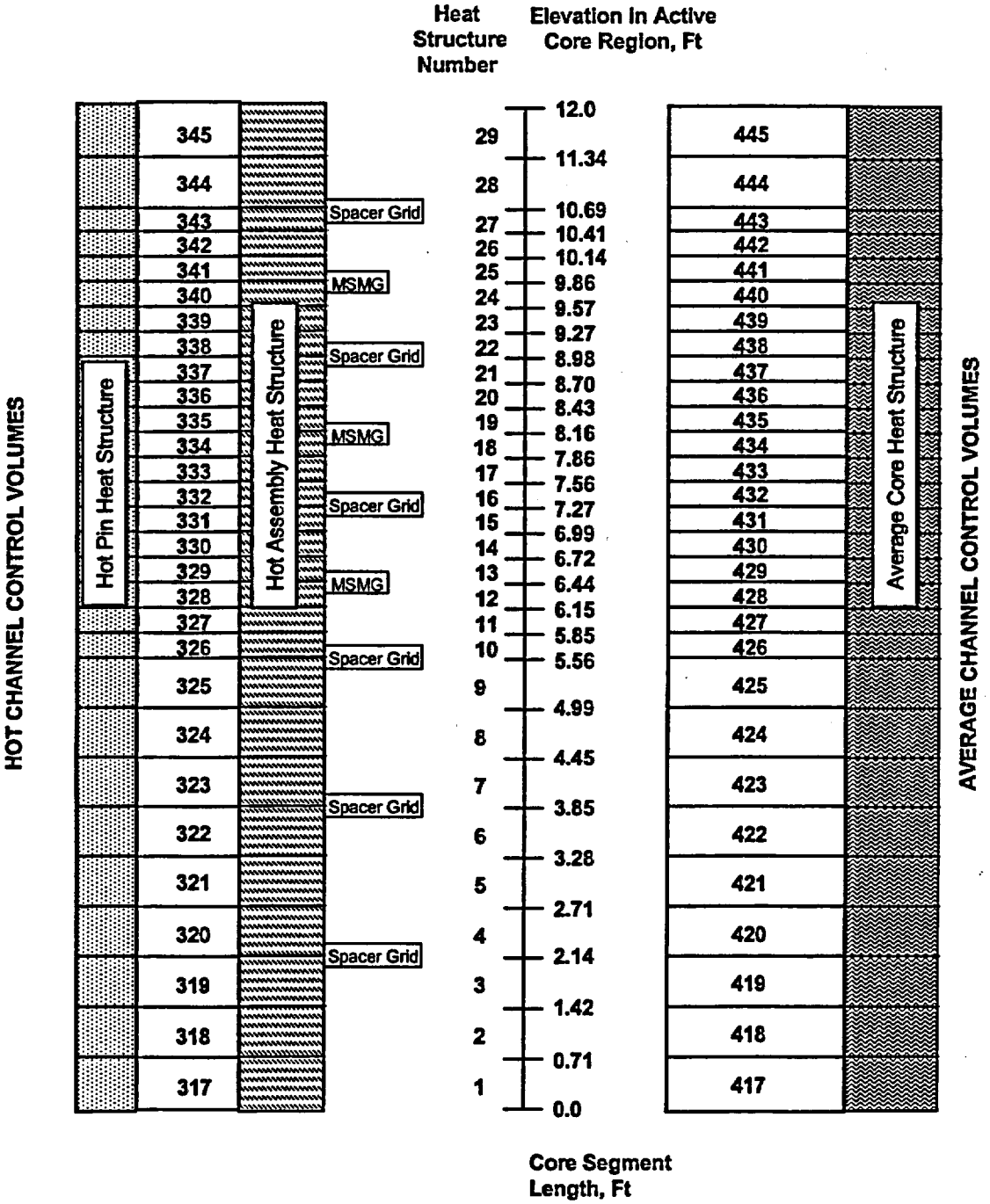


Figure 3-6
RCS Pressure for MOX LOCA Calculations during Blowdown (10.3 ft Axial Peak)

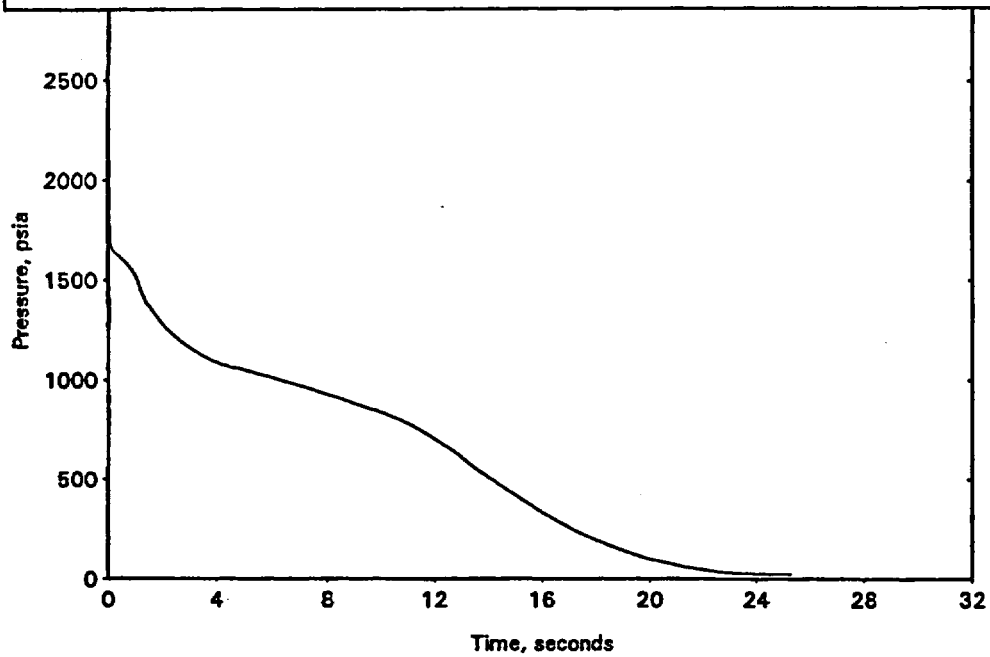


Figure 3-7
Reflooding Rate for MOX LOCA Calculations (10.3 ft Axial Peak)

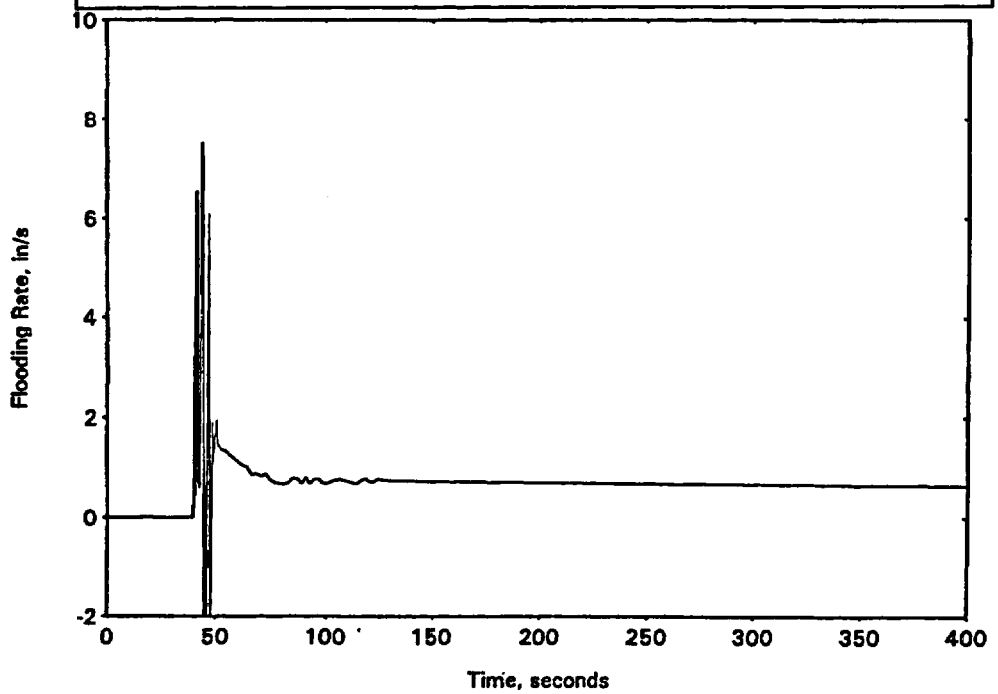


Figure 3-8
Fuel to Clad Gap Multiplier Study Results (10.3 ft Axial Peak)

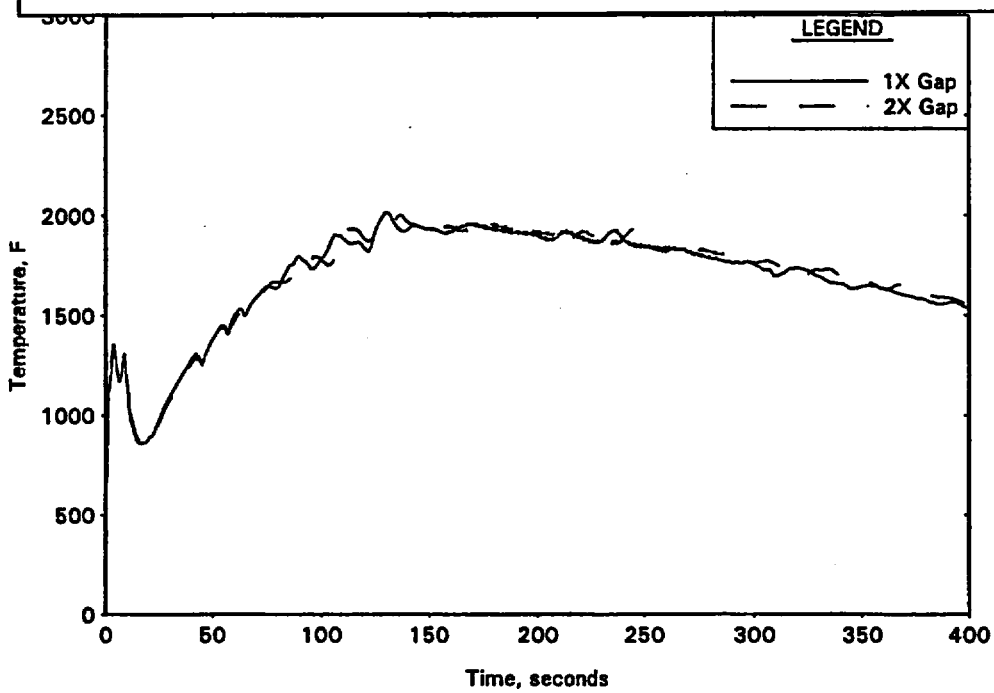


Figure 3-9
Fuel to Clad Gap Multiplier Study
Ruptured Node Results (10.3 ft Axial Peak)

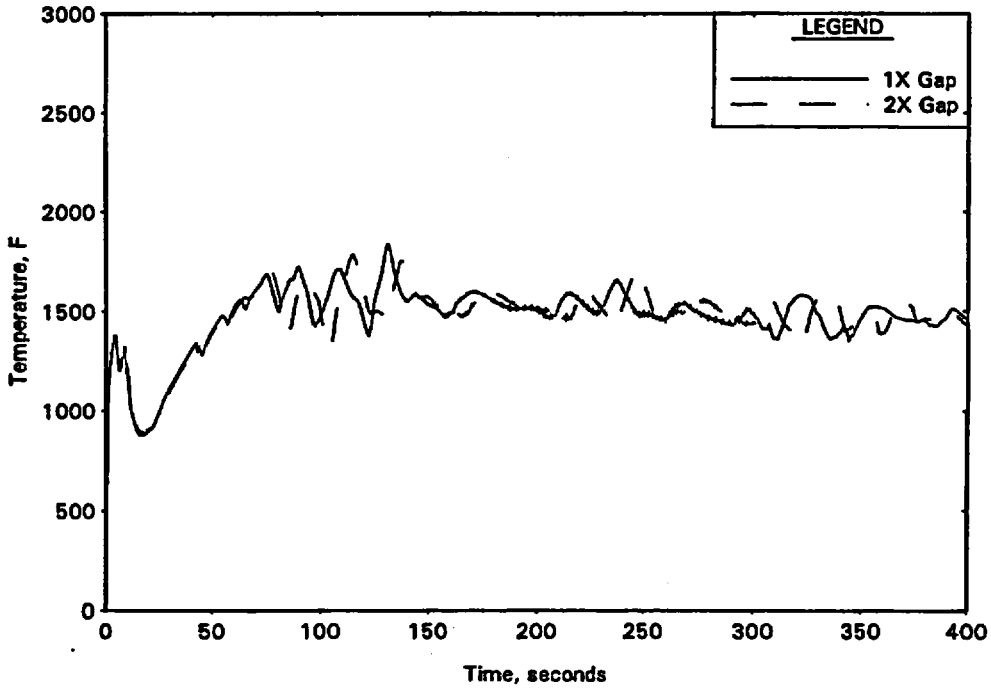


Figure 3-10
Hot Pin PCT MOX Lead Assembly vs. LEU Assembly (10.3 ft Axial Peak)

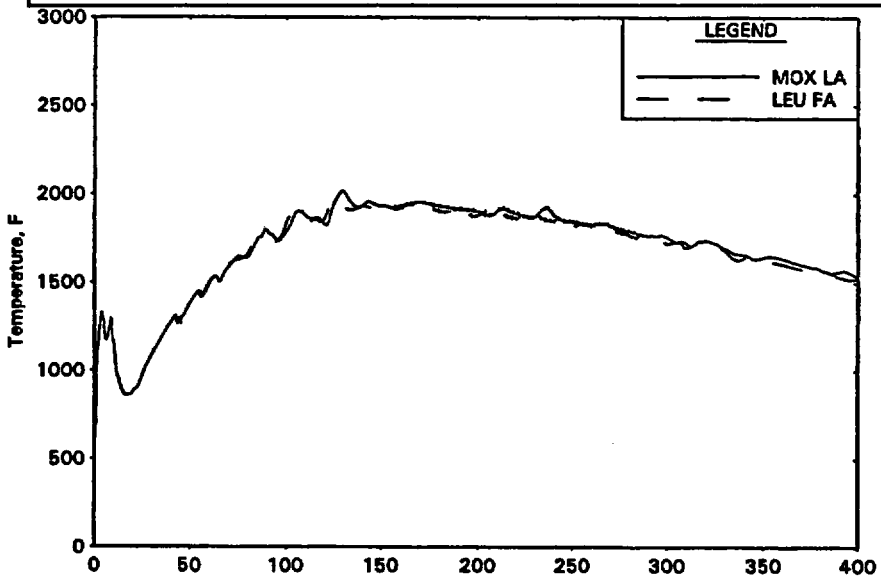


Figure 3-11
Hot Pin Ruptured Location PCT MOX Lead Assembly vs. LEU Assembly (10.3 ft Axial Peak)

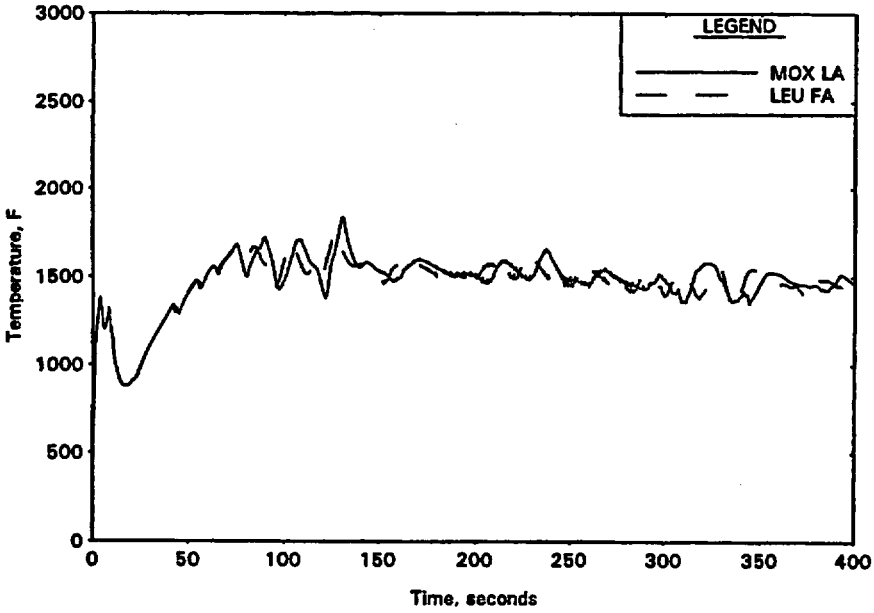
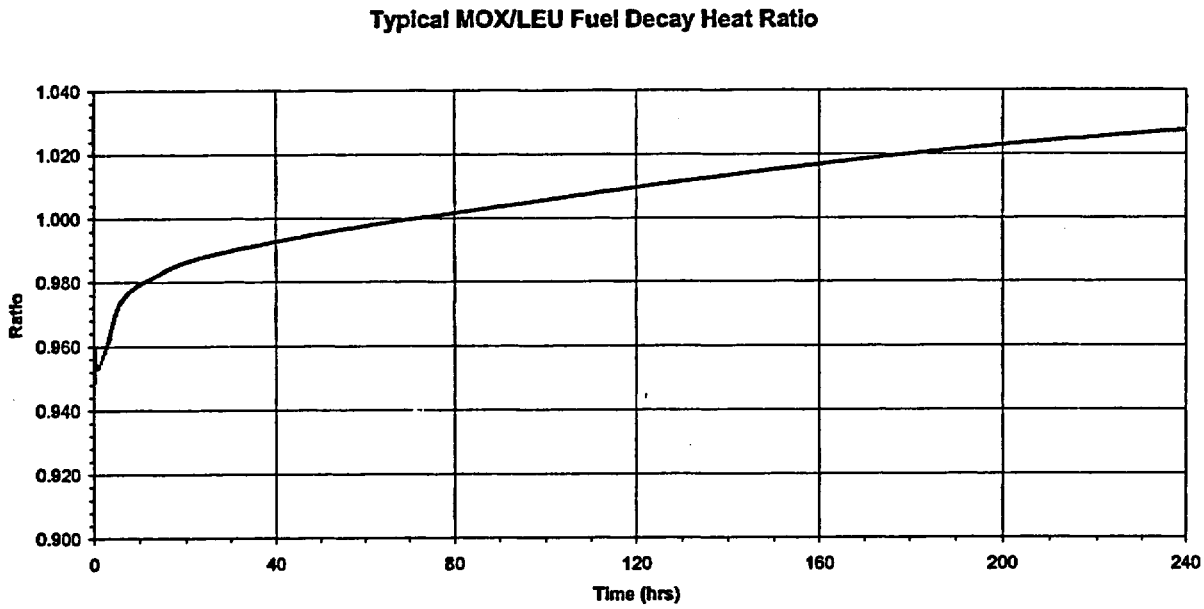


Figure 3-12
Ratio of Short Term
MOX Fuel/LEU Fuel Decay Heat



Appendix 3-1

Criticality Evaluation of MOX Fuel Storage in the McGuire and Catawba Spent Fuel Pools

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A3.1. Introduction / Background

The focus of this analysis is to evaluate storage of MOX fuel lead assemblies in the McGuire and Catawba spent fuel pools. In particular, this analysis will determine whether the current low-enriched uranium (LEU) fuel storage configurations and strategies employed at McGuire and Catawba will be adequate to store MOX fuel in accordance with regulatory subcriticality limits.

Section A3.2 describes the pertinent fuel storage facilities at McGuire and Catawba. The layout, physical characteristics, and storage capacities of the new fuel and spent fuel storage areas are detailed. In addition, the general procedure for receiving, storing, and reactor loading of fuel assemblies at the facility is discussed.

Section A3.3 describes the reference MOX fuel assembly design, and compares this with the LEU fuel currently used in the McGuire and Catawba reactors. Pertinent criticality modeling parameters are provided for the reference MOX fuel assembly.

A brief discussion of the general neutronic behavior of MOX fuel constitutes Section A3.4. These nuclear characteristics will help both to guide the subsequent MOX criticality calculations and to explain the observed results from these calculations.

In Section A3.5 the computer codes used for the MOX fuel storage criticality analyses are identified. This section also describes and discusses code benchmarking to appropriate MOX fuel critical experiments.

Finally, Section A3.6 contains the criticality analysis for MOX fuel storage in the McGuire and Catawba spent fuel pools (SFPs). This criticality evaluation is performed in accordance with the methods that have been used in counterpart SFP analyses for LEU fuel storage. Both normal and accident conditions are considered. To help simplify the requisite calculations, the analysis does not take any reactivity credit for MOX fuel burnup.

A3.2. Fuel Storage Facilities at McGuire and Catawba

Figure A3-1 shows an overhead view of the pertinent fuel storage areas in one of the McGuire fuel buildings. This layout is typical of the two (2) fuel buildings at McGuire as well as the two (2) fuel buildings at Catawba. Fresh fuel is first received in the new fuel receiving area and stored temporarily prior to being removed from its shipping container. Upon removal from the shipping container LEU fuel assemblies are placed in a new fuel storage vault (NFV) location for inspection and then are either kept in the NFV or transferred to the spent fuel pool (SFP) for storage prior to reactor irradiation. MOX fuel assemblies, on the other hand, will be placed directly in the SFP once they have been received on-site – the NFVs will not be used to store MOX fuel assemblies. Fresh fuel and irradiated reload fuel assemblies (both LEU and MOX) are transported to the reactor via the water-filled Fuel Transfer Area. Discharged fuel assemblies from the reactor are also returned to the Spent Fuel Pool through the Fuel Transfer Area. Qualified spent fuel assemblies (currently LEU only) may be loaded into dry storage casks in the Cask Area. Once the dry storage casks are drained, sealed, and decontaminated, they are taken to the on-site independent spent fuel storage installation (ISFSI) for interim storage.

The SFPs are designed to store fresh and irradiated fuel assemblies in a wet, borated environment. The McGuire SFPs are divided into two regions: Region 1 and Region 2. The Region 1 storage racks have a flux trap design, with the rack cell walls composed of stainless steel. Boraflex poison panels are attached to the outsides of each of the Region 1 rack cell walls. Figure A3-2 depicts the storage of four fuel assemblies in the Region 1 cells. McGuire Region 1 is normally used for storage of fresh fuel and irradiated fuel that will be reloaded into the reactor core.

Region 2 in the McGuire SFPs is designed to store fuel assemblies that have been permanently discharged from the reactor. Generally these are high-burnup fuel assemblies with low enough reactivity that they can be stored in the tighter Region 2 configuration. Figure A3-3 shows the McGuire Region 2 storage layout. This design is called the "cell / off-cell" or "egg-crate" pattern, because it consists of a tight checkerboarded cluster of stainless steel rack cells. The holes in this pattern are the off-cells, and fuel assemblies are stored in these off-cells as well. As with Region 1, Boraflex poison panels are attached to each of the cell walls in the Region 2 racks.

The Catawba SFPs differ from McGuire's in that they contain just one storage region – that is, all rack cells are the same design. The Catawba racks, as shown in Figure A3-4, are arranged in a flux trap pattern, similar to McGuire Region 1. However, the spacing between storage cells is larger in the Catawba racks, and the cell walls are thicker. As a result, the Catawba racks contain no Boraflex poison material, since the additional reactivity holddown is not needed.

Table A3-1 provides McGuire and Catawba SFP rack data important to the criticality modeling of these storage regions. Note that, as a result of measured and projected degradation of the Boraflex panels in the McGuire SFP storage racks, McGuire Region 1 currently is only allowed credit for a maximum of 25% of its original Boraflex loading (as listed in Table A3-1), while McGuire Region 2, pending NRC approval, can take credit for up to 40% of it. To account for advanced Boraflex degradation in some

McGuire storage cells, below the 25% and 40% thresholds described above, the McGuire storage regions have been further subdivided as follows:

McGuire Region 1A – 25% of original Boraflex loading
McGuire Region 1B – 0% of original Boraflex loading
McGuire Region 2A – 40% of original Boraflex loading
McGuire Region 2B – 0% of original Boraflex loading

Table A3-1. General Design Information for the McGuire and Catawba SFP Storage Racks

	McGuire Region 1	McGuire Region 2	Catawba
# of storage locations in each SFP	286	1177	1429
Storage cell pitch (cm)	26.4	23.2 (average)	34.3
Original Boraflex Loading (g/cm^2)	0.020	0.006	None
Storage cell wall thickness (cm)	0.19	0.19	0.64
Normal SFP water temperature range (degrees F)	68 - 150	68 - 150	68 - 150
Minimum required SFP boron concentration (ppm)	2675	2675	2700

A3.3. Reference MOX Fuel Assembly Design

The MOX fuel lead assembly design is described in Attachment 3, of the MOX fuel lead assembly license amendment request. The reference MOX fuel assembly evaluated for SFP storage contains a total plutonium concentration of 4.37 weight percent up to a maximum fissile plutonium concentration of 4.15 weight percent and a maximum U-235 enrichment of 0.35 weight percent. Figure A3-6 illustrates the fuel rod layout within the MOX fuel lead assembly. With the exception of the fuel material itself, the other Mark-BW/MOX1 fuel design parameters important to neutronic analysis (pellet diameter, fuel density, active stack length, rod pitch, etc.) are identical or nearly identical to those parameters of the current LEU fuel assemblies being used at McGuire and Catawba.

Table A3-2 provides the plutonium and uranium nominal isotopic fractions for the unirradiated Mark-BW/MOX1 fuel. Expected manufacturing variations from the nominal values are also listed, and these variations are considered in the mechanical uncertainty analysis in Section A3.6.

Table A3-2. MOX Fuel Initial Isotopic Fractions

Isotope	Nominal % of base element	Expected Manufacturing Range (%)
Pu-238	0.03	≤ 0.05
Pu-239	92.5	90.0 – 95.0
Pu-240	6.92	5.0 – 9.0
Pu-241	0.5	≤ 1.0
Pu-242	0.05	≤ 0.1
U-235	0.35 (max)	NA

A3.4. General Neutronic Behavior of MOX Fuel in Spent Fuel Pool Storage Conditions

The reference MOX fuel assembly contains significant quantities of Pu-239, which is a more effective thermal and epithermal neutron absorber than U-235. As a result, other thermal neutron absorbers in the MOX fuel lattice (such as boron) are worth less than in a LEU fuel lattice. The boron atoms, whether dissolved in the coolant or in lumped burnable poison rods, do not compete for thermal neutrons as effectively with the Pu-239 in MOX fuel as they do with U-235 in LEU fuel.

Another important effect is the reactivity letdown characteristic of MOX fuel. Higher plutonium isotopes build in more quickly with burnup in MOX fuel than in LEU fuel, because the MOX fuel assemblies start with appreciable amounts of Pu-239. This difference in the buildup and burnup characteristics of plutonium isotopes results in a flatter MOX fuel reactivity curve (reactivity drops off less steeply with burnup) than an equivalent LEU fuel reactivity curve.

Reference 1 provides a more extensive discussion of the nuclear characteristics of weapons grade MOX fuel and how those characteristics affect storage criticality calculations.

A3.5. Computer Code Validation and Usage for MOX Fuel Criticality Analyses

The SCALE 4.4 / KENO V.a computer code system (Reference 2) is employed for the MOX and LEU fuel criticality analyses documented in Section A3.6. This code system is well-suited to spent fuel pool criticality applications, and has been extensively benchmarked to both MOX fuel and LEU fuel critical experiments as well as reactor operational data.

As noted in A3.1, the criticality computations for this evaluation of the MOX fuel lead assemblies will consider only unirradiated MOX fuel. That is, no burnup credit will be taken, and so no reactivity-equivalencing curves will be necessary. Therefore, the criticality calculations for the MOX fuel lead assemblies can be performed solely with KENO V.a. Note that KENO V.a does have the capability of modeling burned fuel. However, this requires first generating isotopic number densities (typically via the SAS2H module in SCALE 4.4), and then putting that isotopic data into KENO V.a. Because SAS2H, which was not originally intended for fuel criticality applications, is a 1-D transport code, it is preferable to use a 2-D transport code (e.g., CASMO-4) for burned fuel evaluations. 2-D calculations should more accurately model fuel assemblies that are not uniform radially (such as the variable MOX rod zoning in the Mark-BW/MOX1 design – see Figure A3-6).

Several benchmark reports for using SCALE with MOX fuel have been previously developed. References 3 through 5 describe results from benchmarking SCALE against MOX fuel critical experiments (MIX-COMP-THERM) and against isotopic measurements from reactor-irradiated (Beznau and San Onofre) MOX fuel. Duke Power benchmarking of SCALE 4.4 to MOX fuel critical experiments has yielded good agreement in k_{eff} predictions, with similar biases and slightly higher uncertainties than those previously determined for LEU fuel. Duke Power has evaluated the following critical experiments (References 6 through 9) in this benchmarking effort:

- MIX-COMP-THERM-001. Battelle ONL Experiments, 1978 (4 experiments)
- MIX-COMP-THERM-002. Plutonium Recycle Critical Facility PNL, 1975-1976 (6 experiments)
- MIX-COMP-THERM-003. Critical Reactor Experiment Facility Westinghouse, 1965 (7 experiments)
- MIX-COMP-THERM-004. Tokai Research Establishment of JAERI, 1972-1975 (11 experiments)

All of these MOX experiments contained a mixture of plutonium oxide and uranium oxide fuel with plutonium oxide concentrations ranging from 2.0 wt % to 19.7 wt %. Four of the MOX experiments used the addition of fuel rods to measure critical parameters, 17 experiments varied water level to reach critical, and six of the experiments were intended to measure power distribution (the measurement of critical parameters was secondary).

Results of the SCALE 4.4 / KENO V.a benchmark calculations with MOX fuel are shown in Table A3-3. Note that the KENO V.a models for these benchmark cases used

the 238-group ENDF-V cross-section library. Because these MOX fuel critical experiments yielded a similar method bias and uncertainty, as compared with counterpart LEU fuel critical experiments, the 95/95 k_{eff} computations in Section A3.6 for the McGuire Region 1A and Catawba SFP storage racks (which model MOX and LEU fuel together) use the MOX-only bias (conservatively trended) and uncertainty documented at the end of Table A3-3.

Table A3-3. MOX Critical Experiment SCALE 4.4 / KENO V.a Benchmarking Results

Critical Experiment	Total Pu Conc (wt %)	Soluble Boron Conc (ppm)	Measured k_{eff}	KENO V.a Calculated k_{eff}	KENO V.a Calculated σ (k_{eff})
mct001-01	19.7	0	1.00000	0.99939	0.00214
mct001-02	19.7	0	1.00000	0.99593	0.00193
mct001-03	19.7	0	1.00000	0.99983	0.00195
mct001-04	19.7	0	1.00000	1.00095	0.00199
mct002-01	2.0	2	1.00018	0.99195	0.00192
mct002-02	2.0	688	1.00006	0.99450	0.00183
mct002-03	2.0	1	1.00019	0.99968	0.00179
mct002-04	2.0	1090	1.00022	1.00313	0.00185
mct002-05	2.0	2	1.00096	1.00377	0.00185
mct002-06	2.0	767	1.00013	1.00715	0.00148
mct003-01	6.6	0	1.00000	0.99524	0.00192
mct003-02	6.6	0	1.00000	0.99721	0.00190
mct003-03	6.6	0	1.00000	0.99915	0.00245
mct003-04	6.6	337	1.00000	0.99462	0.00199
mct003-05	6.6	0	1.00000	1.00389	0.00233
mct003-06	6.6	0	1.00000	1.00332	0.00207
mct003-07	6.6	0	1.00000	1.00508	0.00205
mct004-01	3.0	0	1.00000	0.99291	0.00193
mct004-02	3.0	0	1.00000	0.99859	0.00167
mct004-03	3.0	0	1.00000	0.99602	0.00201
mct004-04	3.0	0	1.00000	0.99786	0.00167
mct004-05	3.0	0	1.00000	0.99923	0.00194
mct004-06	3.0	0	1.00000	0.99714	0.00196
mct004-07	3.0	0	1.00000	0.99954	0.00176
mct004-08	3.0	0	1.00000	1.00356	0.00198
mct004-09	3.0	0	1.00000	0.99884	0.00178
mct004-10	3.0	0	1.00000	1.00110	0.00208
mct004-11	3.0	0	1.00000	1.00110	0.00148

Calculated Results for these 28 MOX critical experiments:

Avg k_{eff} = 0.99943
Method Bias = + 0.00075 Δk (average)
Method Uncertainty = ± 0.00750 Δk

A3.6. MOX SFP Criticality Analyses

Current regulations allow partial credit for soluble boron in maintaining adequate subcriticality in SFPs. The requirements for adopting this method are provided very generally in 10CFR50.68 (b), with more specific guidance in the NRC-approved Reference 10 methodology. This boron credit methodology has been approved for use with all LEU fuel in the McGuire SFPs. McGuire must meet the following criteria in using this methodology:

- With the SFP racks loaded with fuel of the maximum permissible enrichment and flooded with full-density unborated water, the maximum 95/95 k_{eff} shall be less than **1.0**, including all pertinent mechanical and calculational uncertainties.
- With the SFP racks loaded with fuel of the maximum permissible enrichment and flooded with full-density water at a boron concentration of 850 ppm, the maximum 95/95 k_{eff} shall be less than **0.95**, including all pertinent mechanical and calculational uncertainties.

The Catawba SFP storage racks still do not take any credit for soluble boron, and thus, in accordance with 10CFR50.68, Catawba must meet the following criteria in using this methodology:

- With the SFP racks loaded with fuel of the maximum permissible enrichment and flooded with full-density unborated water, the maximum 95/95 k_{eff} shall be less than **0.95**, including all pertinent mechanical and calculational uncertainties.

Given the above regulatory requirements, the MOX fuel criticality analysis for the McGuire and Catawba SFPs comprises the following general steps:

- The design information is obtained for the MOX fuel lead assemblies and LEU fuel assemblies that are being or will be used in the McGuire and Catawba SFPs. Design details for the SFP racks themselves are also necessary, in order to properly model fuel storage in these racks.
- SCALE 4.4 / KENO V.a computer models for the MOX fuel lead assembly design and the highest-reactivity LEU fuel assembly design are constructed. These assemblies are modeled in the McGuire Region 1A, 1B, 2A, 2B, and Catawba SFP storage racks.
- From these nominal models, mechanical uncertainties are determined as discussed later in this section.
- With the nominal models, k_{eff} results are determined for each MOX or MOX / LEU assembly configuration considered for that particular SFP storage rack. To each k_{eff} result various reactivity penalties are added to account for mechanical uncertainties (from the previous step) and code methodology biases/uncertainties, which gives the no-boron 95/95 k_{eff} for that storage configuration combination.

- In the McGuire SFPs, the maximum calculated 95/95 k_{eff} results must be less than 1.00 for these no-boron cases. In the Catawba SFPs, the maximum calculated 95/95 k_{eff} results must be less than 0.95 for these no-boron cases. .
- For each of the McGuire SFP MOX or MOX / LEU configurations considered, the bounding amount of soluble boron credit that reduces the previously determined no-boron 95/95 k_{eff}s from less than 1.00 to less than or equal to 0.95 is calculated. This amount of soluble boron credit required is verified to ensure it does not exceed the amount remaining following a worst-case credible boron dilution event.
- Several potential spent fuel pool accident scenarios are also evaluated, including an assembly misloading event, accidents that increase or decrease the fuel pool water temperature, and a heavy load drop (weir gate) event. The amount of soluble boron needed to keep the 95/95 k_{eff} at or below 0.95 is determined for each of these accidents, and the maximum amount required is verified to ensure it does not exceed the minimum spent fuel pool boron concentration for normal operations (2675 ppm for McGuire, 2700 ppm for Catawba).

The following assumptions are used in the SFP criticality analysis for MOX fuel:

- All conditions are modeled at both 68 and 150 °F, the normal operating temperature bounds for the SFPs. Only the most reactive temperature is used to set the storage requirements.
- All calculations are performed in 2-D; i.e. no axial effects are directly modeled for the nominal rack criticality cases. Because no burnup credit is taken for the MOX fuel lead assemblies in this storage analysis, there are no axial burnup reactivity "end effects" to consider. Therefore it is conservative to model the MOX assemblies in 2-D. Note that, although a qualified LEU Filler fuel assembly stored alongside MOX fuel (in the McGuire Region 1A and Catawba Restricted / Filler configuration – see Figure A3-5) may take credit for burnup, there are still 3 MOX assemblies for every one LEU assembly in this configuration, and so the overall system here still will not have a positive "end effect" bias.
- LEU Filler fuel stored with MOX Restricted fuel in the McGuire Region 1A and Catawba Restricted / Filler will be modeled in SCALE 4.4 / KENO V.a with no burnup, using the pertinent reactivity equivalencing curve previously approved. LEU enrichment vs. burnup data points from these curves are listed in Table A3-5.
- No credit is taken for the spacer grid material. A slight reactivity penalty is applied for spacer grids in the heavy load drop accident evaluations, since these are analyzed in highly borated (2675 ppm or 2700 ppm) conditions, where the water displacement caused by the presence of spacer grids can actually increase system reactivity.

The total SFP 95/95 k_{eff} equation has the following form:

$$\text{[SFP]} \quad 95/95\ k_{eff} = k_{nominal} + \sum B_x + \sqrt{\sum ks_x^{\,2}}$$

where:

$k_{nominal}$ is the k_{eff} computed for the nominal case being considered.

B_x is the computational method bias (from the code benchmarking described in Section A3.5).

ks_x is a 95/95 uncertainty on $k_{nominal}$ from Table A3-4.

For each of the mechanical uncertainty branch cases evaluated by SCALE 4.4 / KENO V.a, the resulting computed k_{eff} is compared with the nominal case k_{eff}. The mechanical uncertainty component is then determined by statistically combining the Δk_{eff} from the calculated nominal condition with the KENO uncertainty for the sensitivity case, according to the following equation:

$$(ks_x)^2 = (\text{KENO mech unc } k_{eff} - k_{nominal})^2 + (1.752 * \sigma_{MUx})^2$$

where σ_{MUx} is the KENO-computed standard deviation of the k_{eff} mean for the mechanical uncertainty branch case of interest. The 1.752 multiplier is the one-sided 95/95 tolerance factor for 600 neutron generations. Each of the SCALE 4.4 / KENO V.a cases in this calculation was run using 600 neutron generations.

Each of the biases and uncertainties listed in Table A3-4 is discussed in more detail in the following paragraphs.

Table A3-4. Pertinent Biases and Uncertainties in the Criticality Calculations for the McGuire and Catawba SFP Storage Racks (with Mark-BW/MOX1 Fuel)

Biases	Uncertainties
Benchmark Method Bias	Benchmark Method Uncertainty
	Plutonium Concentration Manufacturing Uncertainty
	Fuel Density Manufacturing Uncertainty
	Storage Rack Cell Wall Thickness Manufacturing Uncertainty
	Storage Rack Center-to-Center Cell Spacing Uncertainty
	Boraflex Uncertainties

Benchmark Method Bias

As described in Section A3.5, this bias is determined from the benchmarking of the code system used (SCALE 4.4 / KENO V.a), and represents how much the code system is expected to overpredict (negative bias) or underpredict (positive bias) the "true k_{eff}" of

the physical system being modeled. The SCALE 4.4 / KENO V.a benchmark method bias for MOX fuel is presented at the end of Table A3-3.

Benchmark Method Uncertainty

This uncertainty is determined from the benchmarking of the code system used (SCALE 4.4 / KENO V.a), and is a measure of the expected variance (95/95 one-sided uncertainty) of predicted reactivity from the "true k_{eff}" of the physical system being modeled. The critical experiment benchmarks for these codes were described in Section A3.5. The SCALE 4.4 / KENO V.a benchmark method uncertainty for MOX fuel appears at the end of Table A3-3.

Plutonium Concentration Manufacturing Uncertainty

A plutonium concentration uncertainty of ± 0.075 wt % Pu is used for the MOX fuel analysis, based on fabrication specifications.

Fuel Density Manufacturing Uncertainty

For both LEU and MOX fuel, tolerances on pellet dishing, pellet diameter, and pellet densification can increase the effective fuel density from the nominal values. No manufacturing tolerances for the Mark-BW/MOX1 are currently available. However, previous LEU fuel data show tolerances on pellet dishing of up to 0.4% reduction, a tolerance on pellet diameter of up to a 0.0005-inch increase in OD, and a tolerance of up to a 1.5 % increase in fuel pellet densification. If these three variations are taken together to maximize a MOX assembly fuel loading, this maximum loading can be used with the nominal pellet dimensions to determine a maximum fuel density. Maximizing the tolerances above yields up to 10.565 g $(U,, Pu)\ O_2$ /cc.

Storage Rack Cell Wall Thickness Manufacturing Uncertainty

Consistent with previous criticality calculations, a conservative maximum tolerance of 0.01 inches is applied to cell wall thickness.

Storage Rack Center-to-Center Cell Spacing Uncertainty

This uncertainty accounts for possible variations in the SFP rack geometry. The rack cells in the McGuire Region 1, McGuire Region 2, or Catawba SFPs are brought together as close as the tolerances allow on center-to-center spacing.

Boraflex Uncertainties

Applicable to the McGuire Region 1A and 2A SFP racks, the Boraflex uncertainties are bundled uncertainties that account for gaps in the Boraflex panels, axial and radial material shrinkage, and physical self-shielding effects. For MOX fuel these uncertainties are slightly less than those previously computed for LEU fuel. This is expected, since strong thermal absorbers such as Boraflex are less effective in the presence of MOX fuel.

The mechanical and calculational biases and uncertainties described above add between 0.02 and 0.03 Δk to the overall 95/95 k_{eff} for MOX fuel, depending on the SFP rack. This is comparable to the totals for LEU fuel. As discussed in Section A3.5, the MOX bias / uncertainty reactivity adders will be used for the combined MOX / LEU storage configurations considered here (McGuire Region 1A and Catawba SFPs).

In addition, the analyses conservatively assume plutonium isotopic fractions of 94% Pu-239, 5% Pu-240, and 1% Pu-241. The exact plutonium isotopics of the MOX fuel lead assemblies are not yet known, but are expected to be similar to the Table A3-2 values, and therefore less reactive than these assumed isotopics.

Using the evaluation procedure outlined at the beginning of this section, criticality calculations were carried out for various storage patterns in the McGuire and Catawba SFPs. Figure A3-5 shows the different types of patterns that have been qualified for storing MOX and MOX / LEU fuel in the McGuire and Catawba SFPs. These patterns are described below:

- Restricted / Filler Storage (McGuire Region 1A, Catawba SFP storage racks) – Fresh or irradiated MOX fuel assemblies qualify as Restricted assemblies in these storage regions. In addition, LEU fuel assemblies that exceed their LEU Unrestricted enrichment limit or do not meet the minimum required burnup for LEU Unrestricted storage can be stored as Restricted fuel in these storage regions. Note the low-reactivity "Filler" fuel assembly in this configuration must be a LEU fuel assembly, meeting the Filler minimum burnup requirements in Table A3-5.
- Checkerboard / Empty Storage (McGuire Regions 1B, 2A, and 2B) – Fresh or irradiated MOX fuel assemblies qualify as Checkerboard assemblies in these storage regions. LEU fuel assemblies that do not meet their enrichment / burnup limits to qualify as LEU Restricted fuel storage can be stored as Checkerboard fuel in these storage regions.

Table A3-5. Minimum LEU Filler Fuel Burnup Requirements for MOX Restricted Storage in the McGuire Region 1A and Catawba SFPs

Initial LEU Enrichment (wt % U-235)	McGuire Region 1A minimum LEU Filler burnup (GWD/MTU)	Catawba minimum LEU Filler burnup (GWD/MTU)
1.76	0.00	--
1.90	--	0.00
2.00	5.12	16.83
2.50	13.57	26.05
3.00	19.80	35.11
3.50	25.85	43.48
4.00	31.50	51.99
4.48	--	60.00
4.50	36.93	N/A
4.75	39.54	N/A

Both the normal and accident SFP conditions (described at the beginning of this section) were analyzed for all the MOX storage configurations considered in the McGuire and Catawba SFP storage regions. For normal conditions in the Catawba SFPs, the maximum no-boron 95/95 k_{eff} in the MOX / LEU Restricted / Filler configuration remained below 0.95. For normal conditions in the McGuire SFPs, the maximum no-boron 95/95 k_{eff} in the MOX and MOX / LEU configurations remained below 1.00. The highest boron concentration required for MOX fuel to reduce the 95/95 k_{eff} below 0.95 was still less than the 850 ppm that is allowed for LEU fuel. Therefore, all the normal storage regulatory subcriticality requirements were met for MOX fuel in the configurations shown in Figure A3-5.

For three of the accident conditions that needed to be evaluated for fuel storage (fuel assembly misload, dropped fuel assembly, and abnormal SFP temperature changes), the required boron concentrations to maintain the 95/95 k_{eff} below 0.95 are far below the minimum available in the SFP (2675 ppm for McGuire, 2700 ppm for Catawba), even with MOX fuel.

The other accident condition is the heavy load drop onto the SFP racks. The largest loads that can be carried over the McGuire and Catawba SFPs are the weir gates (see their locations in Figure A3-1). These 3000 – 4000 lb steel gates, if dropped onto the SFP racks, are capable of crushing up to seven (7) fuel assemblies. In accordance with NUREG-0612 (Reference 11), heavy load drop evaluations must assume the racks and the fuel assemblies within them are crushed uniformly to an optimum pin pitch. Figure A3-7 depicts the model for this weir gate drop in the McGuire (Region 1) SFP. The affected assemblies are crushed into a tighter and tighter configuration until maximum reactivity is achieved. Since the McGuire storage racks are already much more closely spaced than those in the Catawba SFPs, the crushed-rack evaluation with MOX fuel in the McGuire racks determined a worst-case 95/95 k_{eff} well below the 0.95 limit, with

2675 ppm boron in the SFP. The MOX fuel k_{eff} was similar to the highest LEU fuel k_{eff} in the McGuire weir gate drop analysis.

However, in the Catawba racks, the heavy load drop yields a maximum k_{eff} with MOX fuel that is significantly higher than that computed for LEU fuel. This is due to the fact, as discussed in Section A3.4, that the MOX fuel assembly is more undermoderated than the LEU fuel assembly. Therefore, the optimum crushed-rack pin pitch for MOX fuel is larger than that of the LEU fuel array, and therefore, the MOX fuel crushed cluster of fuel assemblies is still relatively "coupled" with the surrounding uncrushed rack cells. On the other hand, the more tightly crushed LEU fuel assemblies are effectively isolated from the rest of the rack. Nevertheless, the crushed-rack evaluation with MOX / LEU fuel in a Restricted / Filler configuration in the Catawba racks yielded a worst-case 95/95 k_{eff} that remained below the 0.95 limit, with 2700 ppm boron in the SFP.

A3.7. Conclusions

This evaluation has examined the feasibility of MOX fuel storage in the McGuire and Catawba SFPs. The reference MOX fuel design (the Mark-BW/MOX1) has been identified and evaluated in the SFPs at McGuire and Catawba. The results from all of these McGuire and Catawba SFP criticality analyses demonstrate that a reference MOX fuel design, with a maximum fissile plutonium concentration of 4.15 weight percent and a maximum U-235 enrichment of 0.35 weight percent, can be stored fresh or irradiated in the patterns shown in Figure A3-5, without any modifications to the existing SFP storage racks. This evaluation bounds the planned lead assembly fuel design of 4.37 weight percent total plutonium and 0.25 weight percent U-235 demonstrating that it also can be safely stored in the SFP storage racks.

A3.8. References

1. J. Coletta, *Evaluation of MOX Fuel Storage at McGuire and Catawba,*, American Nuclear Society 2001 Topical Meeting on Practical Implementation of Nuclear Criticality Safety, Reno, NV, November 13, 2001.

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

3. B. Murphy and R.T. Primm III, *Prediction of Spent MOX and LEU Fuel Composition and Comparison with Measurements,* Oak Ridge National Laboratory, May 2000.

4. O. Hermann, *Benchmark of SCALE (SAS2H) Isotopic Predictions of Depletion Analyses for San Onofre PWR MOX Fuel*, ORNL/TM-1999/326, February 2000.

5. D. Hollenbach and P. Fox, *Neutronics Benchmarks of Mixed-Oxide Fuels Using the SCALE/CENTRM Sequence,* ORNL/TM-1999/299, February 2000.

6. *Water-Reflected Mixed Plutonium-Uranium Oxide (20 wt. % Pu) Pins*, NEA/NSC/DCO/(95)03/VI, MIX-COMP-THERM-001, August 1996.

7. *Rectangular Arrays of Water-Moderated UO_2-2 wt. % PuO_2 (8% ^{240}Pu) Fuel Rods,* NEA/NSC/DCO/(95)03/VI, MIX-COMP-THERM-002, September 1997.

8. *Rectangular Arrays of Water-Moderated UO_2-6.6 wt.% PuO_2 Fuel Rods*, NEA/NSC/DCO/(95)03/VI, MIX-COMP-THERM-003, September 1997.

9. *Critical Arrays of Mixed Plutonium-Uranium Fuel Rods with Water-to-Fuel Volume Ratios Ranging from 2.4 to 5.6"* NEA/NSC/DCO/(95)03/VI, MIX-COMP-THERM-004, September 1997.

10. *Westinghouse Spent Fuel Rack Criticality Analysis Methodology*, WCAP 14416-NP-A, Revision 1, November 1996.

11. *Control of Heavy Loads at Nuclear Power Plants*, (Resolution of Generic Technical Activity A-36), NUREG-0612, U.S. Nuclear Regulatory Commission, July 1980.

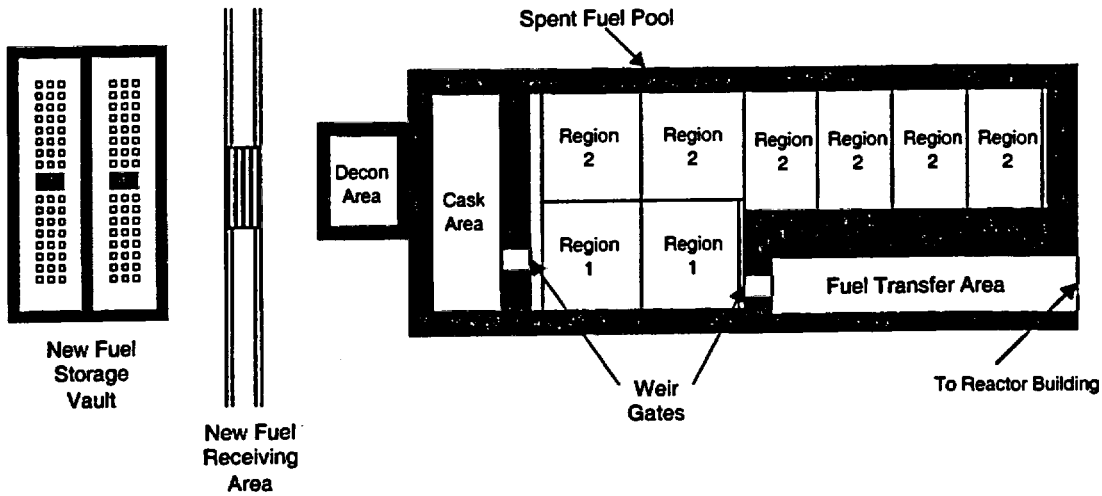


Figure A3-1. Overhead View of McGuire Fuel Building

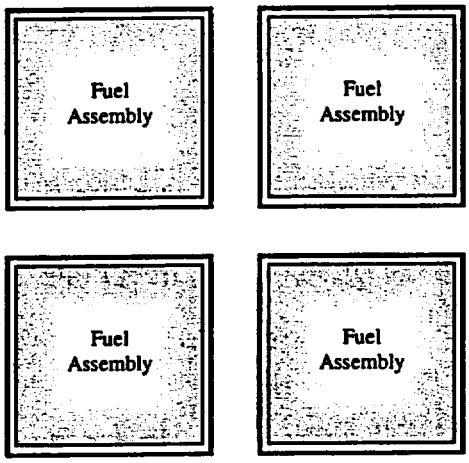


Figure A3-2. McGuire Spent Fuel Pool Region 1 Flux Trap Arrangement

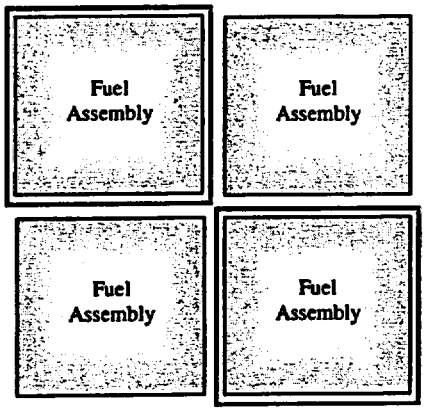


Figure A3-3. McGuire Spent Fuel Pool Region 2 "Cell / Off-Cell" Arrangement

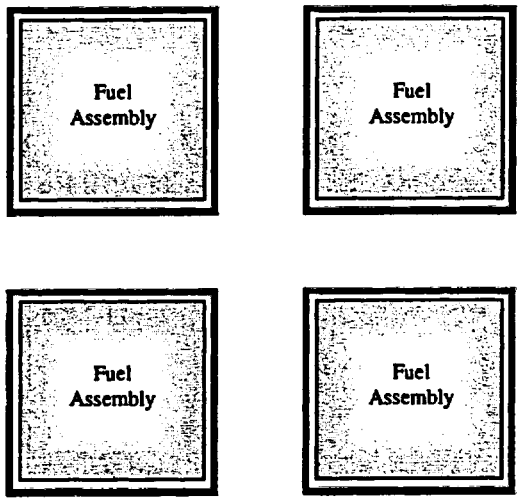
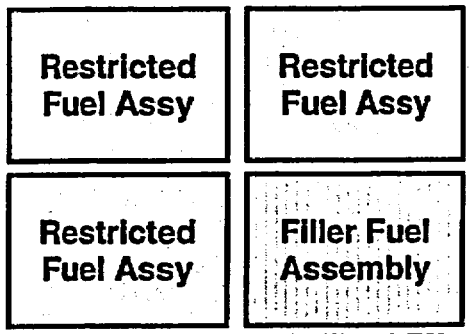
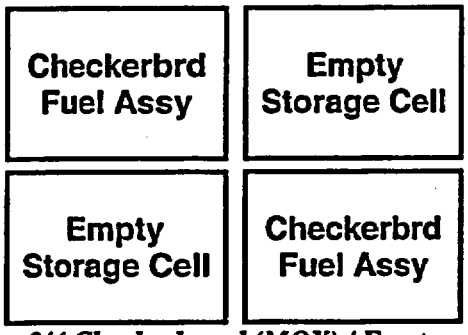


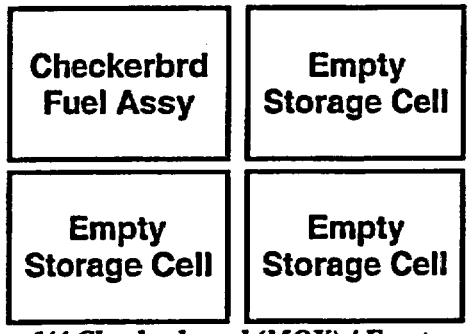
Figure A3-4. Catawba Spent Fuel Pool Flux Trap Arrangement



3/4 Restricted (MOX) / Filler (LEU) Storage Pattern – McGuire 1A and Catawba



2/4 Checkerboard (MOX) / Empty Storage Pattern – McGuire 1B and McGuire 2A



1/4 Checkerboard (MOX) / Empty Storage Pattern – McGuire 2B

Figure A3-5. Repeating Patterns for MOX Fuel Storage in the McGuire / Catawba Spent Fuel Pools

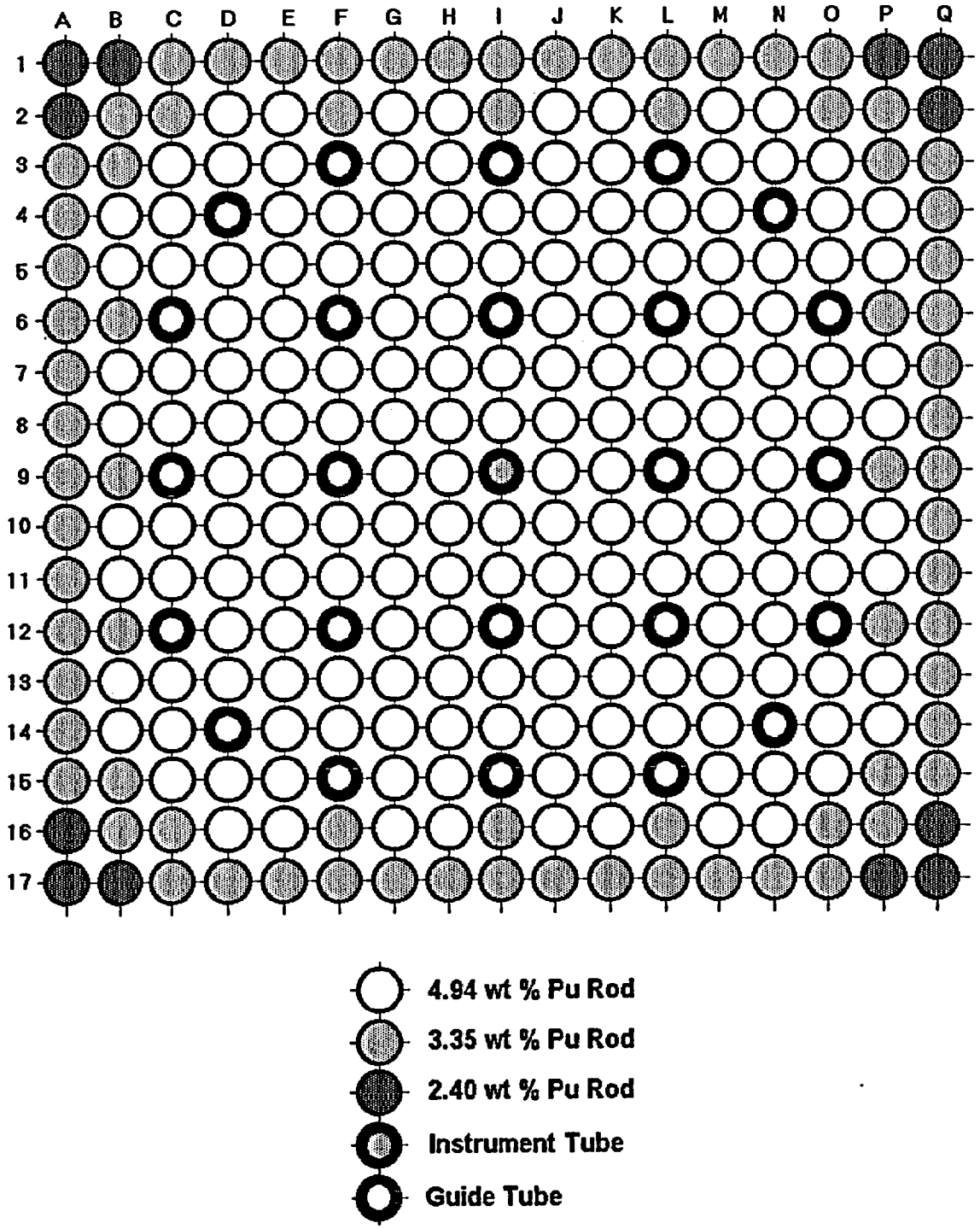


Figure A3-6. Fuel Rod Layout for MOX Fuel Lead Assembly (Average 4.37 wt % Total Plutonium Loading)

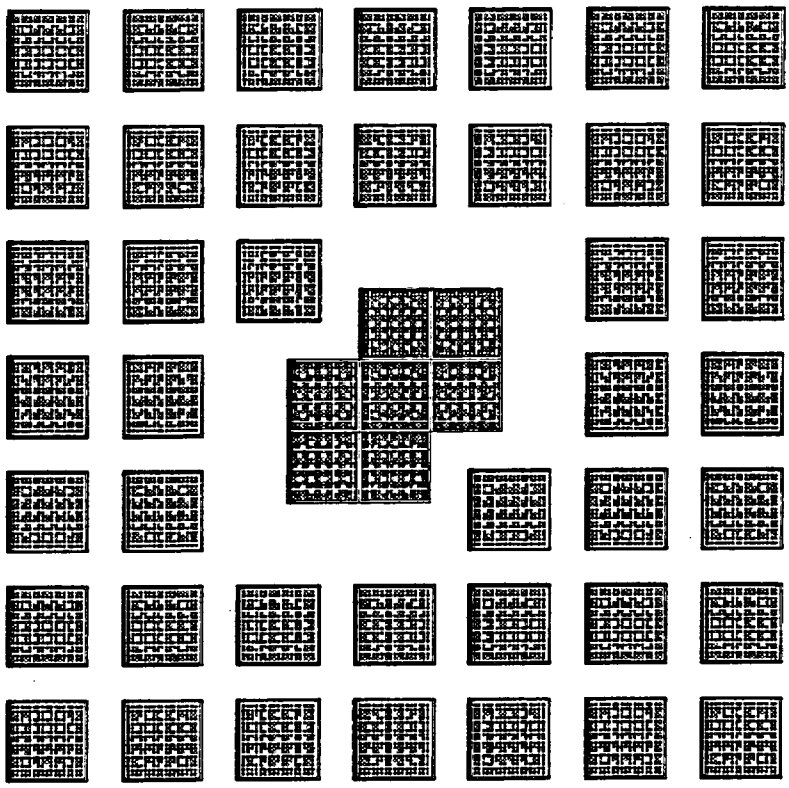


Figure A3-7. Worst-case Crushed Rack Configuration for Weir Gate Drop Event

Attachment 4

No Significant Hazards Consideration Evaluation

4. NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

4.1 BACKGROUND

The following analysis required by 10 CFR 50.91 is provided to justify the determination that the proposed license amendment and associated technical specification changes needed to support the introduction of mixed oxide (MOX) fuel lead assemblies onto the McGuire or Catawba site and into a reactor do not involve a significant hazards consideration. The standards in 10 CFR 50.92 are applied to the proposed changes to support this determination. The evaluation contained herein demonstrates that the proposed license amendments and associated technical specification changes do not:

a) Involve a significant increase in the probability or consequences of an accident previously evaluated;
b) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
c) Involve a significant reduction in a margin of safety.

In promulgating the final rule that implemented the current significant hazards consideration process, the NRC attempted to take some of the subjectivity out of the standards in 10 CFR 50.92 by listing specific examples of amendments that would be likely to involve significant hazards considerations.[1] Some of these examples are very straightforward; *e.g.*, renewal of an operating license or an increase in authorized maximum core power level are cited as examples of amendments that would involve a significant hazards consideration. Other examples are subject to interpretation; *e.g.*, "a *significant* (emphasis added) relaxation of the criteria used to establish safety limits" is an example that would also involve a significant hazards consideration. However, the word "significant" in the example is still open to interpretation.

A similar list of examples was provided for amendments that would *not* likely involve significant hazards considerations. The particular example of a license amendment that has been cited as possibly applicable to any license amendment request involving MOX fuel is:

> "A change resulting from a nuclear core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that the NRC has previously found such methods acceptable."

At the time that this rule was published, many of the core operating limits were contained as limiting conditions for operation in the technical specifications. The core design for each reload

[1] 51 Federal Register 7744, 7750 – 7751

resulted in slight changes in some of these limits and, as a result, many licensees were required to amend their technical specifications at each refueling. The above example provides clear guidance that license amendments and technical specification changes associated with a routine core reload do *not* involve a significant hazards consideration.

However, the converse of this example does not compel the conclusion that a significant hazards consideration is involved. It was the Commission's intention that any request for an amendment must be evaluated against the standards in the regulations, and that the examples simply provided supplementary guidance.[2]

There is also precedent for the requested licensing action. Previous applications to utilize MOX fuel assemblies in operating reactors have been submitted and approved by the NRC. In particular, Rochester Gas and Electric requested approval to receive, store, and utilize four MOX fuel assemblies in its Ginna reactor in an application dated December 14, 1979, transmitted by separate letter.[3] In its review of this application, the NRC staff produced two safety evaluation reports, the first[4] evaluating the receipt and storage of MOX fuel assemblies and the second[5] evaluating the use of the four MOX fuel assemblies. The conclusions in both reports were that no significant hazards considerations were involved with any of the activities. The standards in 10 CFR 50.92 that were used by the staff to evaluate the Ginna application have not changed. Thus, the precedent is relevant to Duke's application to irradiate four MOX fuel lead assemblies in the McGuire or Catawba reactors.

The following evaluation addresses each of the standards in 10 CFR 50.92 relative to the receipt, storage, and use of MOX fuel lead assemblies in either a McGuire or Catawba reactor.

4.2 EVALUATION

This section evaluates the proposed license amendment against the standards in 10 CFR 50.92.

4.2.1 Probability and Consequences Evaluation

The proposed license amendment to allow the use of MOX fuel lead assemblies does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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[2] *Id.* at 7750.

[3] Letter from Harry H. Voigt, LeBoeuf, Lamb, Leiby & MacRae to Mr. Harold R. Denton, U.S. Nuclear Regulatory Commission, dated December 20, 1979.

[4] Letter from Dennis L. Ziemann, U.S. Nuclear Regulatory Commission to Mr. Leon D. White, Jr., Rochester Gas and Electric Corporation, dated February 13, 1980.

[5] Letter from Dennis L. Ziemann, U.S. Nuclear Regulatory Commission to Mr. Leon D. White, Jr., Rochester Gas and Electric Corporation, dated April 15, 1980.

The "accidents" previously evaluated are described in the UFSAR and fall into one of the following four categories:

- Normal Operation and Operational Transients
- Faults of Moderate Frequency
- Infrequent Faults
- Limiting Faults

Inspection of the UFSAR descriptions reveals that the presence of MOX fuel lead assemblies could potentially impact the probability of occurrence for only two "accidents;" Radioactivity in Reactor Coolant Due to Cladding Defects and Fuel Handling Accidents. An evaluation of each of these events follows.

4.2.1.1 Radioactivity in Reactor Coolant Due to Cladding Defects Probability

Cladding defects are imperfections in the cladding material of a fuel assembly that allow fission products from the active fuel material to migrate to the reactor coolant. They can be caused by manufacturing defects that go undetected until the stresses of pressure, temperature, and/or irradiation eventually result in fuel cladding failure. This type of cladding failure occurs very infrequently in low-enriched uranium (LEU) fuel. The Mark BW design, which is the basis for the Mark BW/MOX1 design to be used in the MOX fuel lead assemblies, has experienced a failure rate of less than one per 100,000 rods, from all manufacturing related causes, since its inception in 1987. There is no reason to expect that the probability of this type of failure in a MOX fuel assembly will be any different than for a LEU fuel assembly because the probability of fuel failure due to these factors is no different for MOX fuel assemblies than for LEU fuel assemblies. The MOX fuel lead assemblies will be manufactured using the same quality standards that are used in the manufacture of LEU fuel, under a Quality Assurance program that conforms to 10 CFR 50 Appendix B. Likewise, the same operational procedures and precautions to preclude loose parts and debris in the reactor coolant will equally preclude fuel failures from these mechanisms for the MOX and LEU fuel assemblies.

Other mechanisms that could potentially cause fuel cladding failure are physical interaction of the cladding with loose debris in the reactor coolant system or corrosion product transport and buildup on cladding material. The design of both the current LEU fuel assemblies and the planned MOX fuel assemblies minimizes these types of interactions such that the probability of fuel failure is equally unlikely for both MOX and LEU fuel assemblies.

4.2.1.2 Fuel Handling Accident Probability

There is nothing in the physical design of a MOX fuel lead assembly that would make it more susceptible to a fuel handling accident than a LEU assembly. The physical dimensions are virtually identical, the difference in weight between a MOX assembly and a LEU assembly is less than 1%, and the top nozzle engages the manipulator crane and handling fixture in the same manner as LEU fuel.

The shipping container and associated unloading procedure for a fresh MOX fuel assembly are slightly different from that of a LEU fuel assembly but such differences do not result in a significant increase in the probability of an accident. The MOX fuel lead assembly shipping container is an end-loaded container with capacity for one fuel assembly as opposed to a LEU shipping container which is side loaded and has the capacity for two fuel assemblies. The MOX fuel assembly container is unloaded by uprighting the container, removing the closure lid, grappling the assembly with the Fuel Handling Tool, and lifting the assembly with a straight vertical lift out of the container. This is a straightforward lifting operation that will be practiced in a dry run involving a dummy fuel assembly, the MOX fuel shipping package, and specific fuel handling procedures. The same plant equipment will be used to grapple and lift a MOX fuel assembly that is used to lift a LEU fuel assembly. Once the MOX fuel lead assemblies are unloaded and placed into the spent fuel pool, subsequent handling operations are identical to LEU fuel handling. Thus, it is concluded that the probability of a fuel handling accident involving a MOX fuel assembly drop, either inside containment or inside the fuel building, is no different than for a LEU assembly.

The other scenarios considered as part of the fuel handling accident analyses are a weir gate drop into the spent fuel pool and a tornado-generated missile entering the spent fuel pool. There is no connection between the type of fuel assembly and the probability of occurrence of either of these accidents. The probability of a tornado missile entering the spent fuel pool is a natural event whose frequency of occurrence will not change with the storage of MOX fuel assemblies in the fuel pool. The probability of dropping a weir gate into the spent fuel pool is dependent on the reliability of handling fixtures, crane rigging procedures, and the number of handling operations, none of which will be affected adversely by the handling or presence of MOX fuel assemblies.

The conclusion is that amending the McGuire and Catawba licenses to allow the receipt, handling, storage, and use of MOX fuel lead assemblies does not result in a significant increase in the probability of occurrence of any accident previously evaluated in the UFSAR.

4.2.1.3 Consequences Evaluation

In order for a postulated accident to result in a significant increase in consequences, it must be shown that the accident results in a significant increase in dose to the public or to the control room operators. The UFSAR for both McGuire and Catawba contain the results of dose calculation for those accidents which have offsite or control room operator dose consequences. The dose consequences of these accidents were conservatively evaluated for a core consisting of four MOX fuel assemblies and 189 LEU fuel assemblies. The limiting design basis accidents for operations involving MOX fuel assemblies are the fuel handling accident and weir gate drop accident. The calculated dose consequences increased about 3%, for the fuel handling and weir gate drop accidents.

The insertion of MOX fuel lead assemblies would have a small effect on calculated radiation doses for accidents that are limited by departure from nucleate boiling (DNB). These include rod ejection, single rod withdrawal, and locked rotor accidents. For these design basis accidents, the increase in dose consequences due to MOX fuel lead assemblies was 1% or less. The calculated dose due to a postulated loss of coolant accident (LOCA) increased by less than 0.2%. These

small increases are not significant and the calculated doses are still well within the limits of 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criterion 19.

Based on this evaluation, it is concluded that amending the McGuire and Catawba licenses to allow the receipt, handling, storage, and use of MOX fuel lead assemblies does not result in a significant increase in the consequences of any accident previously evaluated in the UFSAR.

4.2.2 New or Different Accident Evaluation

The proposed license amendment to allow the use of MOX fuel lead assemblies will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The MOX fuel assemblies have similar mechanical and thermal-hydraulic properties to and nuclear characteristics only slightly different from the current LEU fuel assemblies. The use of MOX fuel lead assemblies does not involve any alterations to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. The existing design basis accidents described in the UFSAR remain appropriate and have been evaluated to demonstrate that there is no significant adverse safety impact related to the use of MOX fuel lead assemblies.

The main physical difference between a fresh MOX fuel assembly and a LEU fuel assembly is the presence of more radioactivity from the actinides in the MOX fuel matrix, resulting in a measurable dose rate in the immediate vicinity of a MOX fuel assembly. As a result, fresh MOX fuel is transported in a sealed leaktight shipping container by an enclosed tractor trailer truck. There are also differences in the fresh MOX fuel handling procedures, but these differences do not lead to a new or different type of accident.

A fuel handling accident involving a fresh MOX fuel assembly has potential for off-site dose consequences; however, the results of this fuel handling accident are bounded by the current analysis of a spent LEU fuel assembly drop accident. The calculated site boundary and control room dose consequences for a fresh MOX fuel handling accident are much less than the calculated doses for an accident involving a spent LEU fuel assembly and are well within the guidelines in 10 CFR 100. This accident does not involve a new release path, does not result in a new fission product barrier failure mode, and does not create a new sequence of events that would result in significant cladding failure. Therefore, this accident is not a new or different kind of accident.

In conclusion, amending the McGuire and Catawba licenses to allow the receipt, handling, storage, and use of MOX fuel lead assemblies does not create the possibility of a new or different kind of accident.

4.2.3 Margin of Safety Evaluation

The proposed license amendment to allow the use of MOX fuel lead assemblies will not involve a significant reduction in a margin of safety.

There are provisions in the McGuire and Catawba Technical Specifications that allow a "limited number of lead test assemblies" to be placed in "nonlimiting core regions." These provisions will not change and will apply to the planned use of MOX fuel lead assemblies. The effect of these provisions is to place restrictions on the allowable power distribution limits for a MOX fuel lead assembly.

The core design process assures that the limiting fuel rod in the core, whether LEU or MOX, has adequate nuclear power design limits under normal, transient, and accident conditions. If the core design process reveals unacceptable margin, adjustments are made to restore the needed margin. The operating limits are established in the Core Operating Limits Report to assure the design limits are not exceeded, thus assuring that adequate design margins for the fuel are maintained. This iterative design process is used to analyze the core containing MOX fuel lead assemblies to assure that there is no significant reduction in a margin of safety.

Because these lead assemblies will be located in nonlimiting locations *i.e.*, will have margin above that of the limiting assemblies, the results of safety analyses will likewise assure that appropriate margins to safety are maintained during transients and accidents.

4.3 CONCLUSION

The proposed receipt, handling, storage, and use of MOX fuel lead assemblies does not present any significant safety concerns as shown in the safety analysis in Attachment 3. This no significant hazards consideration evaluation demonstrates that amending the McGuire and Catawba licenses to allow the receipt, handling, storage, and use of MOX fuel lead assemblies does not:

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

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

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

Accordingly, it is concluded that the proposed changes will not involve a significant hazards consideration.

Attachment 5

Environmental Report

5. ENVIRONMENTAL REPORT FOR MOX FUEL LEAD ASSEMBLIES

5.1 IDENTIFICATION OF THE PROPOSED ACTION

The proposed action evaluated by this Environmental Report (ER) is the modification of the Catawba Nuclear Station and McGuire Nuclear Station operating licenses and Technical Specifications to allow the irradiation of up to four mixed oxide (MOX) fuel lead assemblies at either McGuire or Catawba. This proposed action includes the receipt, handling, storage, and use of MOX fuel lead assemblies at Catawba or McGuire Nuclear Station.

To evaluate the performance of MOX fuel, current plans are for four assemblies out of 193 total assemblies in the reactor core to be replaced with MOX fuel lead assemblies. These lead assemblies will be placed in non-limiting core locations. All of the lead assemblies will be irradiated for at least two fuel cycles. Poolside non-destructive post irradiation examination (PIE) will be conducted after each fuel cycle. Some or all of the lead assemblies will be irradiated for a third cycle. After the third cycle of irradiation, selected fuel rods will be removed and shipped to a hot cell PIE facility for destructive examination.

After removal from the reactor core, the irradiated MOX fuel assemblies will be stored along with low enriched uranium (LEU) spent fuel assemblies in the spent fuel pool adjacent to the reactor used for irradiation pending final disposal in a geologic repository.

5.2 ALTERNATIVE ACTIONS

5.2.1 No-Action Alternative

The no-action alternative is to deny the license amendment. The consequence of the no-action alternative is that without the use of lead assemblies to confirm MOX fuel performance characteristics neither Catawba Nuclear Station nor McGuire Nuclear Station would use any MOX fuel at the reactors, and would therefore not provide irradiation services to the Department of Energy (DOE) for the surplus plutonium disposition program. McGuire and Catawba would continue to operate with LEU fuel. For DOE to continue the plutonium disposition program, it would need to either find other reactors to use MOX fuel or use other methods to dispose of the surplus plutonium.

5.2.2 Other Alternatives

No alternatives other than the proposed action or no-action are viable. Consequently, no other alternatives were considered.

5.3 RELATED ACTIONS

The environmental impacts resulting from the fabrication of MOX fuel lead assemblies were discussed in Section 2.18.2 of the *Surplus Plutonium Disposition Final Environmental Impact Statement* (SPD EIS) (Reference 1). In the SPD EIS, five DOE sites were evaluated for the fabrication of up to ten lead assemblies: Savannah River Site (SRS), Los Alamos National Laboratory (LANL), Lawrence Livermore National Laboratory (LLNL), the Hanford site, and Idaho National Engineering and Environmental Laboratory (INEEL). Two DOE sites were evaluated for post-irradiation examination: Argonne National Laboratory – West at INEEL and Oak Ridge National Laboratory (ORNL). In the Record of Decision (ROD) associated with the SPD EIS, DOE selected LANL as the site to fabricate lead assemblies and ORNL as the site to conduct post-irradiation examination of selected fuel rods from the MOX lead assemblies.

Subsequent to the issuance of the ROD, DOE decided to revisit the decision to fabricate lead assemblies at LANL and evaluate other fabrication options. The primary option currently being pursued is the European fabrication (Eurofab) of the lead assemblies. Eurofab involves preparation (polishing) of PuO_2 powder at LANL with shipment of the polished powder to a European plant for pellet and fuel rod fabrication followed by fuel rod assembly. The completed MOX fuel assemblies will then be shipped back to either McGuire or Catawba to be irradiated as lead assemblies. The environmental impacts of the related actions associated with Eurofab that are within the scope of the National Environmental Policy Act (NEPA) were either evaluated, or are in the process of being evaluated, in DOE environmental documents as discussed in Sections 5.3.1 through 5.3.7.

Several of the related actions involve activities that are planned to take place outside the United States either in other countries or in the global commons. Executive Order (EO) 12114, *Environmental Effects Abroad of Major Federal Actions*, requires that when making decisions about major Federal actions that could have significant effects on the environment outside the geographical borders of the United States and its territories and possessions, Federal agencies take into consideration documents that describe and analyze the potential environmental impacts of those actions. Most nuclear actions, including the proposed fabrication of lead assemblies in Europe, are exempt from analysis under EO 12114 except for the potential impacts on the environment of the global commons outside the jurisdiction of any nation (e.g., the oceans or Antarctica).

5.3.1 Powder Preparation

This related activity, which consists of an aqueous polishing process to remove impurities from the PuO_2 powder, is being performed at LANL. The environmental impacts of this activity were addressed in the SPD EIS (Reference 1). These environmental impacts will not be addressed further in this ER.

5.3.2 Powder Shipment to Fabrication Facility

Shipment of the polished PuO_2 powder is a related activity involving intermodal transport from LANL to the selected European fuel fabrication facility. The PuO_2 powder will be shipped via truck to a United States port, via ship to a European port, and subsequently via land transport from the European port to the fuel fabrication facility. The environmental impacts of domestic transportation and overseas ship transport are being addressed by DOE in a supplemental environmental analysis in accordance with DOE implementing procedures (10 CFR Part 1021) and will not be addressed in this ER. Consistent with EO 12114, the land transport of polished PuO_2 powder in Europe will not be addressed in this ER.

5.3.3 Fabrication of MOX Fuel Lead Assemblies

The environmental impacts of fabricating fuel pellets, fuel rods, and MOX fuel lead assemblies in Europe is a related activity that is not included in this ER consistent with EO 12114..

5.3.4 MOX Fuel Lead Assembly Transportation

The completed MOX fuel lead assemblies will be shipped from the assembly plant in Europe to either the McGuire or Catawba site. The associated material[1] from the fuel fabrication process will be transported from the assembly plant in Europe to a DOE facility in the United States for storage. Transportation overland from the assembly plant to a European port is outside the scope of NEPA. The environmental impacts of overseas transport of lead assemblies from a European port to an east coast U.S. port will be addressed by DOE as part of the supplemental environmental analysis described in Section 5.3.1. It should be noted that the SPD EIS evaluated the impacts of transporting up to eight MOX fuel lead assemblies a distance of 3,100 miles. These environmental impacts bound the impacts of shipping four MOX fuel lead assemblies from an East Coast port to McGuire or Catawba because 1) transportation impacts are proportional to distance and the distance from the East Coast port to McGuire or Catawba is much less than 3,100 miles, and 2) the amount of material to be shipped is less than what was evaluated by DOE. These environmental impacts will not be addressed further in this ER.

5.3.5 Transport and Post Irradiation Examination of Irradiated Fuel Rods

The SPD EIS evaluated the environmental impacts associated with the transport of irradiated fuel rods (from irradiated MOX fuel lead assemblies) from McGuire or Catawba to a hot cell PIE facility.[2] The SPD EIS also evaluated the environmental

[1] Associated material refers to plutonium-bearing archive pellets, fuel rods, and/or any remaining scrap material.

[2] The SPD EIS evaluated transportation of irradiated fuel rods from McGuire to Argonne National Laboratory- West, in Idaho, as a limiting case. This evaluation bounds the impacts of transportation of

impacts of conducting the post-irradiation examination, as well as the storage and disposal of the fuel rods after the examination. These environmental impacts will not be addressed further in this ER.

5.3.6 Transport and Disposal of Spent MOX Fuel

The transportation and disposal of spent MOX fuel lead assemblies from McGuire or Catawba to a geologic repository are not part of this proposed licensing action. The environmental impacts associated with transport and disposal of spent MOX fuel are essentially the same as those for uranium fuel. These impacts are addressed in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (S&D PEIS) (Reference 2) and the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (Reference 3). These environmental impacts will not be addressed further in this ER.

5.3.7 Future Batch Use of MOX Fuel

The proposed future use of batch quantities of MOX fuel in the Catawba Nuclear Station and McGuire Nuclear Station reactors will require separate license amendments. The environmental impacts of batch use of MOX fuel will be evaluated as part of any future batch license amendment requests. These environmental impacts will not be addressed in this ER.

5.4 BACKGROUND

On September 27, 1991, President George H. Bush announced the end of the 42-year Cold War with the Soviet Union. This announcement eventually led to a determination that our nuclear weapons stockpile needed to be reduced, resulting in surplus plutonium and surplus highly-enriched uranium (HEU). In 1992, General Brent Scowcroft, then National Security Advisor to President Bush, requested the National Academy of Science Committee on International Security and Arms Control (NAS CISAC) to perform a study of the management and disposition options for surplus weapons-usable plutonium. The request was later confirmed by President Clinton when he assumed office in January 1993. The results of the CISAC study were published in *Management and Disposition of Excess Weapons Plutonium* (NAS 1994).

The CISAC recommended, among other actions, that the United States and Russia pursue a long-term plutonium disposition option that results in a form from which the plutonium would be as difficult to recover for weapons use as the larger and growing quantity of plutonium in commercial spent fuel. This recommendation became known as the Spent

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irradiated MOX fuel rods from either McGuire or Catawba to Oak Ridge National Laboratory, which is the preferred PIE facility.

Fuel Standard. The CISAC report discussed two approaches that could be used to achieve the Spent Fuel Standard. One approach is fabrication and use of MOX fuel in nuclear reactors. The plutonium in the MOX fuel would be irradiated and become part of the spent fuel that will be disposed in a geologic repository. The second approach is incorporation of plutonium in a vitrified HLW matrix (*i.e.*, immobilization) with disposition in the same geologic repository. The study noted that there may be some public opposition to the proven MOX fuel option. The study also noted the existence of technical difficulties and longer implementation time with the immobilization option.

In December 1996, DOE published the S&D PEIS (Reference 3). This document analyzed the potential environmental consequences of alternative strategies for the long-term storage of weapons-usable plutonium and the disposition of weapons-usable plutonium that has been or may be declared surplus to national security needs. The ROD for the S&D PEIS, issued in January 1997, outlined DOE's decision to pursue a hybrid approach to plutonium disposition that would make surplus weapons-usable plutonium inaccessible and unattractive for weapons use. DOE's disposition strategy, consistent with the preferred alternative analyzed in the S&D PEIS, allowed for both the immobilization of some (and potentially all) of the surplus plutonium and irradiation of some of the surplus plutonium as MOX fuel in existing domestic, commercial reactors.

In September 2000, the governments of the United States and the Russian Federation signed the "Agreement Between the Government of the United States of America and the Government of the Russian Federation Concerning the Management and Disposition of Plutonium Designated as No Longer Required for Defense Purposes and Related Cooperation" (White House 2000). The agreement commits the United States to disposal of 28.2 tons (25.57 metric tons) of plutonium through conversion to MOX fuel and subsequent irradiation in power reactors.

In January 2000, DOE issued the SPD EIS Record of Decision (DOE 2000), which contained the following statement:

> "The fundamental purpose of the program is to ensure that plutonium produced for nuclear weapons and declared excess to national security needs (now and in the future) is never again used for nuclear weapons. .....The Department has selected the Savannah River Site in South Carolina as the location for all three disposition facilities. Based upon this selection, the Department will authorize DCS to fully implement the base contract."

In April 2002 DOE issued a revised ROD that cancelled the immobilization portion of the disposition strategies announced in previous decisions. DOE determined that in order to be able to proceed using available funds, only one disposition strategy should be pursued. Because selection of an immobilization-only approach would lead to loss of Russian interest in and commitment to surplus plutonium disposition, DOE decided that the MOX approach was preferable.

5.5 NEED FOR THE PROPOSED ACTION

The proposed action is essential to the successful implementation of the joint United States-Russian plutonium disposition agreement. Before batch quantities of MOX fuel can be irradiated in the Catawba Nuclear Station or McGuire Nuclear Station, expected performance characteristics must be verified using lead assemblies.

DOE has previously determined that there is a clear need for the MOX Fuel Program. As stated in the SPD EIS:

> "The purpose of and need for the proposed action .... is to reduce the threat of nuclear weapons proliferation worldwide by conducting disposition of surplus plutonium in the United States in an environmentally safe and timely manner. Comprehensive disposition actions are needed to ensure that surplus plutonium is converted to proliferation-resistant forms."

In order to meet the requirements for the Spent Fuel Standard, the decision was made to fabricate weapons grade plutonium into MOX fuel assemblies and irradiate the MOX fuel assemblies in a reactor creating spent MOX fuel. Prior to irradiating significant quantities of MOX fuel, Duke must first confirm the performance of MOX fuel lead assemblies in its reactors. It is necessary for the Nuclear Regulatory Commission (NRC) to approve the proposed action in order for Duke to irradiate the MOX fuel lead assemblies. A successful lead assembly program would subsequently allow Duke to proceed with obtaining the necessary NRC approvals for batch implementation.

5.6 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

5.6.1 Plant Effluents

There are no anticipated changes to water use or non-radiological liquid discharges as a result of this action. The proposed action does not involve changes to ground-water withdrawals or non-radiological discharges. There are no anticipated changes to non-radiological emissions as a result of this action. Consequently, there are no anticipated changes in the types or amounts of plant effluents resulting from the use of four MOX fuel lead assemblies.

5.6.2 Impacts to Human Health

Slight increases in occupational exposure will result from handling of MOX fuel during receipt and handling operations. The increase in dose is due to a higher dose rate associated with a fresh MOX fuel assembly as compared to a fresh LEU fuel assembly. Total neutron and gamma dose rate at 10 cm from the face of a fresh MOX fuel assembly

averages about 6 mrem/hour, falling off to about 1.8 mrem/hr at 100 cm. This is a relatively low radiation field; however, it is larger than that associated with a LEU fuel assembly, which has virtually no radiation field at these distances. The initial receipt and handling activities for one MOX fuel lead assembly could result in a conservatively estimated total occupational dose in the range of .020 to .042 person-rem. Application of ALARA[3] principles could result in lower doses. Radiation exposures of this magnitude are well within occupational exposure limits and will have no impact on worker health.

No detectable increase in public dose during normal operations is anticipated due to the presence of four MOX fuel assemblies at McGuire or Catawba. Use of the lead assemblies in the reactor core will not change the characteristics of plant effluents or water use. During normal plant operation, the type of fuel material will have no effect on the chemistry parameters or radioactivity in the plant water systems. The fuel material is sealed inside fuel rods that are seal-welded and leaktight. Therefore, there would be no direct impact on plant effluents.

Large quantities of MOX fuel can have an effect on the nuclear properties of the reactor core that could impact the chemistry parameters of the reactor coolant, specifically, soluble boron concentration. However, with only four MOX fuel assemblies, there will be no discernible change in the core nuclear properties and therefore no change in coolant chemistry that could potentially impact plant discharges.

In the event of a leaking fuel rod, an uncommon occurrence, there will be increased activity in the primary coolant. While there is a difference in the radioactive isotopic inventory between an irradiated MOX fuel assembly and an irradiated LEU fuel assembly, this would not translate into a significant difference in plant effluents. For both fuel types, plant process systems would limit the release of radioactive isotopes through holdup, filtering, demineralization, and decay such that there would be no significant difference in radioactive effluents between the two fuel types. Plant releases will comply with all regulatory limits with MOX fuel lead assemblies in the reactor, such that there will be no impact on public health and safety.

The four irradiated MOX fuel lead assemblies will be stored with other spent fuel assemblies until they are shipped to a geologic repository for disposal. The long term MOX fuel decay heat is slightly greater than the decay heat produced by a LEU spent fuel assembly. Figure 5-1 provides a comparison of the long term decay heat power between a MOX fuel assembly and a LEU fuel assembly expressed as a ratio. Although not depicted on this figure, it is important to note that the absolute value of decay heat decreases over time. However, with only four MOX fuel lead assemblies, the long term increase in heat load on the spent fuel pool during storage will be inconsequential.

[3] ALARA – As low as reasonably achievable; refers to a radiological control program to minimize radiation exposure to workers through application of procedural controls, engineered features, and experience.

5.6.3 Impacts of Postulated Accidents

This section summarizes the evaluation of potential accidents at McGuire or Catawba nuclear stations involving MOX fuel lead assemblies. The information presented in this section is based on the Safety Analysis section contained in Attachment 3 of the license amendment request. The analyses used conservative assumptions and produced comprehensive, bounding results. The evaluations of events show that the environmental risk from a facility accident is low.

5.6.3.1 Design Basis Accidents

Based on a review of the various accident scenarios in the respective Safety Analysis Reports for McGuire and Catawba, it was determined that MOX fuel lead assemblies had the most impact on the results of the fuel handling and weir gate drop accidents. For these particular accidents, comparisons of the dose consequences between a LEU fuel assembly and a MOX fuel assembly were made. For all cases evaluated, the thyroid doses were limiting. For the MOX fuel cases, the thyroid radiation doses were calculated to be about 3% higher than the LEU fuel cases. Results remain within regulatory limits.

For all other design basis accidents, the increase in dose consequences for a reactor core with four MOX fuel assemblies and 189 LEU fuel assemblies was less than 1%. The limiting accident is the locked rotor accident, which has a calculated dose about 0.4% greater than the corresponding accident dose for an all-LEU core. The other two potentially limiting accidents are the rod ejection accident and loss of coolant accident (LOCA). The calculated dose increases for these two accidents were 0.2% and 0.1%, respectively.

The consequences of a drop of a fresh MOX fuel assembly in air were also calculated. The analysis assumed the drop of a complete MOX fuel assembly with resultant damage to the assembly. Specifically, cladding damage was postulated to occur and fuel pellet damage was assumed, which resulted in the airborne release of a respirable fraction of particulate nuclides. The activity was then transported to a receptor at the site boundary with resulting exposure from the particulate activity. Exposure was computed using Federal Guidance Report 11 (Reference 4) conversion factors. Even using extremely conservative assumptions with no credit for ventilation system filters, the resulting calculated dose was less than 0.4 rem, which is well below regulatory limits for design basis accidents.

5.6.3.2 Severe Accidents

Duke uses probabilistic risk assessment (PRA) analyses to evaluate the risk to public health and safety due to operation of its nuclear plants. PRA analyses quantify the probability and consequences of severe accidents that involve postulated core melt and

containment failure events. Key considerations in PRA analyses are equipment requirements to prevent core melt (success criteria); ice melt times, containment pressurization rates, and potential containment failures (containment performance); and doses to the public (offsite consequences). The attributes of MOX fuel that impact these areas are fundamentally similar to uranium fuel, as discussed below.

- Plant configuration: The plant equipment, including passive and active safety systems, is unaffected by the type of fuel in the core.
- Fuel characteristics: As discussed in Reference 5, MOX fuel is fundamentally similar to conventional LEU fuel. Both MOX and LEU fuel consist of sintered ceramic pellets with very similar material properties. Both MOX and LEU fuel are clad with zirconium alloy metal. The MOX fuel assembly mechanical design is very similar to the resident LEU fuel.
- Decay heat: Short term decay heat from MOX fuel is slightly lower than decay heat from an equivalent LEU fuel assembly during the time interval of concern for core melt accidents.
- Radionuclide inventory: Irradiated MOX fuel has a somewhat different radionuclide inventory than LEU fuel. For fission products, the same radionuclides are present, in generally similar amounts, as discussed in Section 3.7.3 (Radiological Consequences of Design Basis Accidents). Irradiated MOX fuel contains significantly more actinides (plutonium, neptunium, americium, and cerium) than equivalent uranium fuel. However, these elements do not transport to the environment to the same extent as lighter radionuclides. Therefore, the dose consequences of the actinides are small relative to radionuclides like iodine and cesium.

Due to the fundamental similarity between MOX fuel assemblies and uranium fuel assemblies, four MOX fuel lead assemblies will have no appreciable impact on the thermal-hydraulic response of the core. Success criteria are dominated by the plant configuration and core decay heat, and four MOX fuel lead assemblies will therefore have no adverse impact in this area. Containment performance is also dominated by the plant configuration and core decay heat; again, no adverse impact on severe accident consequences would be expected.

The largest impact of MOX fuel use will derive from differences in the original radionuclide inventory. The most important radionuclides from an offsite dose perspective are volatile fission products like cesium and iodine. Those radionuclide inventories are generally similar for MOX and LEU fuel. Higher actinide concentrations in MOX fuel have the potential to cause higher offsite doses, but this is a smaller effect. In Appendix K.7 of Reference 1 DOE evaluated several severe accident sequences at McGuire and Catawba both for cores containing all-uranium fuel and cores containing approximately 40% MOX fuel. In the DOE analyses, offsite consequences from severe accidents ranged from minus 4% to plus 14% compared to LEU fuel. For cores containing four MOX fuel lead assemblies (2% of the total number of fuel assemblies), the potential impact on offsite consequences from severe accidents would range from less

than minus 1% to a maximum of plus 0.3% compared to LEU fuel. Accordingly, it is concluded that operation with four MOX fuel lead assemblies will have no significant impact on public health risk at either McGuire or Catawba.

5.6.4 Impacts to Fuel Cycle

The proposed action will involve the use of four MOX fuel lead assemblies in a commercial reactor. This is a small percentage of the number of assemblies reloaded in the United States each year, such that the impact of the proposed action on the remainder of the fuel cycle will be negligible. In addition, there are no extra mining, milling or enrichment related activities required in connection with the proposed use of MOX fuel lead assemblies. As noted in Sections 5.3.1 through 5.3.6, the impacts associated with fabrication and transportation of these fresh MOX fuel assemblies and the transportation and disposal of these spent MOX fuel assemblies are evaluated in other environmental impact statements, where appropriate.

In sum, inasmuch as these four MOX fuel lead assemblies represent a small fraction of a fuel batch that will be loaded into the McGuire or Catawba reactor at the same time (4 of about 80 assemblies), and because the environmental impacts associated with their fuel cycle and transportation are small, the issuance of the license amendments sought will not significantly alter the overall environmental impact as previously reviewed by the NRC.

5.6.5 Impacts to Decommissioning

The proposed action will involve the irradiation of a maximum of four lead assemblies in a McGuire or Catawba reactor. These will replace low-enriched uranium (LEU) assemblies. The radiological characteristics of spent MOX fuel assemblies are similar to those of spent LEU assemblies. There is no anticipated change to decommissioning costs, waste generation or environmental impact resulting from the proposed action.

5.7 COMPARISON OF ALTERNATIVES

The only viable alternative is the no-action alternative. The no-action alternative would eliminate the use of MOX fuel lead assemblies in the Catawba Nuclear Station and McGuire Nuclear Station. In this instance both Catawba Nuclear Station and McGuire Nuclear Station would continue operations using LEU fuel. Both the proposed action and no-action alternative would have negligible impact on the ecology surrounding either the Catawba Nuclear Station or the McGuire Nuclear Station. Irradiation of MOX fuel lead assemblies could result in a slight increase in worker dose above what would be anticipated for the no-action alternative. There would be no measurable increase in dose to the public resulting from normal operations using MOX fuel lead assemblies

For postulated design basis accidents, potential dose to the public for the most limiting accident would be about 3% above what would be anticipated for the no-action alternative.

5.8 SUMMARY AND CONCLUSIONS

The United States nuclear non-proliferation policy involves the conversion of surplus weapons-usable plutonium to MOX fuel, then irradiating that fuel in a commercial reactor to meet the Spent Fuel Standard proposed by the National Academy of Sciences. Duke will provide the fuel irradiation service to the DOE using the reactors at the Catawba and McGuire Nuclear Stations. Irradiation of lead assemblies is necessary to evaluate the performance characteristics of the MOX fuel. The proposed action is to amend the licenses and technical specifications for the Catawba and McGuire Nuclear Stations to allow the irradiation of four MOX fuel lead assemblies.

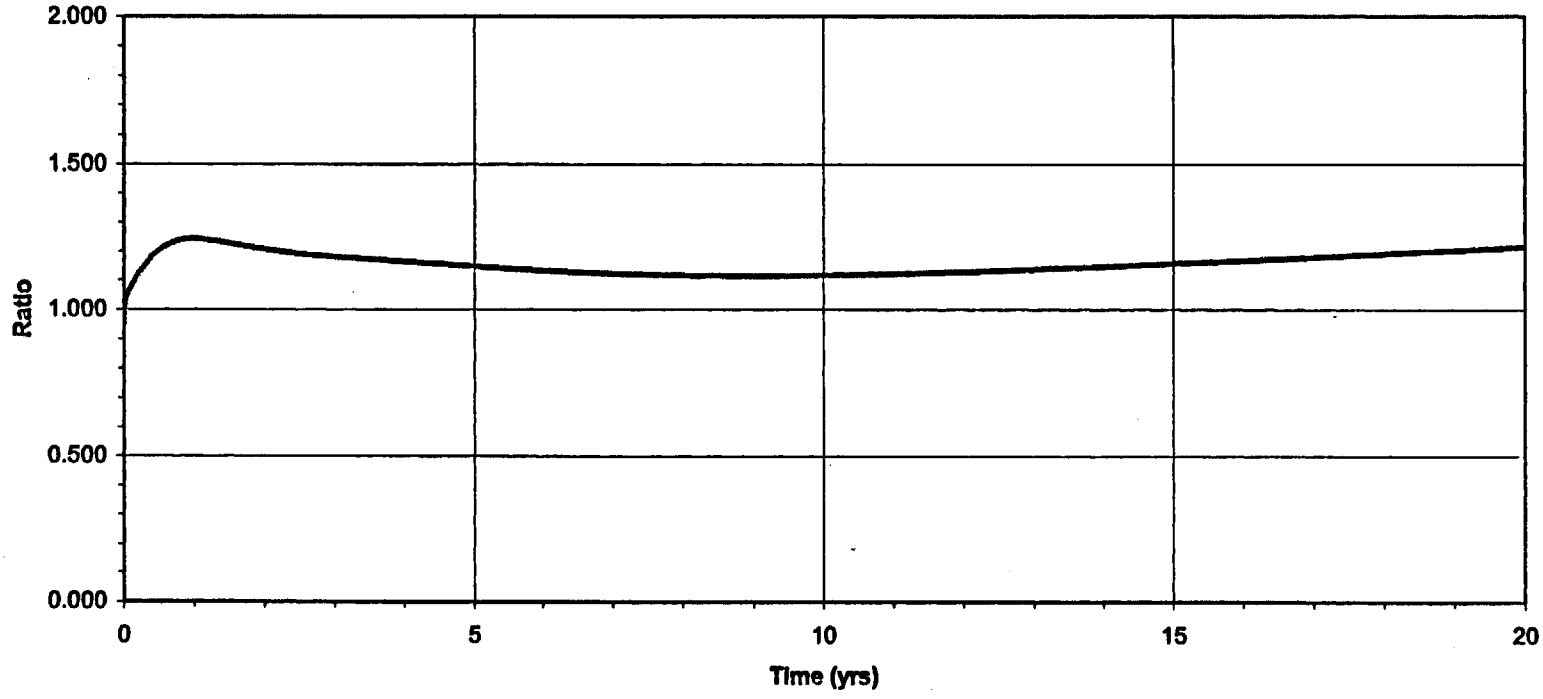
There are no anticipated environmental impacts in the vicinity of either McGuire or Catawba Nuclear Station resulting from the approval and implementation of the proposed licensing action. Normal operations could result in a very small increase in worker dose resulting from fuel receipt and handling activities. No measurable increase in public dose is expected during normal operations. The most limiting potential accident scenarios could result in an increase in public dose about 3% above what would be expected for similar LEU accident scenarios.

Approval of the proposed action would result in no significant impact on the environment.

Figure 5-1

Long Term MOX Fuel Decay Heat Comparison

Typical MOX/LEU Fuel Decay Heat Ratio



5.9 REFERENCES

1 DOE (U.S. Department of Energy), *Surplus Plutonium Disposition Final Environmental Impact Statement*, DOE/EIS-0283, Office of Fissile Materials Disposition, Washington, D.C., November 1999.

2 DOE, (U.S. Department of Energy) *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement*, DOE/EIS-0229, Office of Fissile Materials Disposition, Washington, D.C., December 1996.

3 DOE (U.S. Department of Energy), *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*, DOE/EIS-025F, Office of Civilian Radioactive Waste Management, North Las Vegas, NV, February 2002.

4 EPA (Environmental Protection Agency) Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion*, EPA-520/1-88-020, September 1988.

5 Framatome ANP, *MOX Fuel Design Report*, BAW-10238(NP), Revision 0, March 2002.

Attachment 6

Request for Exemptions

Attachment 6
Request for Exemptions from
Certain Provisions of 10 CFR Part 50

I. INTRODUCTION

Duke Energy Corporation (Duke) hereby files this request for exemptions from portions of certain regulations contained in 10 CFR Part 50, in conjunction with the receipt, handling, storage and use of mixed oxide (MOX)fuel at certain of its nuclear generating facilities licensed by the U.S. Nuclear Regulatory Commission (NRC or Commission). This exemption request accompanies and is filed in conjunction with Duke's license amendment request whose focus is the utilization of four MOX fuel lead assemblies at either McGuire or Catawba Nuclear Station. Each element of the Commission's standards for the issuance of exemptions, set forth in 10 CFR § 50.12, is discussed below and a demonstration provided that such standard is met.

These exemptions are necessitated by several factors. In several instances, NRC regulations assume the use of fuel rods containing only low enrichment uranium (LEU) fuel and not MOX fuel and have not been updated to permit the use of advanced types of cladding, either for conventional fuel or for MOX fuels. Other regulations currently refer specifically to uranium oxide (UO_2) fuel use, but are generally applicable to similar types of fuel. In each of these situations, an exemption is required from certain provisions in 10 CFR Part 50—namely, portions of Sections 50.44, 50.46 and Appendix K to 10 CFR Part 50.

This request addresses individually each exemption sought. The discussion of each proposed exemption cites the applicable NRC standard for issuance of an exemption from the affected part of the regulations, identifies the specific regulations (or portions thereof) for which an exemption is being sought, and provides detail concerning the extent and temporal scope of the proposed exemptions. A justification for each exemption request is also provided, along with a demonstration that the criteria for issuance of an exemption are met and that the Commission's underlying reason for promulgation of the regulation is satisfied. As noted previously, where Duke is taking additional actions, these are described and justified.

II. REQUEST FOR EXEMPTION FROM FUEL CLADDING REQUIREMENTS REFLECTED IN 10 CFR §§ 50.44, 50.46, AND PART 50, APPENDIX K

Duke requests an exemption from the requirements of 10 CFR 50.44(a), 10 CFR 50.46(a)(1), and 10 CFR Part 50 Appendix K such that explicit consideration of the M5™ cladding planned for use with MOX fuel assemblies is not required in order to be in compliance with these regulations.

A. NRC Standard for Issuance of Exemptions under 10 CFR Part 50

10 CFR § 50.12(a)(1)-(2) provides that upon application by any interested person, the NRC may grant an exemption from the requirements in NRC regulations found in Part 50 if the exemption is authorized by law, if granting the exemption will not present "an undue risk to the public health and safety" and is "consistent with the common defense and security," and if "special circumstances" are shown to be present. Special circumstances are present, for example, when application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. See 10 CFR § 50.12(a)(2)(ii). All of these criteria are met with respect to the exemption request described in Section II, below.

B. Regulatory Provisions from which Exemption Is Needed

10 CFR § 50.44(a) (*Standards for combustible gas control system in light water cooled power reactors)* requires pressurized water reactors (PWRs) "fueled with oxide pellets within cylindrical zircaloy or ZIRLO cladding" to include means for control of hydrogen gas that may be generated following a postulated loss-of-coolant accident (LOCA) by metal-water reactions involving the fuel cladding and the reactor coolant, by radiolytic decomposition of the reactor coolant, and by corrosion of metals.

10 CFR § 50.46(a)(1) *(Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors)* requires each BWR and PWR "fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding" to be provided with an emergency core cooling system (ECCS) designed so that its calculated cooling performance following a postulated LOCA meets certain criteria in Section 50.46(b). ECCS cooling performance must be calculated in accordance with "an acceptable evaluation model," for a number of postulated LOCAs. The evaluation model must include sufficient justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. Alternatively, an ECCS evaluation model may be developed in conformance with "the required and acceptable features" of the ECCS evaluation models set forth in Appendix K to 10 CFR Part 50. (See Section 50.46(a)(1)(i).)

Consistent with Sections 50.44 and 50.46, Appendix K to 10 CFR Part 50 (*ECCS Evaluation Models*) also reflects certain assumptions regarding the use of fuel cladding that is either zircaloy or ZIRLO.

C. Circumstances Requiring an Exemption from the Fuel Cladding Assumptions in 10 CFR §§ 50.44, 50.46 and Appendix K to 10 CFR Part 50

Duke plans to utilize MOX fuel at McGuire and Catawba as a portion of the units' cores as part of the its support of the DOE's mission to dispose of surplus weapons grade plutonium. The proposed use of McGuire and Catawba as "mission reactors" for this purpose requires that Duke obtain from the NRC an exemption from the assumption (and, thus, the implicit requirement) in Section 50.44, Section 50.46, and Appendix K to Part 50 that zircaloy or ZIRLO fuel cladding will be used in every commercial nuclear reactor. In particular, the MOX fuel pellets to be used in the four lead assemblies by Duke at McGuire or Catawba will be enclosed in M5™ cladding, an advanced cladding which has a number of properties that enhance cladding performance. The chemical composition of M5™ cladding differs somewhat from that of both zircaloy and ZIRLO. Because the use of M5™ cladding for the MOX fuel is not consistent with 10 CFR

Sections 50.44, 50.46, and 10 CFR Part 50, Appendix K as written, Duke is requesting an exemption from these fuel cladding requirements.

D. Exemption Required in Lieu of Complying with Cladding Requirements in 10 CFR Sections 50.44, 50.46, and Part 50, Appendix K As Written

Duke requests an exemption from the requirements of Section 50.44, Section 50.46, and Appendix K to 10 CFR 50, as those requirements relate to the fuel cladding to be used on the MOX fuel lead assemblies to be used at the McGuire or Catawba facilities. This exemption is requested to specifically permit the use of the Framatome Advanced Nuclear Power (F-ANP) M5™ advanced alloy as an acceptable fuel cladding material for the MOX fuel lead assemblies to be used at these facilities.

E. Basis and Justification for Grant of Exemption

As shown below, all of the requirements of 10 CFR 50.12 for the issuance of an NRC exemption have been satisfied.

1. This exemption request is authorized by law

As required by 10 CFR 50.12(a)(1), this requested exemption is "authorized by law." The selection of a specific cladding material in 10 CFR 50.44 and 50.46, and implied in Appendix K to Part 50, was adopted at the discretion of the Commission consistent with its statutory authority. No statute required the NRC to adopt this specification. Additionally, the NRC has the authority under Section 50.12 to grant exemptions from the requirements of Part 50 upon a showing of proper justification by the applicant. Further, it should be noted that, by submitting this exemption request, Duke does not seek an exemption from the acceptance and analytical criteria of 10 CFR 50.44, 50.46 and Appendix K to 10 CFR 50. The intent of the request is solely to allow the use of existing criteria set forth in these regulations for application to the M5™ cladding material.

2. Granting this exemption request will not present an undue risk to public health and safety

As demonstrated below, the acceptance criteria of 10 CFR 50.44 and 50.46 are applicable to M5™ cladding. In addition, the Baker-Just equation, required by Appendix K to 10 CFR Part 50 to be used to predict the clad oxidation rate, is also shown to conservatively predict the oxidation rate for M5™ cladding. The impact of M5™ cladding on LOCA analysis is specifically evaluated in Framatome ANP topical report BAW-10227P-A.[1] This report demonstrated and NRC accepted that the acceptance criteria in 10 CFR 50.46 and 10 CFR Appendix K are valid for M5™ cladding. The plant safety analyses will assure that these acceptance criteria are met following the implementation of the use of M5™ cladding. Preliminary calculations in Attachment 3, Section 3.7.1 of this license amendment request have demonstrated acceptable LOCA results for MOX fuel with M5™ cladding. The MOX fuel using M5™ cladding will be evaluated using NRC approved analytical methods and will specifically address the cladding material properties for M5™ cladding. The safety analysis for McGuire and Catawba will be supported by the applicable Technical Specifications. Fuel assemblies utilizing M5™ cladding will be operated in accordance with operating limits as specified in the Technical Specifications. Thus, the granting of this exemption request will not pose an undue risk to public health and safety.

3. Granting this exemption request is consistent with common defense and security

This exemption request is only to allow the application of the aforementioned regulations to a different, more advanced, cladding material. The existing requirements and acceptance criteria currently found in the affected regulations will be maintained if the exemption is granted. Accordingly, the granting of this exemption request is consistent with the common defense and security.

[1] David B. Mitchell, Bert M. Dunn, *Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel*, BAW-10227P-A, February 2000.

4. Special circumstances support the issuance of an exemption

10 CFR § 50.12(a)(2) allows the NRC to grant an exemption to the regulations when special circumstances are present. As discussed below, the special circumstances described in 10 CFR 50.12(a)(2)(ii) support the granting of this exemption application, in that application of these regulations in the particular circumstances described is not necessary to achieve the underlying purpose of the affected regulations—in this case, 10 CFR 50.44, 50.46, and Appendix K to 10 CFR Part 50.

The strict application of the existing fuel cladding requirement in the particular circumstances represented by this exemption application would not serve the underlying purpose of the rule, and, in addition, is not necessary to achieve the underlying purpose of the rule. In this case, the underlying purpose of 10 CFR 50.46 is to ensure that facilities have adequate acceptance criteria for their emergency core cooling systems to assure adequate core cooling. In its topical report,[2] F-ANP demonstrated that the ECCS acceptance criteria applied to reactors with zircaloy clad fuel rods are also applicable to reactors with M5™ clad fuel rods. This report also showed that the M5™ cladding was capable of satisfying these design and acceptance criteria. Therefore, the underlying purposes of Section 50.44 and Part 50 Appendix K, paragraph I.A.5, are achieved through the use of M5™ as a fuel rod cladding material.

5. Relevant Precedent Also Supports Issuance of the Requested Exemption

As further support for this requested exemption, Duke notes that relevant precedent exists for the grant of an exemption from the fuel cladding requirements in 10 CFR 50.44, 50.46, and 10 CFR Part 50, Appendix K, in connection with the anticipated use of MOX fuel at McGuire and Catawba. In March 2000, the NRC issued an exemption to Duke Energy Corporation from certain requirements in 10CFR 50.44, 50.46, and 10 CFR Part 50 Appendix K, to allow the use of Framatome M5™ advanced alloy as a fuel rod cladding material at the Oconee Nuclear

2 *Ibid.*

Station.[3] Duke now proposes to use the identical fuel cladding for the MOX fuel to be irradiated at McGuire and Catawba Nuclear Stations.

[3] Letter from Mr. David E. LaBarge, U.S. Nuclear Regulatory Commission, to Mr. W.R. McCollum, Jr. dated March 23, 2000, "Oconee Nuclear Station , Units 1, 2, and 3 Re: Exemption from Fuel Cladding Requirements."

III. REQUEST FOR EXEMPTION FROM FUEL COMPOSITION REQUIREMENTS REFLECTED IN 10 CFR § 50.46, AND PART 50, APPENDIX K

Duke requests an exemption from the requirements of 10 CFR 50.46(a)(1) and Appendix K to Part 50 such that explicit consideration of MOX fuel is not required in order to be in compliance with these regulations.

A. NRC Standard for Issuance of Exemptions under 10 CFR Part 50

As set forth in Section II.A.1 above, the NRC standard in Section 50.12 for granting an exemption from the requirements in 10 CFR Part 50 requires the applicant to demonstrate that the exemption is authorized by law, that issuance of the exemption will not present an undue risk to public health and safety and is consistent with the common defense and security, and that certain "special circumstances" are present. As set forth below, all of these criteria are met with respect to the exemption request described in Section III, below.

B. Regulatory Provisions from which Exemption Is Needed

10 CFR 50.46(a)(1), as written, applies to power reactors "fueled with uranium oxide pellets," and requires each reactor to be provided with an emergency core cooling system (ECCS) designed so that its calculated cooling performance following a LOCA meets certain criteria in Section 50.46(b). Additionally, Appendix K to Part 50 contains several references that assume that UO_2 fuel pellets are being used, as discussed further below.

C. Circumstances Requiring Exemption from 10 CFR § 50.46 and 10 CFR 50, Appendix K

The proposal to utilize MOX fuel at McGuire and Catawba is inconsistent with certain assumptions in Section 50.46 and 10 CFR Part 50, Appendix K, since these regulations presume the use of reactor fuel in the form of "uranium oxide pellets." In particular, 10 CFR 50.46, as written, implies that all light water reactors must contain uranium oxide pellets as the fuel material, and the performance of the ECCS associated with each reactor must be analyzed with

an acceptable LOCA evaluation model. Clearly, the regulation should also apply to a reactor fueled with MOX fuel pellets.

Similarly, the requirements in Section I.A.1 of Appendix K for calculating the stored energy in the fuel at the onset of the LOCA include the following phrases: "the thermal conductivity of the UO_2 shall be evaluated," and "the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated." For both of these statements, compliance with the underlying requirement can be satisfied for MOX fuel as well as for UO_2 fuel. However, since the requirements refer to UO_2 fuel, an exemption is needed since the regulation as written cannot be met for a reactor containing MOX fuel.

D. Relief Required in Lieu of Complying with Section 50.46 and Part 50, Appendix K As Written

In connection with the proposed utilization of MOX fuel at the McGuire and Catawba facilities, Duke requests an exemption from the assumption in 10 CFR § 50.46 and 10 CFR Part 50, Appendix K that commercial power reactor fuel is exclusively "uranium oxide" fuel. Mixed oxide fuel is not equivalent to uranium oxide fuel.

E. Basis and Justification for Grant of Exemption

As noted earlier, 10 CFR 50.12 allows the NRC to grant exemptions provided that (1) the exemption is authorized by law; (2) the exemption will not present an undue risk to the health and safety of the public; and (3) the exemption is consistent with the common defense and security. In addition, the Commission will not consider granting an exemption unless special circumstances are present. As set forth below, all of the requirements of 10 CFR § 50.12 for the issuance of an NRC exemption have been satisfied.

1. This exemption request is authorized by law

The language in the regulations to specify fuel material in Section 50.46, and in Section I.A.1 of Appendix K to Part 50, was adopted at the discretion of the Commission consistent with its statutory authority. No statute required the NRC to adopt this specification. The NRC has the authority under Section 50.12 to grant exemptions from the requirements of Part 50 upon a showing of proper justification by the applicant. As such, this requested exemption is "authorized by law," as required by 10 CFR § 50.12(a)(1).

2. Granting this exemption request will not present an undue risk to public health and safety

The focus of Section 50.46 is to assure that all light water reactors are equipped with an ECCS and that the ECCS performance be assessed with an acceptable evaluation model. Likewise, the requirements in Appendix K are focused on specifying the required and acceptable features of the LOCA evaluation model. This includes verifying that the evaluation model includes features for describing the thermal conductivity of the fuel material and the fuel-to-cladding gap conductance. Appropriate MOX fuel thermal and mechanical properties will be used in the lead assembly LOCA application analysis. These and all other MOX-specific differences from the approved LOCA evaluation model, BAW-10168P-A Revision 3, are discussed in Section 3.7.1 of the license amendment request. Framatome-ANP will perform a McGuire/Catawba LOCA analysis for the MOX fuel lead assemblies using the methods described in Section 3.7.1 of the license amendment request. The lead assembly analysis will demonstrate that all the acceptance criteria of 10 CFR § 50.46(b) are met. In particular, MOX fuel thermal conductivity and the thermal conductance of the gap between the MOX fuel and the cladding are specifically evaluated. A case representative of expected lead assembly results is presented in Section 3.7.1 of the license amendment request. Both the MOX and LEU representative cases demonstrate acceptable results.

The plant safety analyses will assure that these acceptance criteria are met following the insertion of MOX fuel lead assemblies in Catawba or McGuire. The safety analysis for McGuire and Catawba will be supported by the applicable Technical Specifications. MOX fuel assemblies will be operated in accordance with operating limits as specified in the Technical Specifications. Thus, the granting of this exemption request will not pose an undue risk to public health and safety.

3. Granting this exemption request is consistent with common defense and security

As noted above, this exemption request is only to allow the application of the aforementioned regulations to a different reactor fuel. All of the existing requirements and acceptance criteria currently found in the affected regulations will be maintained if the exemption is granted. Accordingly, the granting of such a request is consistent with the common defense and security.

4. Special circumstances support the issuance of an exemption

10 CFR 50.12(a)(2) allows the NRC to grant an exemption to the regulations when special circumstances are present. As discussed below, the special circumstances described in 10 CFR 50.12(a)(2)(ii) support this exemption application, in that application of these regulations in the particular circumstances described is not necessary to achieve the underlying purpose of the affected regulations.

The underlying purpose of 10 CFR 50.46 and Appendix K to 10 CFR Part 50, paragraph I.A.1, is to establish acceptance criteria for ECCS performance and to ensure that the evaluation model contains provisions for conservatively assessing the amount of stored heat in the fuel at the onset of a postulated LOCA by, *inter alia,* adequately modeling the thermal conductivity of the fuel material and the fuel-to-cladding gap conductance. As is demonstrated in Section 3.7.1 of the lead assembly license amendment request, Framatome-ANP has evaluated the thermal and

material properties of MOX fuel. The properties are very similar to LEU fuel such that the effect of the MOX thermal and material properties on McGuire and Catawba ECCS performance is negligible. Therefore, the underlying purposes of Section 50.46 and Part 50 Appendix K, paragraph I.A.1 are achieved with the use of MOX fuel in lieu of UO_2 fuel. The special circumstance set forth in Section 50.12(a)(2)(ii) thus supports this exemption application.

Moreover, the strict application of the existing UO_2 fuel requirement in the particular circumstances represented by this exemption application is not necessary to achieve the underlying purpose of the affected regulations. In this case, the underlying purpose of 10 CFR 50.46 and 10 CFR 50 Appendix K is to ensure that facilities have adequate acceptance criteria for their emergency core cooling systems. The effects of the different characteristics of MOX fuel on the Framatome-ANP LOCA Evaluation Model and ECCS acceptance criteria are evaluated in the safety analysis of MOX fuel provided in Attachment 3, Section 3.7.1 of this license amendment request. This evaluation demonstrates that the ECCS acceptance criteria are applicable to, and will be met for, McGuire and Catawba reactors with MOX fuel lead assemblies. Therefore, the underlying purpose of 10 CFR 50.46 and 10 CFR 50 Appendix K is achieved with the use of MOX fuel lead assemblies.