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

November 27, 2000

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA9730 AND MA9731)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 197 to Facility Operating License NPF-9 and Amendment No. 178 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated August 1, 2000.

The amendments revise TS Section 3.7.15 and associated Bases, and Section 4.0 of the McGuire Nuclear Stations, Units 1 and 2, licenses to allow the use of credit for soluble boron in spent fuel pool criticality analyses. The request is based on the NRC-approved Westinghouse Owners Group Topical Report WCAP-14416-NP-A, which provides generic methodology for crediting soluble boron. The review has included the evaluation of the criticality aspects, Boraflex degradation, and boron dilution event analysis. The staff's approval is contingent on completion of the proposed Boraflex verification testing in Updated Final Safety Analysis Report Section 16.9-9, "Spent Fuel Pool Storage Rack Poison Material."

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Frank Rinaldi

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 197 to NPF-9
2. Amendment No. 178 to NPF-17
3. Safety Evaluation

cc w/encl: See next page

NRR-058

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

November 27, 2000

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA5220 AND MA5221)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 197 to Facility Operating License NPF-9 and Amendment No. 178 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated August 1, 2000.

The amendments revise TS Section 3.7.15 and associated Bases, and Section 4.0 of the McGuire Nuclear Stations, Units 1 and 2, licenses to allow the use of credit for soluble boron in spent fuel pool criticality analyses. The request is based on the NRC-approved Westinghouse Owners Group Topical Report WCAP-14416-NP-A, which provides generic methodology for crediting soluble boron. The review has included the evaluation of the criticality aspects, Boraflex degradation, and boron dilution event analysis. The staff's approval is contingent on completion of the proposed Boraflex verification testing in Updated Final Safety Analysis Report Section 16.9-9, "Spent Fuel Pool Storage Rack Poison Material."

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 197 to NPF-9
2. Amendment No. 178 to NPF-17
3. Safety Evaluation

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. NPF-9

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-9 filed by the Duke Energy Corporation (licensee) dated August 1, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 197 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 27, 2000

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. NPF-17

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Facility Operating License No. NPF-17 filed by the Duke Energy Corporation (licensee) dated August 1, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 178 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 27, 2000

ATTACHMENT TO LICENSE AMENDMENT NOS. 197 AND 178

FACILITY OPERATING LICENSE NOS. NPF-9 AND NPF-17

DOCKET NOS. 50-369 AND 50-370

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
3.7.15-1	3.7.15-1
3.7.15-2	3.7.15-2
3.7.15-3	3.7.15-3
3.7.15-4	3.7.15-4
3.7.15-5	3.7.15-5
3.7.15-6	3.7.15-6
3.7.15-7	3.7.15-7
3.7.15-8	3.7.15-8
3.7.15-9	3.7.15-9
3.7.15-10	3.7.15-10
-	3.7.15-11
-	3.7.15-12
-	3.7.15-13
-	3.7.15-14
-	3.7.15-15
-	3.7.15-16
-	3.7.15-17
-	3.7.15-18
-	3.7.15-19
-	3.7.15-20
-	3.7.15-21
4.0-1	4.0-1
4.0-2	4.0-2
B 3.7.14-1	B 3.7.14-1
B 3.7.14-2	B 3.7.14-2
B 3.7.14-3	B 3.7.14-3
-	B 3.7.14-4
B 3.7.15-1	B 3.7.15-1
B 3.7.15-2	B 3.7.15-2
B 3.7.15-3	B 3.7.15-3
-	B 3.7.15-4
-	B 3.7.15-5

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 not meet the criteria of Table 3.7.15-1 or Table 3.7.15-2.

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 not meet the criteria of Table 3.7.15-5.

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 not meet the criteria of Table 3.7.15-8.

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.

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

ACTIONS

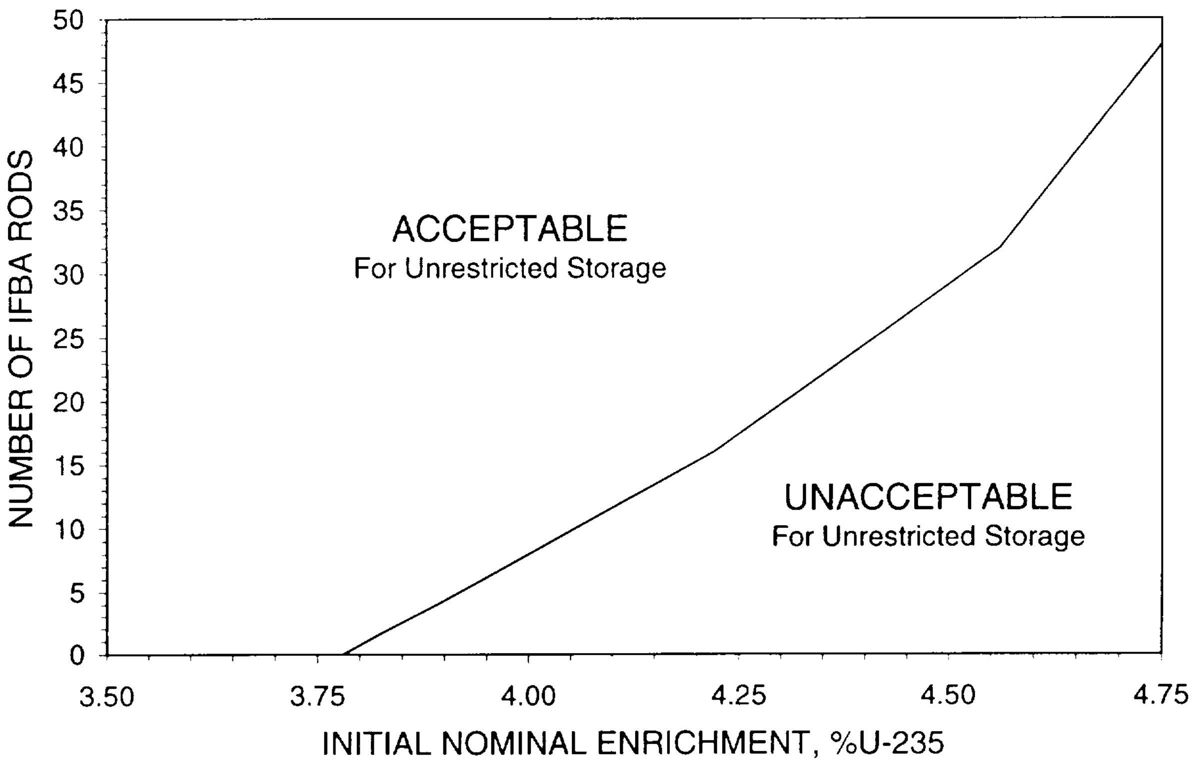
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------------- Initiate action to move the noncomplying fuel assembly to the correct location.	Immediately

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

Table 3.7.15-1 (page 1 of 1)
Minimum Qualifying Number of IFBA Rods Versus Initial Enrichment
for Unrestricted Region 1A Storage of New Fuel

Initial Nominal Enrichment (% U-235)	Number of IFBA Rods
3.78 (or less)	0
4.22	16
4.56	32
4.75	48

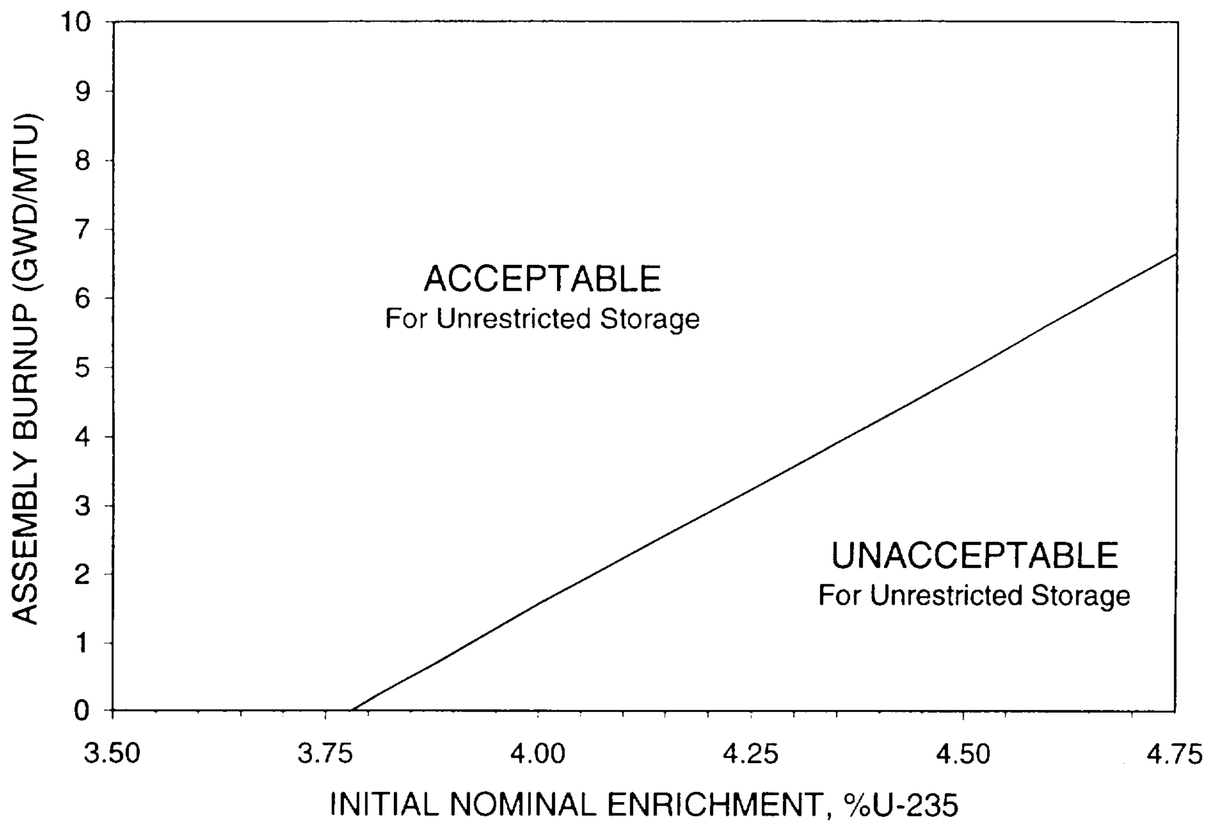


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for Unrestricted Region 1A storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-2 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 1A Storage

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
3.78 (or less)	0
4.00	1.58
4.50	4.92
4.75	6.66

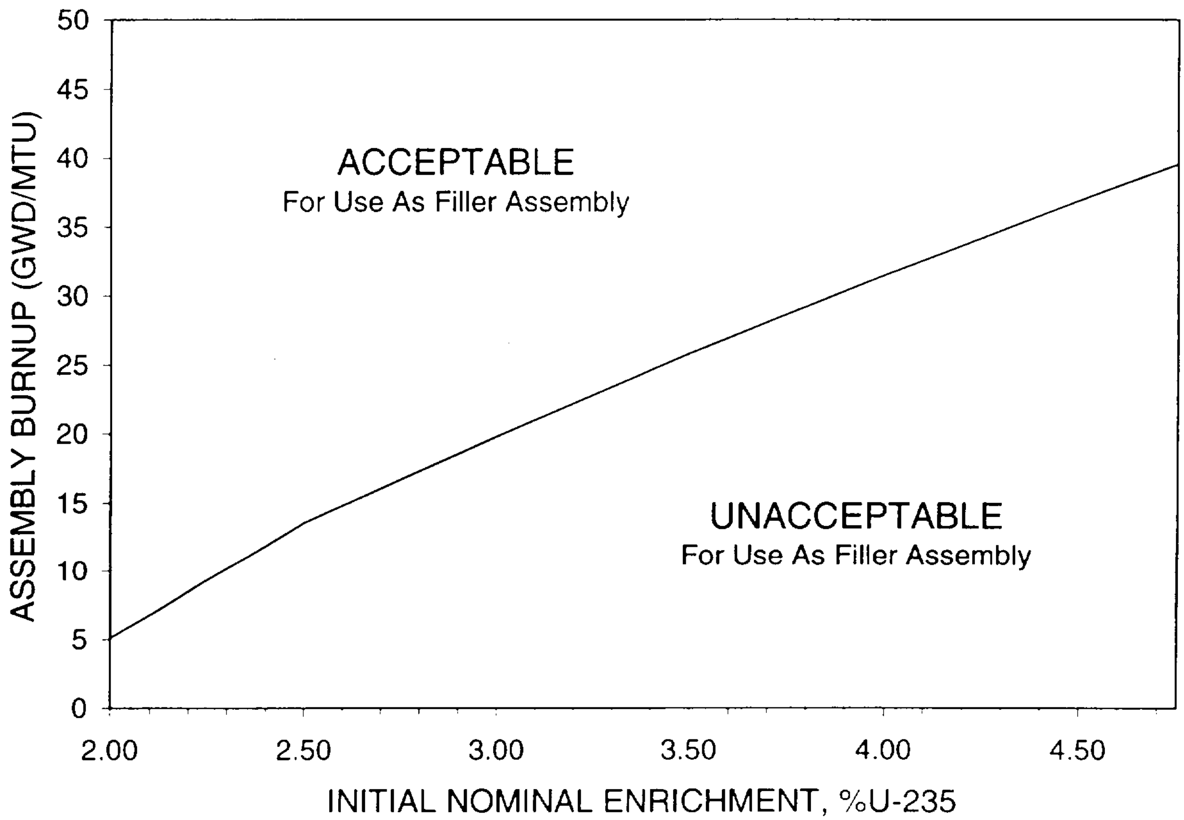


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for Unrestricted Region 1A storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron. Likewise, previously unanalyzed fuel up to a nominal 4.75 weight% U-235 may be qualified for Restricted Region 1A storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-3 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 1A Filler Assemblies

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.76 (or less)	0
2.00	5.12
2.50	13.57
3.00	19.80
3.50	25.85
4.00	31.50
4.50	36.93
4.75	39.54

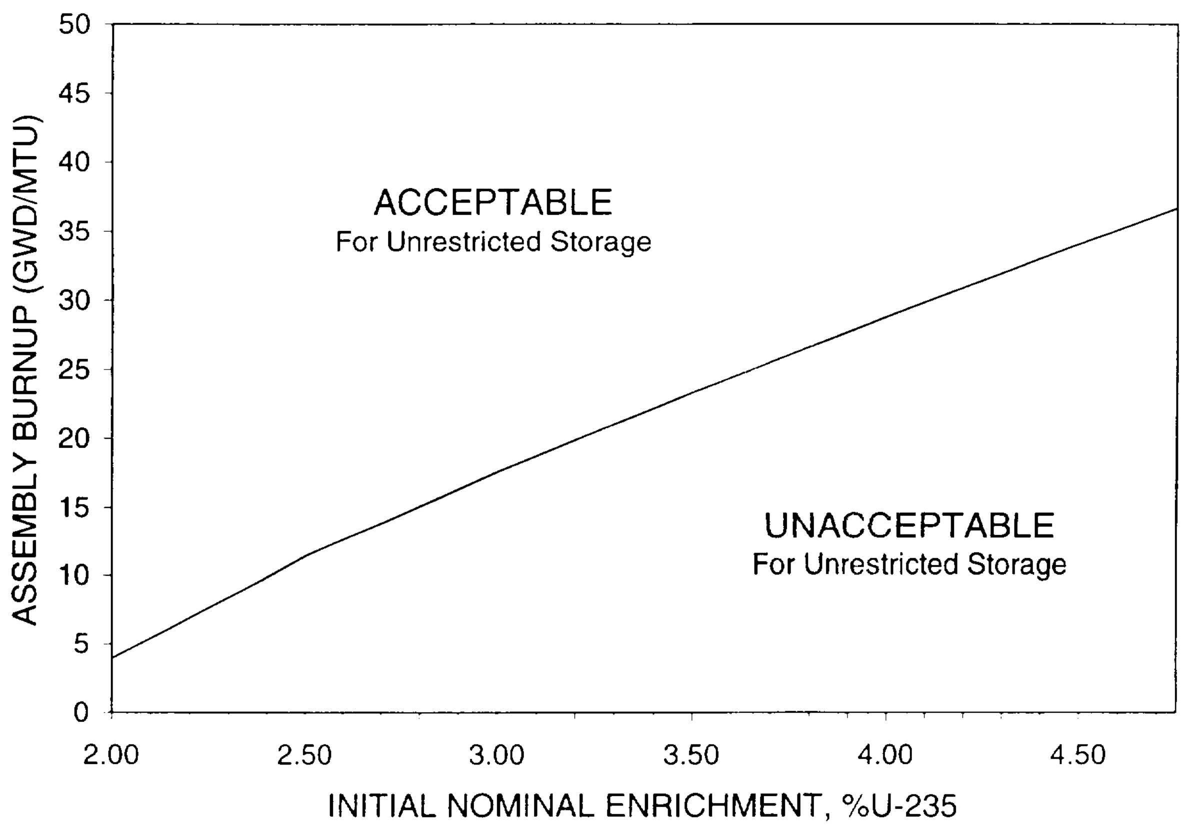


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-3 may be qualified for use as a Region 1A Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-4 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 1B Storage

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.78 (or less)	0
2.00	3.96
2.50	11.35
3.00	17.61
3.50	23.35
4.00	28.86
4.50	34.10
4.75	36.67

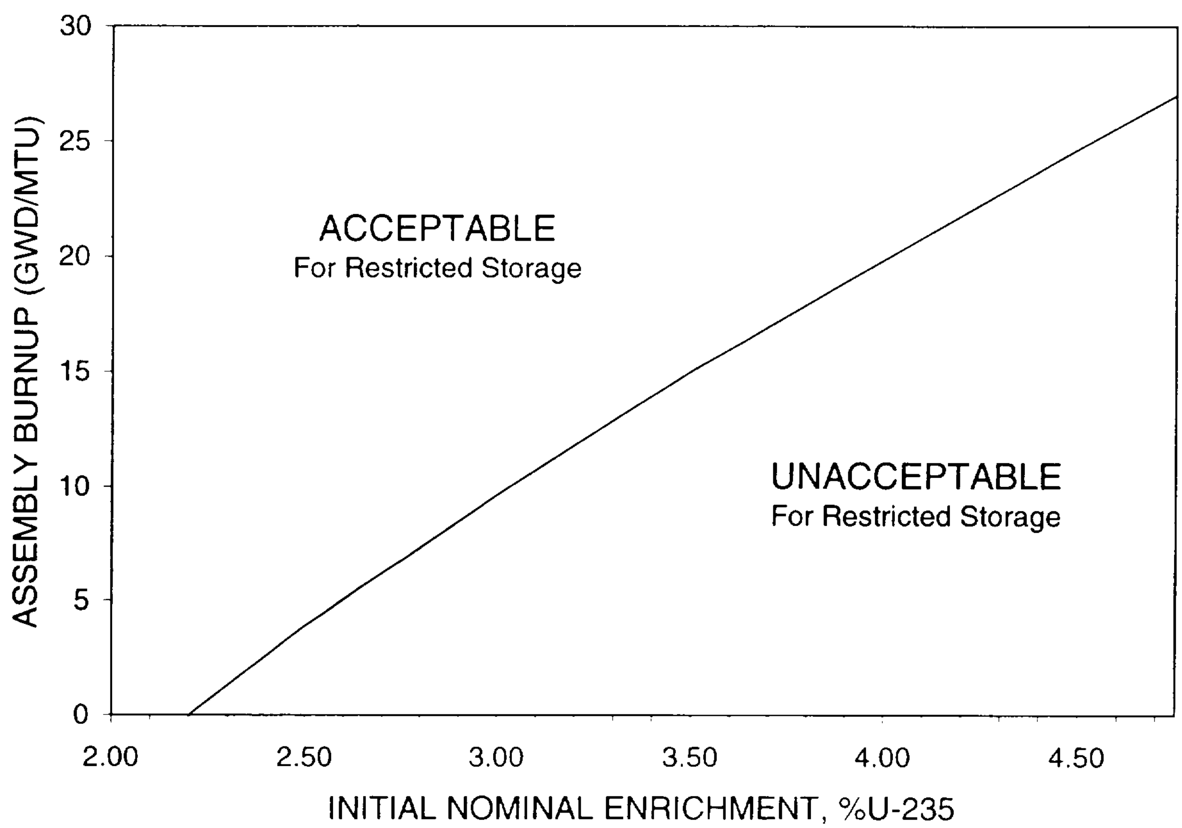


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for Unrestricted Region 1B storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-5 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Restricted Region 1B Storage with Fillers

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
2.20 (or less)	0
2.50	3.91
3.00	9.65
3.50	15.04
4.00	19.87
4.50	24.68
4.75	27.01

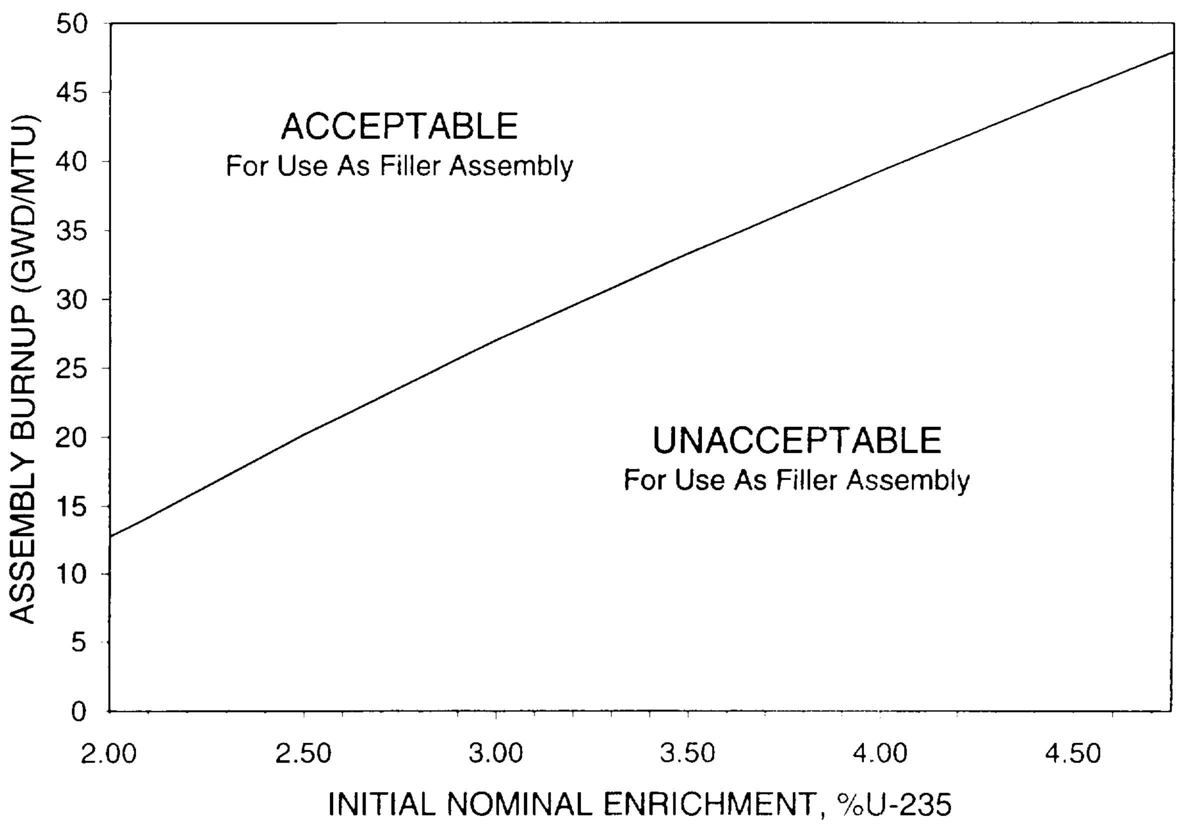


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-5 may be qualified for Restricted Region 1B Storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-6 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 1B Filler Assemblies

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.44 (or less)	0
2.00	12.68
2.50	20.17
3.00	27.03
3.50	33.35
4.00	39.33
4.50	45.07
4.75	47.89

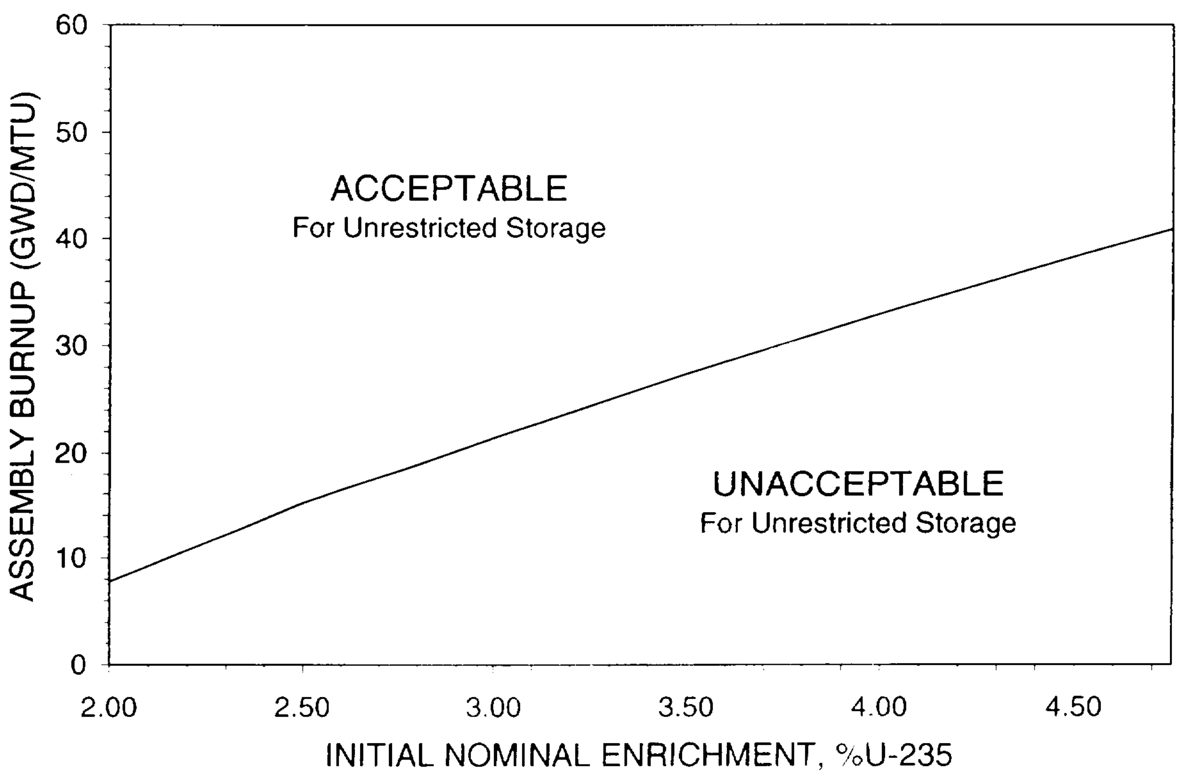


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-6 may be qualified for use as a Region 1B Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-7 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 2A Storage

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.61 (or less)	0
2.00	7.79
2.50	15.14
3.00	21.45
3.50	27.42
4.00	33.00
4.50	38.32
4.75	40.91

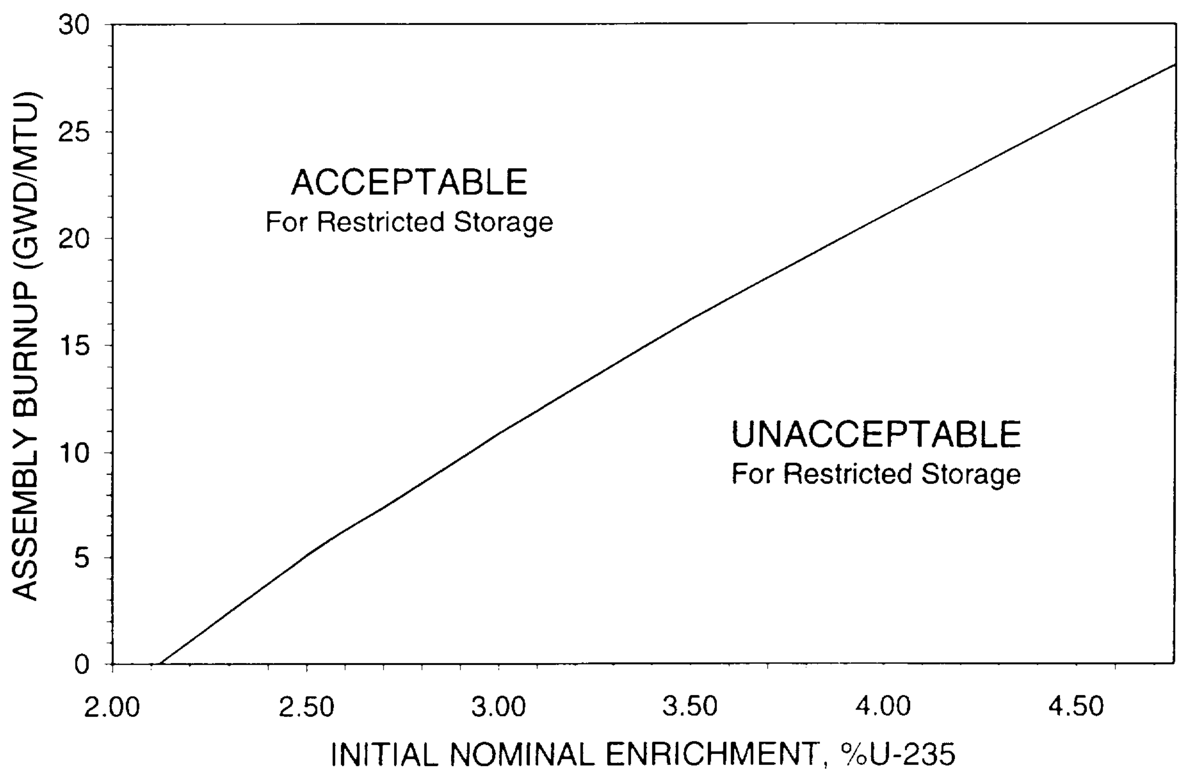


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-7 may be qualified for Unrestricted Region 2A storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-8 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Restricted Region 2A Storage with Fillers

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
2.12 (or less)	0
2.50	5.10
3.00	10.88
3.50	16.19
4.00	21.07
4.50	25.81
4.75	28.11

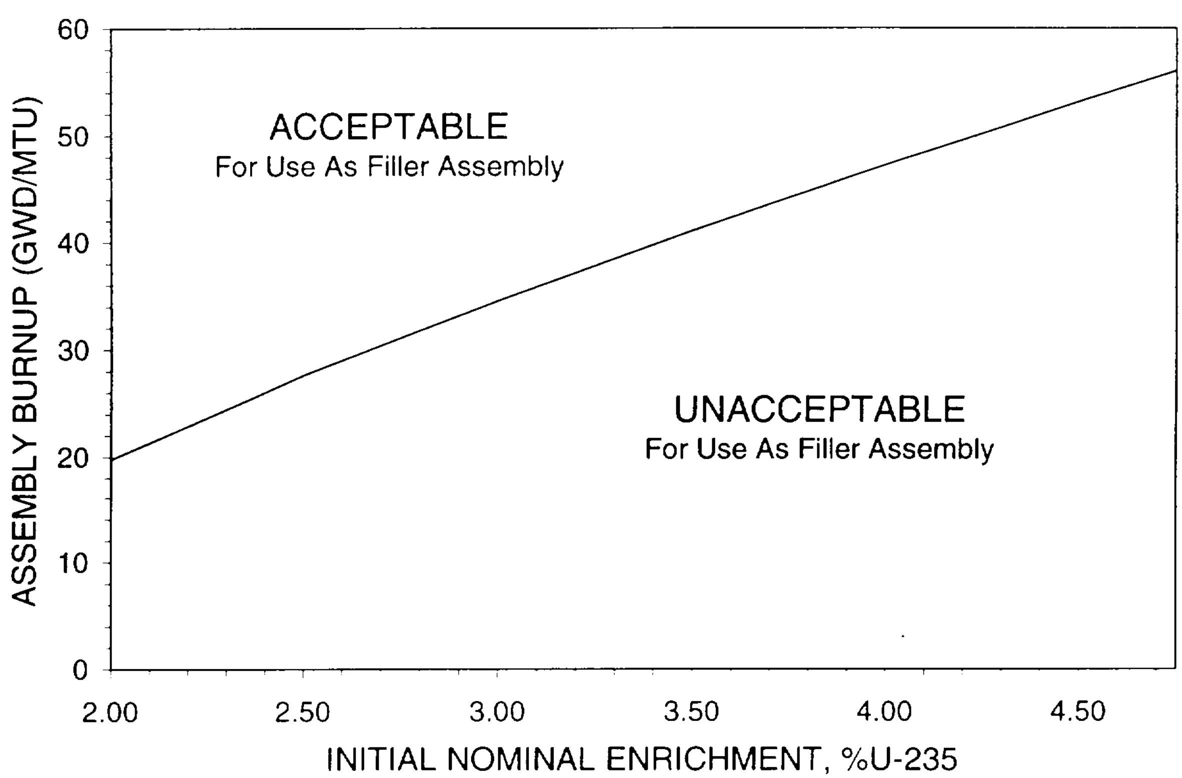


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-8 may be qualified for Restricted Region 2A Storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-9 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 2A Filler Assemblies

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.20 (or less)	0
2.00	19.80
2.50	27.64
3.00	34.56
3.50	41.08
4.00	47.25
4.50	53.15
4.75	56.01

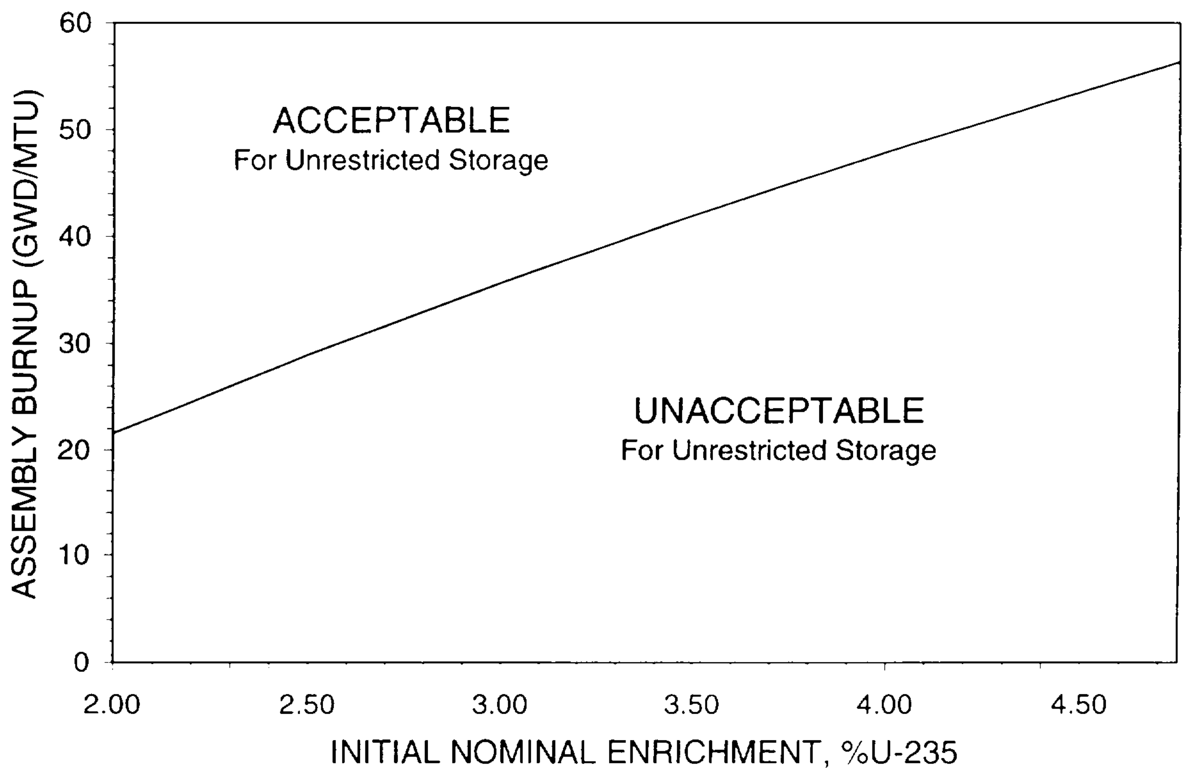


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-9 may be qualified for use as a Region 2A Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-10 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 2B Storage

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.11 (or less)	0
2.00	21.58
2.50	29.00
3.00	35.69
3.50	41.97
4.00	47.90
4.50	53.57
4.75	56.33

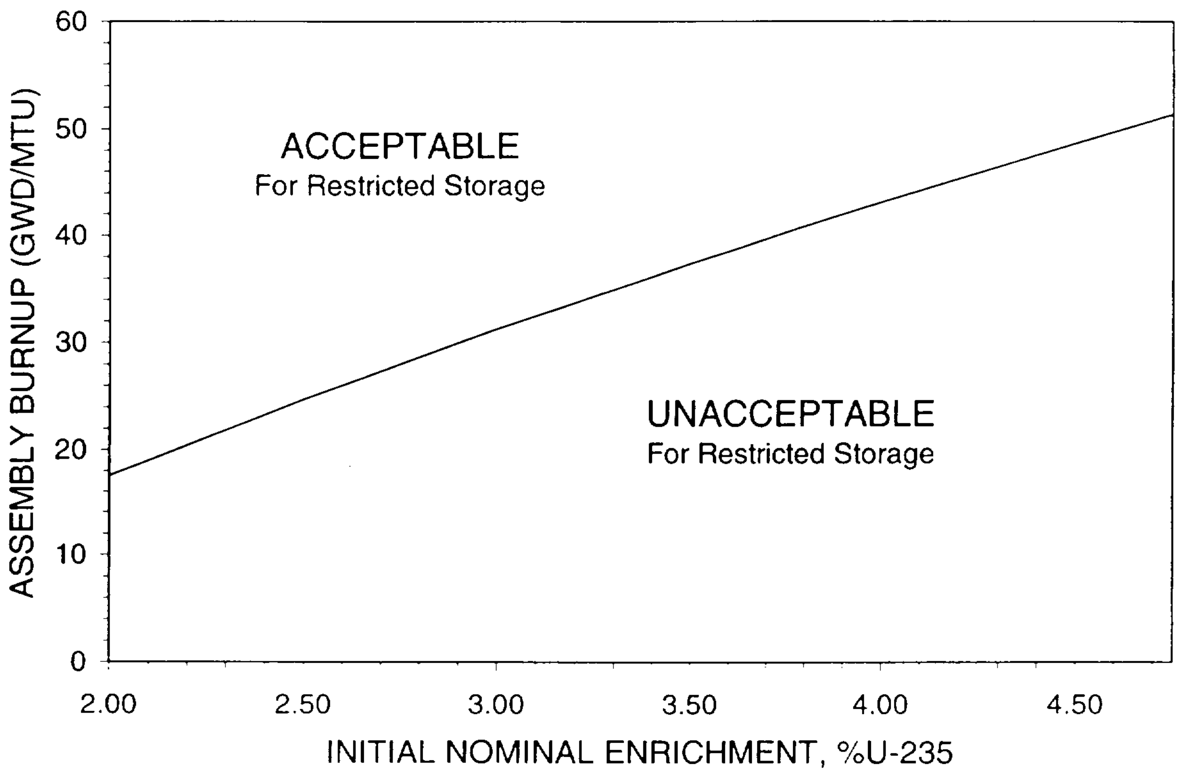


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-10 may be qualified for Unrestricted Region 2B storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

Table 3.7.15-11 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Restricted Region 2B Storage with Fillers

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.22 (or less)	0
2.00	17.55
2.50	24.73
3.00	31.31
3.50	37.40
4.00	43.15
4.50	48.65
4.75	51.33

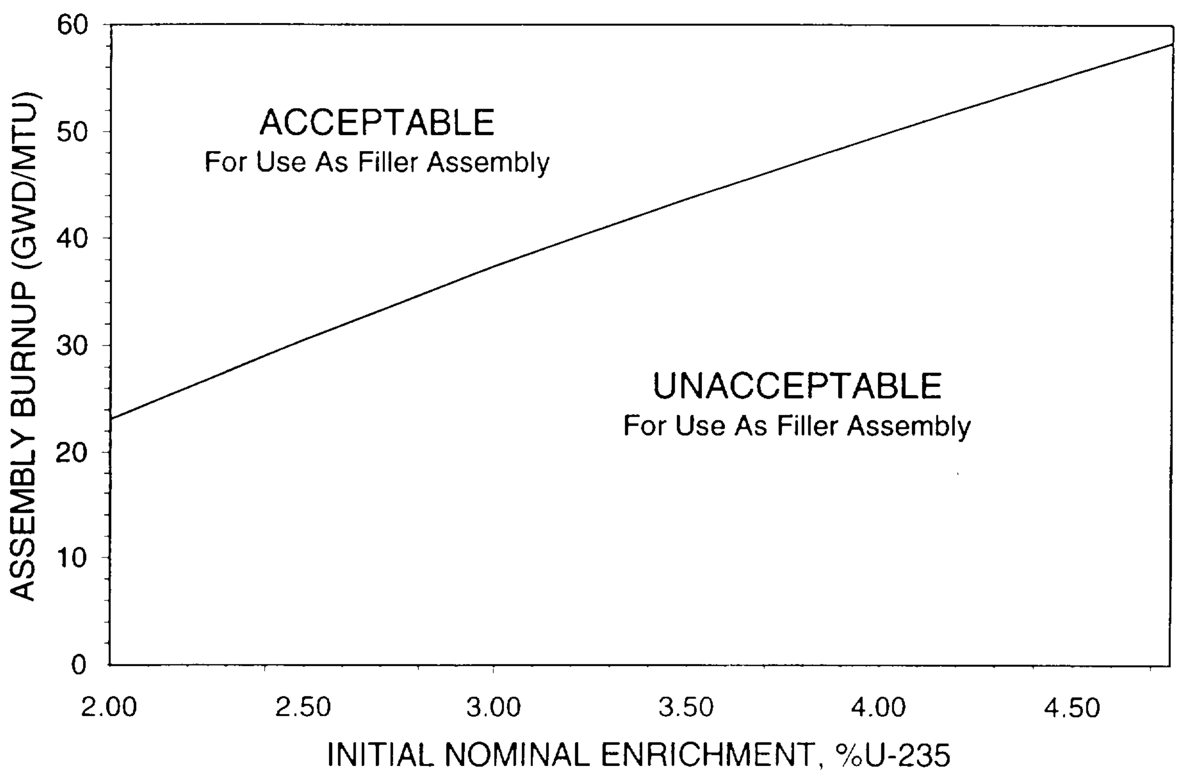


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-11 may be qualified for Restricted Region 2B Storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

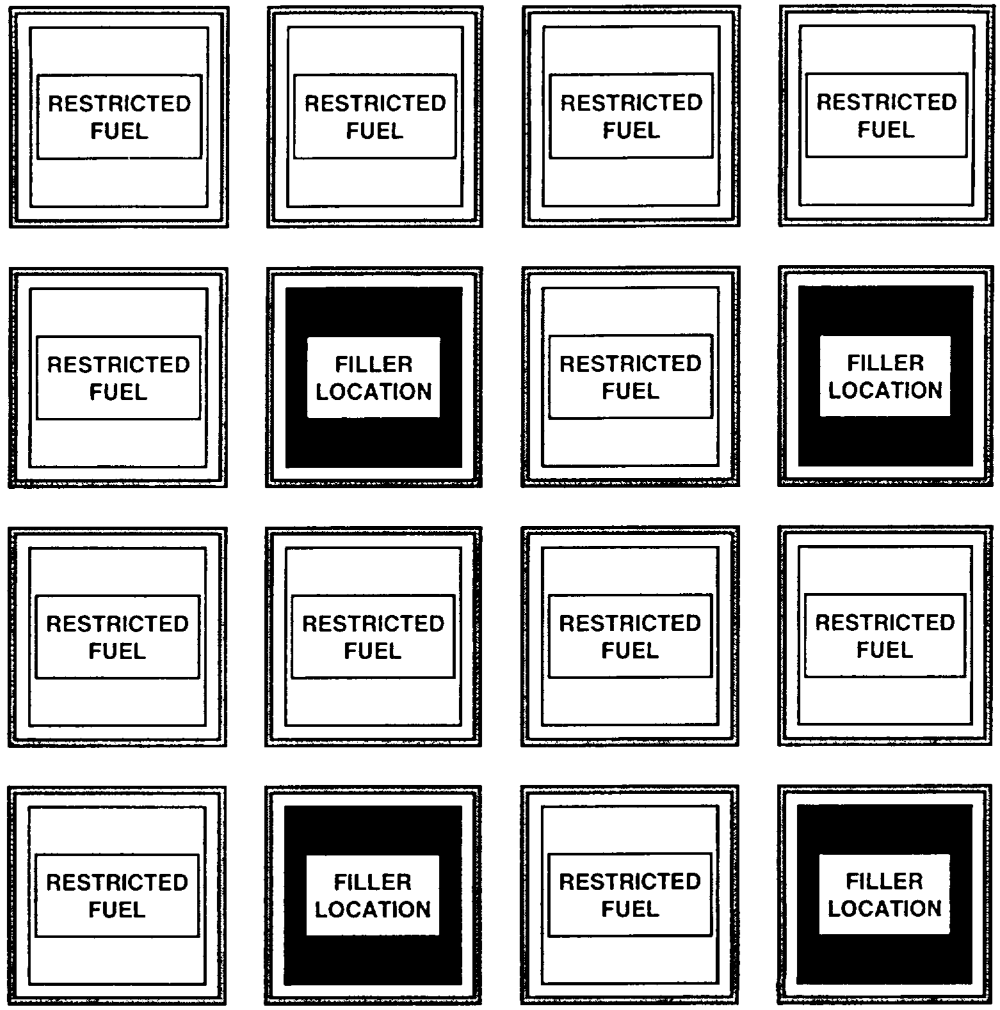
Table 3.7.15-12 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 2B Filler Assemblies

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.08 (or less)	0
2.00	23.14
2.50	30.59
3.00	37.42
3.50	43.74
4.00	49.72
4.50	55.49
4.75	58.33



NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-12 may be qualified for use as a Region 2B Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

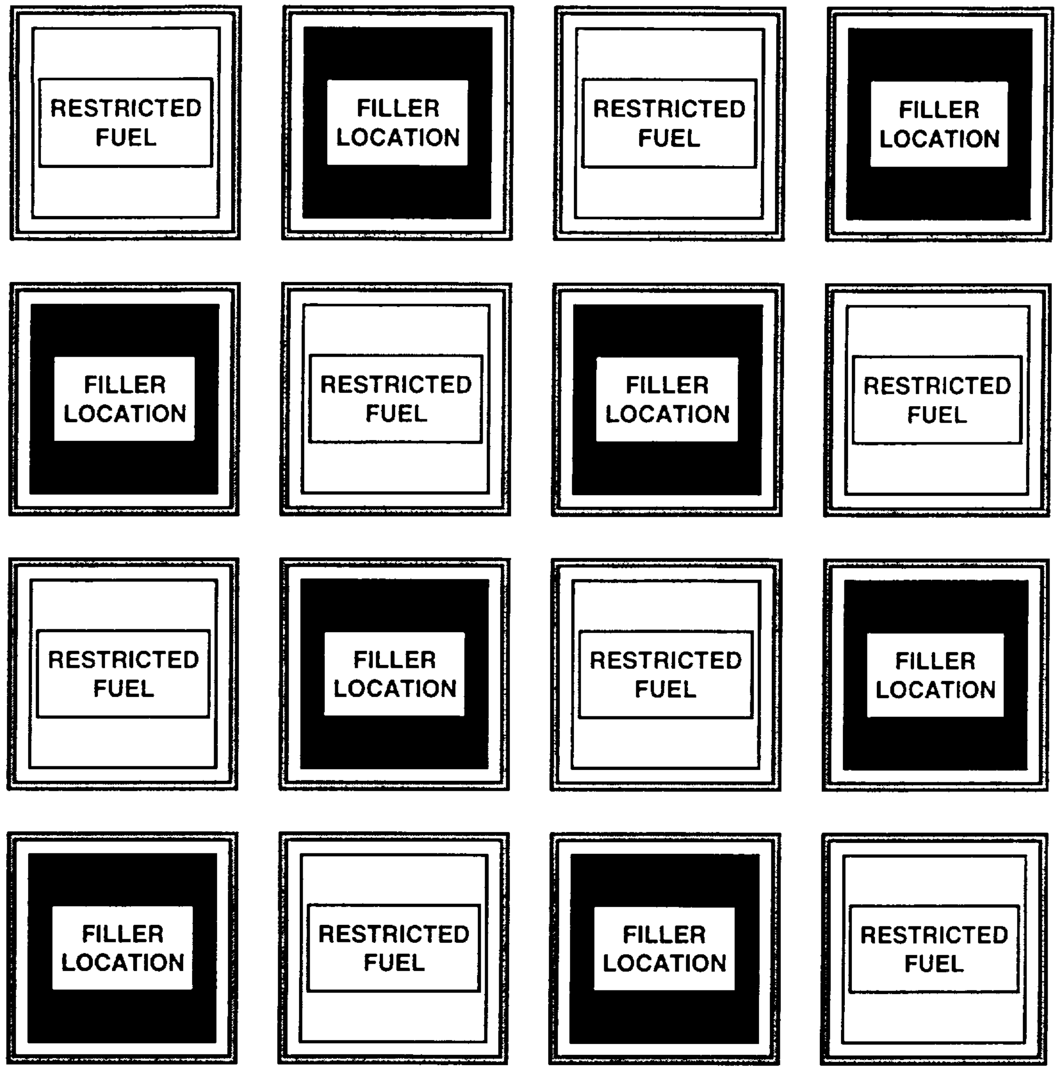


Restricted Fuel: Fuel which does not meet the minimum burnup requirements of either Table 3.7.15-1 or Table 3.7.15. 2. (Fuel which does meet the requirements of Table 3.7.15-1 or Table 3.7.15.-2, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed).

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

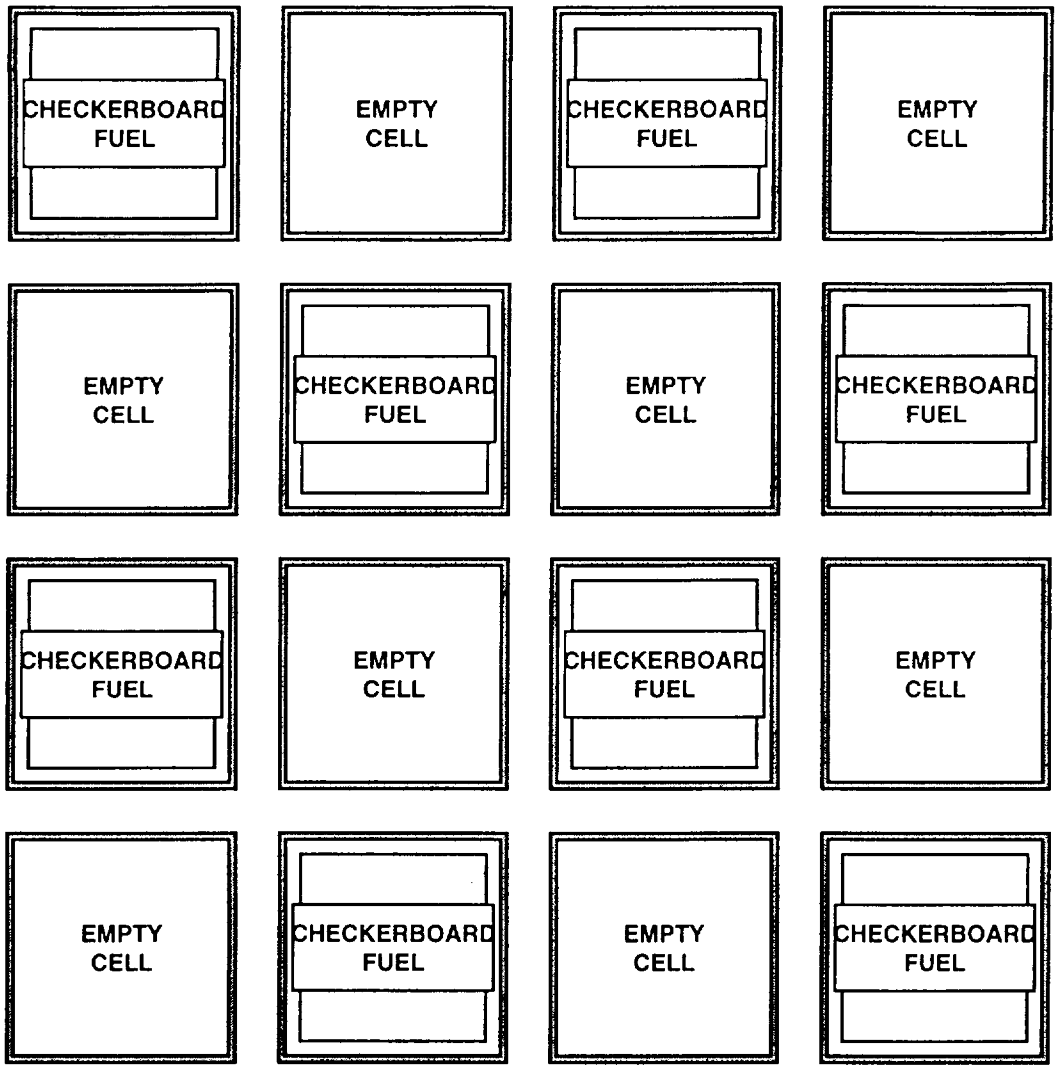
Boundary Condition: Any Restricted Region 1A Storage Area row bounded by any other storage area shall contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other. Example: In the figure above, row 1 or column 1 can 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



Restricted Fuel:	Fuel which meets the minimum burnup requirements of Table 3.7.15-5, or non-fuel components, or an empty location.
Filler Location:	Either fuel which meets the minimum burnup requirements of Table 3.7.15-6, or an empty cell.
Boundary Condition:	No restrictions on boundary assemblies.

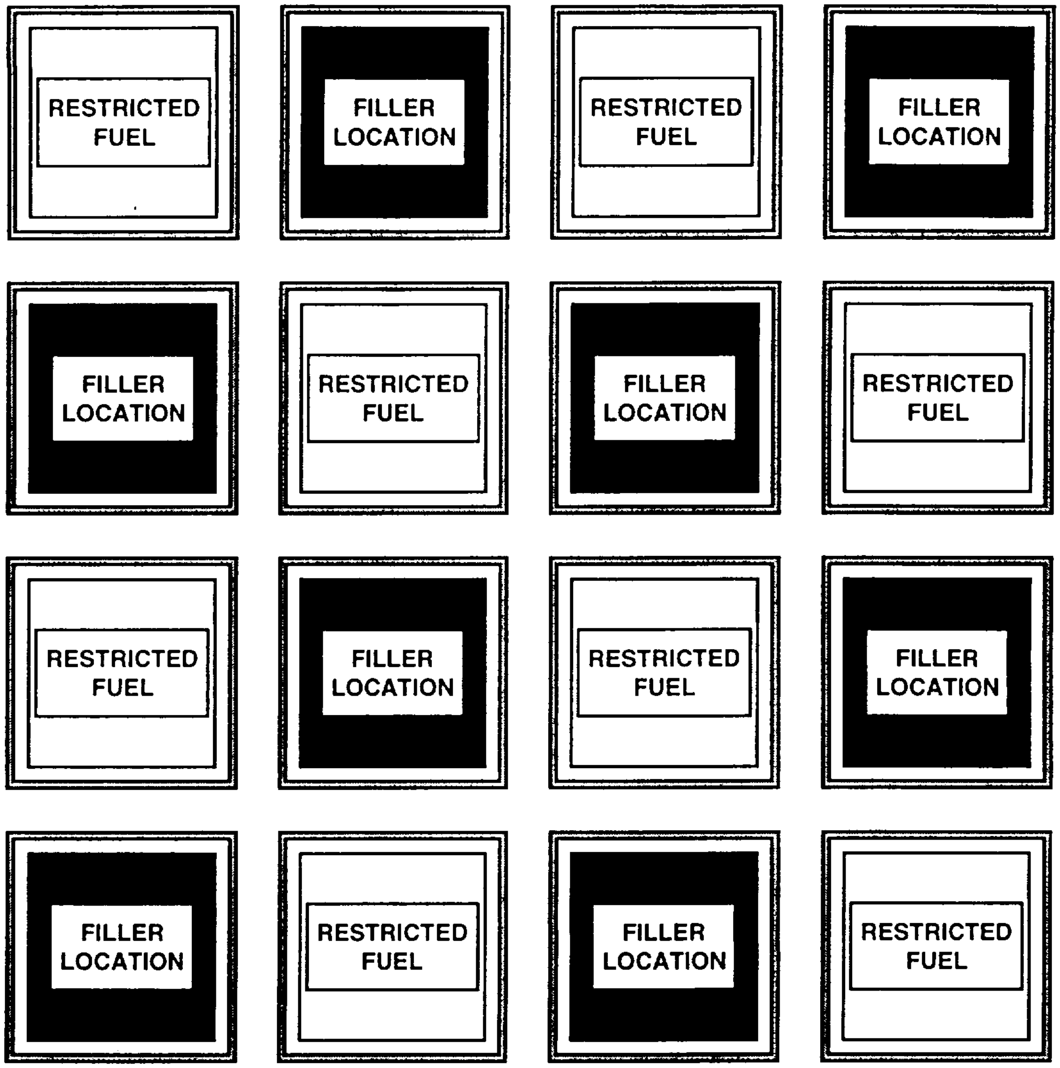
Figure 3.7.15-2 (page 1 of 1)
Required 2 out of 4 Loading Pattern for Restricted Region 1B Storage



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)

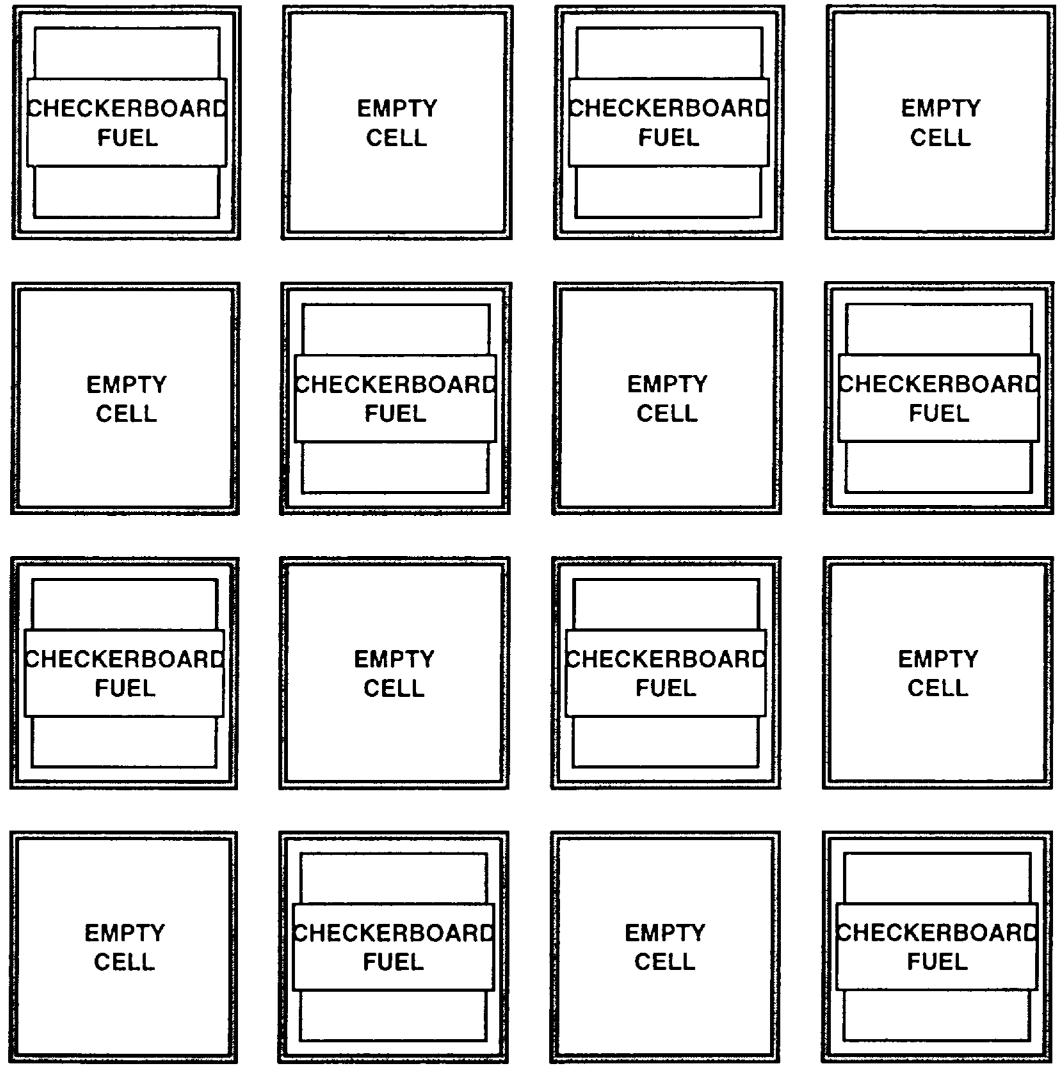
Boundary Condition: No restrictions on boundary assemblies.

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



Restricted Fuel:	Fuel which meets the minimum burnup requirements of Table 3.7.15-8, or non-fuel components, or an empty location.
Filler Location:	Either fuel which meets the minimum burnup requirements of Table 3.7.15-9, or an empty cell.
Boundary Condition:	No restrictions on boundary assemblies.

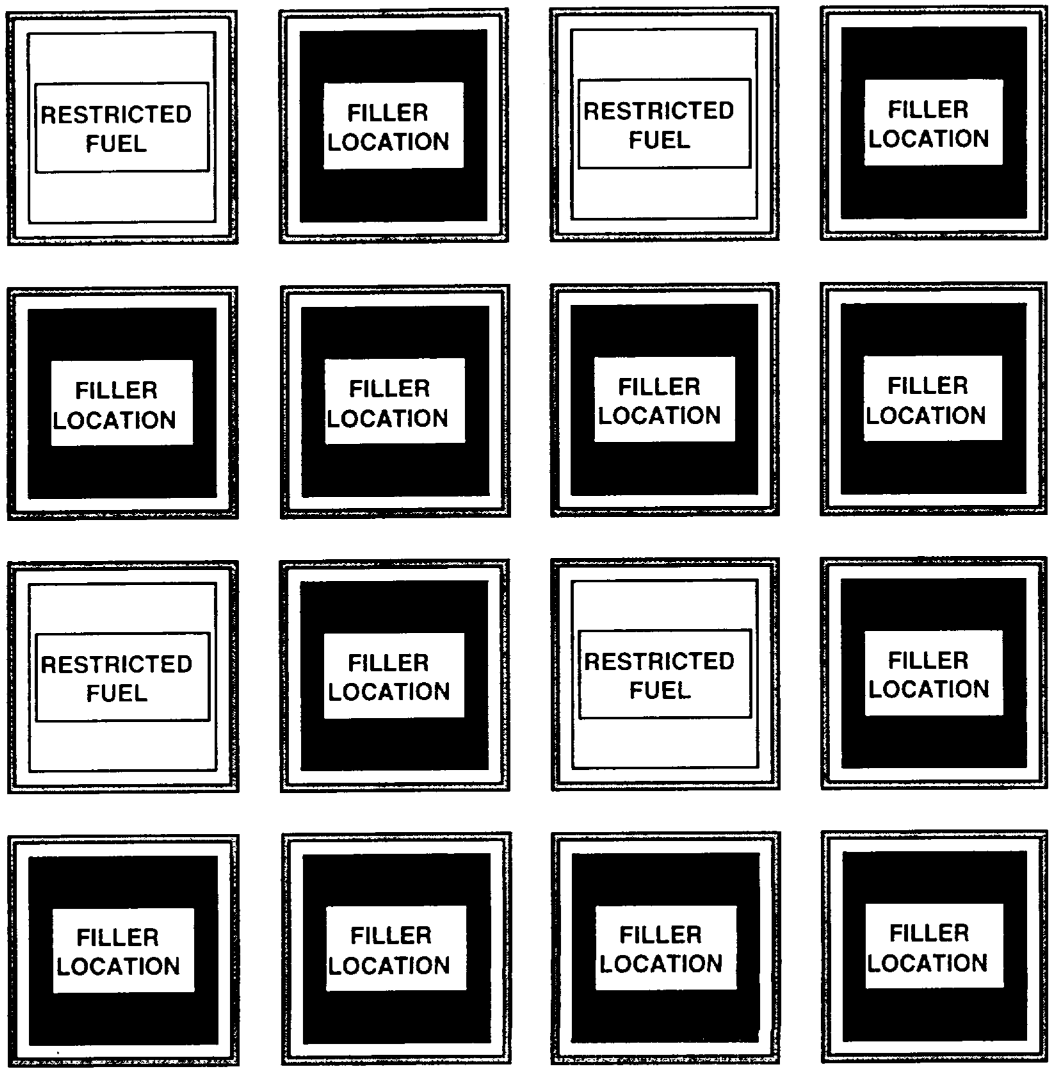
Figure 3.7.15-4 (page 1 of 1)
Required 2 out of 4 Loading Pattern for Restricted Region 2A Storage



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: No restrictions on boundary assemblies.

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

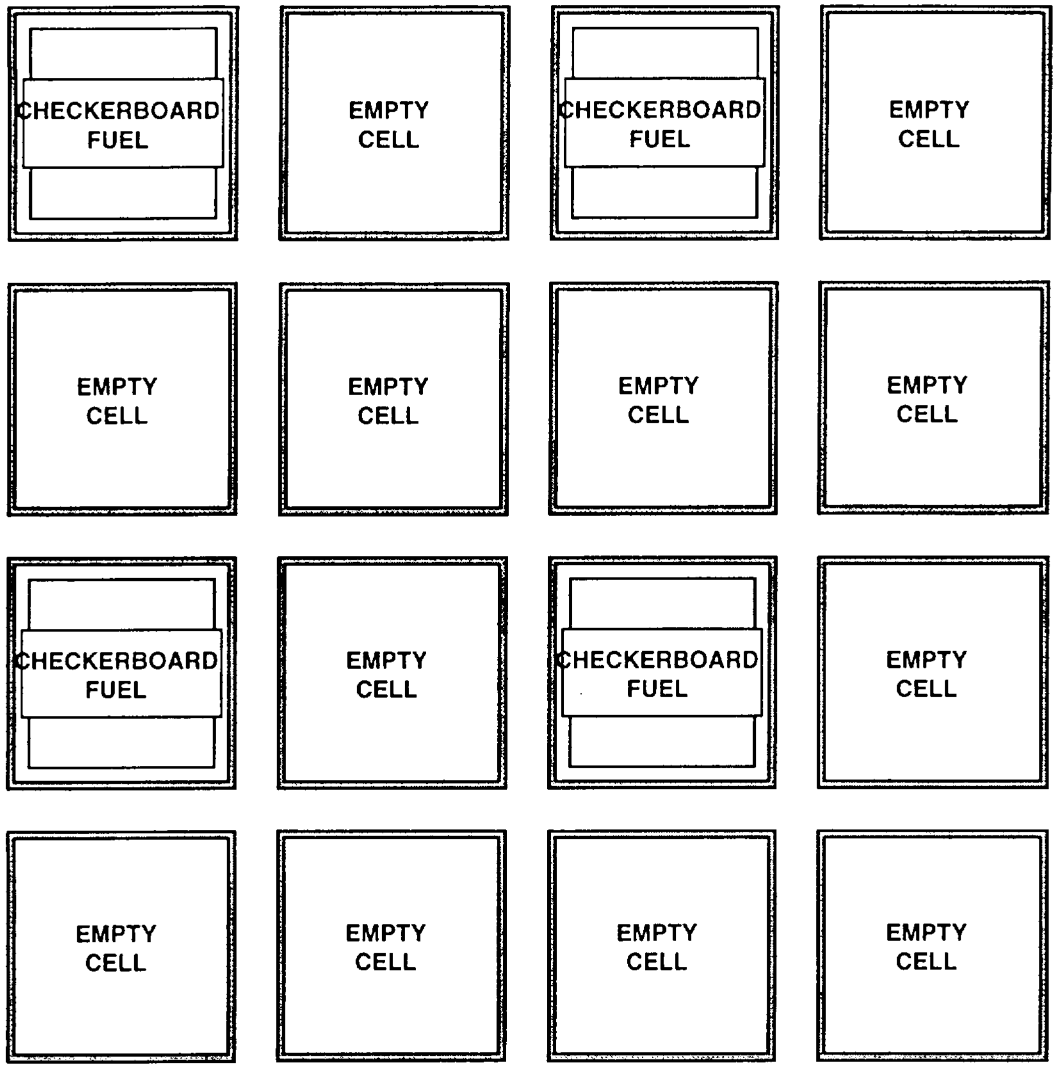


Restricted Fuel: Fuel which meets the minimum burnup requirements of Table 3.7.15-11, or non-fuel components, or an empty location.

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.7.15-12, or an empty cell.

Boundary Condition: Any Restricted Region 2B Storage Area row bounded by any other storage area shall contain only filler locations arranged such that no Restricted Fuel assemblies are adjacent to any other fuel except Region 2B Filler Locations. 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 3.7.15-6 (page 1 of 1)
Required 1 out of 4 Loading Pattern for Restricted Region 2B 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

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

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.

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;

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 730 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

d. A nominal 10.4 inch center to center distance between fuel assemblies placed in Regions 1A and 1B; and

e. A nominal 9.125 inch center to center distance between fuel assemblies placed in Regions 2A and 2B.

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 4.75 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 745 ft.-7 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 total spaces in Regions 1A and 1B and 1177 total spaces in Regions 2A and 2B).

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

In the two region poison fuel storage rack (Refs. 1 and 2) design, the spent fuel pool is divided into two separate and distinct regions. Region 1, with 286 storage positions, is designed and generally reserved for temporary storage of new or partially irradiated fuel. Region 2, with 1177 storage positions, is designed and generally used for normal, long term storage of permanently discharged fuel that has achieved qualifying burnup levels.

The McGuire spent fuel storage racks contain Boraflex neutron-absorbing panels that surround each storage cell on all four sides (except for peripheral sides). The function of these Boraflex panels is to ensure that the reactivity of the stored fuel assemblies is maintained within required limits. Boraflex, as manufactured, is a silicon rubber material that retains a powder of boron carbide (B4C) neutron absorbing material. The Boraflex panels are enclosed in a formed stainless steel wrapper sheet that is spot-welded to the storage tube. The wrapper sheet is bent at each end to complete the enclosure of the Boraflex panel. The Boraflex panel is contained in the plenum area between the storage tube and the wrapper plate. Since the wrapper plate enclosure is not sealed, spent fuel pool water is free to circulate through the plenum. It has been observed that after Boraflex receives a high gamma dose from the stored irradiated fuel ($>10^{10}$ rads) it can begin to degrade and dissolve in the wet environment. Thus, the B4C poison material can be removed, thereby reducing the poison worth of the Boraflex sheets. This phenomenon is documented in NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks".

To address this degradation, each region of the spent fuel pool has been divided into two sub-regions; with and without credit for Boraflex. For the regions taking credit for Boraflex, a minimum amount of Boraflex was assumed that is less than the original design minimum B_{10} areal density.

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 $k_{eff} \leq$

BASES

BACKGROUND (continued)

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 730 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 (Ref. 6) can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 730 ppm required to maintain k_{eff} less than or equal to 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by either of

BASES

APPLICABLE SAFETY ANALYSES (continued)

these postulated accidents and to maintain k_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 1470 ppm was adequate to mitigate these postulated criticality related accidents and to maintain k_{eff} less than or equal to 0.95. Specification 3.7.14 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by these postulated accidents.

Specification 4.3.1.1 c. requires that the spent fuel rack k_{eff} be less than or equal to 0.95 when flooded with water borated to 730 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 730 ppm is not a credible event.

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 5).

LCO

The spent fuel pool boron concentration is required to be within the limits specified in the COLR. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies.

BASES

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. UFSAR, Section 9.1.2.
2. Issuance of Amendments, McGuire Nuclear Station, Units 1 and 2 (TAC NOS. M89744 and M89745), November 6, 1995.
3. WCAP-14416-NP-A, Westinghouse Spent Fuel Rack Criticality Analysis Methodology, Revision 1, November 1996.
4. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.
5. Nuclear Regulatory Commission, Memorandum to Timothy Collins from Laurence Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," August 19, 1998.
6. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
8. UFSAR, Section 15.7.4.

BASES

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the two region poison fuel storage rack (Refs. 1 and 2) design, the spent fuel pool is divided into two separate and distinct regions. Region 1, with 286 storage positions, is designed and generally reserved for temporary storage of new or partially irradiated fuel. Region 2, with 1177 storage positions, is designed and generally used for normal, long term storage of permanently discharged fuel that has achieved qualifying burnup levels.

The McGuire spent fuel storage racks contain Boraflex neutron-absorbing panels that surround each storage cell on all four sides (except for peripheral sides). The function of these Boraflex panels is to ensure that the reactivity of the stored fuel assemblies is maintained within required limits. Boraflex, as manufactured, is a silicon rubber material that retains a powder of boron carbide (B4C) neutron absorbing material. The Boraflex panels are enclosed in a formed stainless steel wrapper sheet that is spot-welded to the storage tube. The wrapper sheet is bent at each end to complete the enclosure of the Boraflex panel. The Boraflex panel is contained in the plenum area between the storage tube and the wrapper plate. Since the wrapper plate enclosure is not sealed, spent fuel pool water is free to circulate through the plenum. It has been observed that after Boraflex receives a high gamma dose from the stored irradiated fuel ($>10^{10}$ rads) it can begin to degrade and dissolve in the wet environment. Thus, the B4C poison material can be removed, thereby reducing the poison worth of the Boraflex sheets. This phenomenon is documented in NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks".

To address this degradation, each region of the spent fuel pool has been divided into two sub-regions; with and without credit for Boraflex. For the regions taking credit for Boraflex, a minimum amount of Boraflex was assumed that is less than the original design minimum B10 areal density.
To address this degradation, each region of the spent fuel pool has been divided into two sub-regions; with and without credit for Boraflex. For the regions taking credit for Boraflex, a minimum amount of Boraflex was assumed that is less than the original design minimum B10 areal density.

Two storage configurations are defined for each region; Unrestricted and Restricted storage. Unrestricted storage allows storage in all cells without restriction on the storage configuration. Restricted storage allows storage of higher reactivity fuel when restricted to a certain storage

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.

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 730 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Ref. 6) can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 730 ppm required to maintain k_{eff} less than or equal to 0.95 under normal conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by either of these postulated accidents and to maintain k_{eff} less than or equal to 0.95. It was found that a spent fuel pool boron concentration of 1470 ppm was adequate to mitigate these postulated criticality related accidents and to maintain k_{eff} less than or equal to 0.95. Specification 3.7.14 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by these postulated accidents.

Specification 4.3.1.1 c. requires that the spent fuel rack k_{eff} be less than or equal to 0.95 when flooded with water borated to 730 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 730 ppm is not a credible event.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36 (Ref. 7).

LCO

a

The restrictions on the placement of fuel assemblies within the Region 1A of the spent fuel pool, which have a number of IFBA rods greater than or equal to the minimum qualifying number of IFBA rods in Table 3.7.15-1 or accumulated burnup greater than or equal to the minimum qualified burnups in Table 3.7.15-2 in the accompanying LCO, ensures the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95, assuming the pool to be flooded with water borated to 730 ppm. Fuel assemblies not meeting the criteria of Tables 3.7.15-1 or 3.7.15-2 shall be stored in accordance with Figure 3.7.15-1.

BASES

LCO (continued)

b

The restrictions on the placement of fuel assemblies within the Region 1B of the spent fuel pool, which have accumulated burnup greater than or equal to the minimum qualified burnups in Table 3.7.15-4 in the accompanying LCO, ensures the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95, assuming the pool to be flooded with water borated to 730 ppm. Fuel assemblies not meeting the criteria of Table 3.7.15-4 shall be stored in accordance with either Figure 3.7.15-2 and Table 3.7.15-5 for Restricted storage, or Figure 3.7.15-3 for Checkerboard storage.

c

The restrictions on the placement of fuel assemblies within the Region 2A of the spent fuel pool, which have accumulated burnup greater than or equal to the minimum qualified burnups in Table 3.7.15-7 in the accompanying LCO, ensures the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95, assuming the pool to be flooded with water borated to 730 ppm. Fuel assemblies not meeting the criteria of Table 3.7.15-7 shall be stored in accordance with either Figure 3.7.15-4 and Table 3.7.15-8 for Restricted storage, or Figure 3.7.15-5 for Checkerboard storage.

d

The restrictions on the placement of fuel assemblies within the Region 2B of the spent fuel pool, which have accumulated burnup greater than or equal to the minimum qualified burnups in Table 3.7.15-10 in the accompanying LCO, ensures the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95, assuming the pool to be flooded with water borated to 730 ppm. Fuel assemblies not meeting the criteria of Table 3.7.15-10 shall be stored in accordance with either Figure 3.7.15-6 and Table 3.7.15-11 for Restricted storage, or Figure 3.7.15-7 for Checkerboard storage.

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

BASES

LCO (continued)

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 3.7.15.1

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

REFERENCES

1. UFSAR, Section 9.1.2.
2. Issuance of Amendments, McGuire Nuclear Station, Units 1 and 2 (TAC NOS. M89744 and M89745), November 6, 1995.
3. WCAP-14416-NP-A, Westinghouse Spent Fuel Rack Criticality Analysis Methodology, Revision 1, November 1996.
4. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.
5. Nuclear Regulatory Commission, Memorandum to Timothy Collins from Laurence Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," August 19, 1998.
6. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
7. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated August 1, 2000 (Ref. 1), Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would allow the use of credit for soluble boron in the spent fuel pool (SFP) criticality analyses. These criticality analyses were performed using the methodology developed by the Westinghouse Owners Group and described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," (Ref. 2). The analyses also account for reduced credit for the degraded spent fuel rack Boraflex neutron absorber panels. Also, the licensee has performed evaluations of potential boron dilution events. Further, DEC proposes an addition to Chapter 16, "Selected Licensee's Commitments" of the McGuire Updated Final Safety Analysis Report (UFSAR) to provide for periodic monitoring of future Boroflex degradation.

In addition to the SFP analysis, the new (un-irradiated) fuel storage vaults were also re-analyzed to accommodate the new Westinghouse Performance Plus fuel design. The analysis showed that the new fuel storage racks continued to maintain k_{eff} less than or equal to 0.95 if fully flooded with unborated water, and less than or equal to 0.98 if moderated by aqueous foam. Therefore, no TS changes are required for the new fuel storage vaults.

2.0 EVALUATION

2.1 Criticality Aspects

The McGuire spent fuel storage racks were analyzed using the Westinghouse methodology, which has been reviewed and approved by the NRC (Ref. 2). This methodology takes partial credit for soluble boron in the SFP criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1) k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level as described in WCAP-14416-NP-A; and

Enclosure

2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in WCAP-14416-NP-A.

Although the referenced Westinghouse methodology was used as the basis for this analysis, some minor differences exist in the application of this methodology since the analysis was performed by Duke Energy using a different set of computer codes. Also, Section 9.1 of the McGuire UFSAR is used as the reference for a description of the 95/95 uncertainties used in the above criteria, rather than WCAP-14416-NP-A. This is in conformance with NUREG-1431, "Standard Technical Specifications Westinghouse Plants" (Ref. 3), and is acceptable.

The analysis of the reactivity effects of fuel storage in the McGuire SFP was performed with the three-dimensional Monte Carlo code, KENO-Va, and the CASMO-3 and SIMULATE-3 codes. CASMO-3 is an integral transport theory code, and SIMULATE-3 is a nodal diffusion theory code. The analytical methods and models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the McGuire storage racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions that include close proximity storage and strong neutron absorbers. We conclude that the analysis methods used are acceptable and capable of predicting the reactivity of the McGuire storage racks with a high degree of confidence.

Each McGuire SFP contains a two-region rack design. The Region 1 area is designed and generally reserved for temporary storage of new (un-irradiated) or partially irradiated fuel. The stainless steel cells are spaced at 10.4 inches and contain Boraflex attached to the exterior cell wall wrapper plate with a minimum 0.02 gm/cm^2 loading of B-10 neutron absorbing material. This region has a nominal capacity of 286 locations. Region 2 is designed and generally used for normal, long-term storage of permanently discharged fuel that has achieved qualifying burnup levels. The stainless steel cells are spaced at 9.125 inches and also contain Boraflex with a slightly lower minimum B-10 areal density (0.006 gm/cm^2). Region 2 has a nominal capacity of 1177 locations. Duke is proposing to divide each region of the SFP into two sub-regions. One sub-region would take credit for a fraction of the original amount of Boraflex and the second sub-region would take no credit for Boraflex. The McGuire spent fuel storage racks have previously been qualified for storage of various Westinghouse and Framatome fuel assembly types with maximum enrichments up to 4.75 weight percent (w/o) U-235.

All fuel designs used, or planned for use, at McGuire were analyzed. These included Westinghouse Standard, Optimized and Robust Fuel (also referred to as Performance Plus or PF+) and Framatome Mark BW fuel designs. Also included were the Oconee fuel assemblies currently stored at McGuire. All fuel designs were analyzed for all cases, and only the most reactive design was used to set the storage requirements. All conditions were modeled at both 68°F and 15°F, and only the most reactive temperature was used to set the storage requirements. The nominal coating on integral fuel burnable absorber (IFBA) rods was assumed to be 75 percent of the minimum standard loading offered by Westinghouse, referred to as 1.0X, to account for the IFBA coating not being applied for the full length of the fuel rod. The analyses accounted for the bias and uncertainty associated with the benchmarking of the methodology, a bias for the underprediction of reactivity due to boron particle self-shielding, and the uncertainty due to mechanical tolerances from the manufacturing process. Additional uncertainties related to irradiated fuel are also included with the burnup credit methodology discussed below. In addition, biases and uncertainties due to Boraflex shrinkage were included as discussed below. These uncertainties were appropriately determined at the 95/95 probability/confidence level. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The uncertainties associated with Boraflex shrinkage included a reactivity bias to account for an assumed 0.25 inches of shrinkage in the width of the panels and a reactivity uncertainty to account for the 95/95 worst case shrinkage in the axial direction. The amount of shrinkage assumed is consistent with that observed in various spent fuel storage racks and is acceptable. End pullback of the top and bottom of the Boraflex panels, as compared to gap formation, was determined to result in the largest reactivity effect.

In addition to Boraflex shrinkage, the analyses also assumed the following reductions in the original amount of Boraflex to account for degradation due to accumulated gamma irradiation and exposure to the wet pool environment. Region 1A assumed credit for 25 percent of the original Boraflex, Region 1B assumed no credit for Boraflex, Region 2A assumed credit for 50 percent of the original Boraflex, and Region 2B assumed no credit for Boraflex.

In order to verify the condition of the Boraflex in the McGuire racks, Duke periodically performs quantitative in-situ measurements using the Boron-10 Areal Density Gage for Evaluating Racks (BADGER), which was developed by Northeast Technology Corporation under contract for the Electric Power Research Institute (Ref. 4). The principle of BADGER is measurement of thermal neutron attenuation by the Boraflex panel(s) between a californium-252 (Cf-252) source and boron trifluoride (BF-3) detectors. The initial in-situ verification for the McGuire Unit 2 spent fuel racks in January of 1997 showed that the amount of Boraflex degradation in both Region 1 and Region 2 was less than the amount of degradation assumed in this criticality analysis. Starting in 2000, additional in-situ testing will be performed at a frequency of three years or less. Based on the evaluation of Boraflex degradation described in Section 2.2 of this safety evaluation, we find that the Boraflex degradation assumptions used in this criticality analysis are acceptably conservative and that the in-situ testing is appropriate to confirm that the Boraflex levels assumed in this criticality analysis remain bounding.

For Region 1A, the nominal enrichment required to maintain k_{eff} less than 1.0 with all cells filled with fresh fuel assemblies and no soluble boron in the pool water was found to be 3.78 w/o U-235. This resulted in a 95/95 k_{eff} of 0.97235. Since this value is less than 1.0 and was determined at a 95/95 probability/confidence level, it meets the NRC criterion for precluding criticality with no credit for soluble boron and is acceptable. Similar calculations for Regions 1B, 2A, and 2B resulted in unrestricted storage of fuel assemblies enriched to 1.78, 1.61, and 1.11 w/o U-235, respectively.

Restricted storage configurations must be employed for assemblies that do not qualify for unrestricted storage. A calculation was done for Region 1A assuming a 3-out-of-4 assembly storage configuration with three fresh assemblies and a fourth low-reactivity filler assembly. This loading pattern is shown in proposed TS Figure 3.7.15-1. The nominal enrichment of the fresh fuel required to maintain k_{eff} less than 1.0 for this configuration and no soluble boron was found to be 4.75 w/o U-235. The maximum allowed enrichment of the filler assembly was 1.76 w/o U-235.

Restricted storage configurations for Region 1B resulted in a 2-out-of-4 pattern with two 2.20 w/o assemblies and two 1.44 w/o filler assemblies, as shown in proposed TS Figure 3.7.15-2. Restricted storage configurations for Region 2A also resulted in a 2-out-of-4 pattern with two 2.12 w/o assemblies and two 1.20 w/o assemblies, as shown in proposed TS Figure 3.7.15-4. Restricted storage configurations for Region 2B resulted in a 1-out-of-4 pattern with one 1.22 w/o assembly and three 1.08 w/o assemblies, as shown in proposed TS Figure 3.7.15-6.

In order to store fuel with enrichments higher than the maximum enrichment limits for fresh fuel, the concept of reactivity equivalencing was used. In this manner, the negative reactivity from fuel burnup is used to offset the positive reactivity from higher enrichments until the reactivity is equivalent to that of the fresh fuel maximum enrichment case (i.e., the no boron 95/95 maximum design k_{eff}). The NRC has previously accepted the use of reactivity equivalencing predicated upon the reactivity decrease associated with fuel depletion. A bias was applied to account for the reactivity increase due to the removal of burnable poison from an assembly after its first cycle of operation. Since most calculations are two-dimensional (i.e., no axial effects are modeled), a reactivity bias is included in the two-dimensional calculations to account for burnup distribution differences between two-dimensional and three-dimensional modeling. The minimum burnup requirements for each initial enrichment was generated for each type of storage and each region to yield a curve of acceptable storage. These minimum burnup requirements are shown in proposed TS Tables 3.7.15-2 through 3.7.15-12.

Reactivity equivalencing was also used to take credit for the IFBA rods. In this technique, which has been previously approved by the NRC, credit is taken for the reactivity decrease due to the ZrB_2 material coated on the outside of the UO_2 pellet. In this technique, the enrichment is varied until the calculated k_{eff} is equivalent to that of the fresh fuel maximum enrichment case (i.e., the no boron 95/95 maximum design k_{eff}). The calculated k_{eff}s are used to determine maximum enrichment for each discrete number of IFBA rods to ensure that the 95/95 storage rack k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron. These IFBA rod requirements are shown in proposed TS Table 3.7.15-1.

Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including 95/95 uncertainties. The soluble boron credit calculations determined the amount of boron required for both unrestricted and restricted storage in each region. As previously described, the individual tolerances and uncertainties, and the temperature and methodology biases, were added to the calculated nominal k_{eff} to obtain a 95/95 value. The resulting 95/95 k_{eff} values were less than 0.95, satisfying the NRC acceptance criterion for precluding criticality. The maximum amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 under normal conditions is 730 ppm. This is well below the minimum SFP boron concentration value of 2675 ppm required by TS 3.7.14 and is, therefore, acceptable.

Although most accidents will not result in a reactivity increase, three accidents can be postulated for each storage configuration which would increase reactivity beyond the analyzed conditions. The first is a drop or placement of a fuel assembly into the cask loading area. The second is a significant change in the SFP water temperature such as a large makeup to the pool with cold water which causes a decrease in the pool water temperature. The third is the misloading of a fuel assembly into a location for which the restrictions on location, enrichment, burnup and number of IFBA rods are not satisfied.

Calculations have shown that the most severe accident would be the misloading of the highest reactive assembly allowed in the pool (fresh 4.75 w/o assembly) in place of the lowest reactive assembly (filler assembly). For such events, the double contingency principle (Ref. 5) can be applied. This states that the assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident These calculations show that the total amount of soluble boron required to offset the maximum reactivity addition accident and to maintain k_{eff} less than or equal to 0.95 is 1470 ppm. Therefore, the minimum amount of boron required by TS 3.7.14 (2675 ppm) is more than sufficient to cover any accident and the presence of the additional boron above the concentration required for normal conditions and reactivity equivalencing can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

In order to prevent an undesirable increase in reactivity, the boundaries between the different storage configurations were analyzed. The results of this analysis show that (1) any restricted Region 1A storage area row bounded by any other storage area must contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other, (2) any restricted Region 2B storage area row bounded by any other storage area must contain only filler locations arranged such that no restricted fuel assemblies are adjacent to any other fuel except Region 2B filler locations, and (3) any checkerboard Region 2B storage area row bounded by any other storage area must contain only empty cells arranged such that no checkerboard fuel assemblies are adjacent to any fuel. These interface restrictions are shown in TS Figures 3.7.15-1, 3.7.15-6, and 3.7.15-7, respectively.

The proposed changes to TS 3.7.15, "Spent Fuel Assembly Storage," and TS 4.3, "Fuel Storage," are consistent with the revised criticality analysis and with the NRC-approved methodology given in Westinghouse topical report, WCAP-14416-NP-A, Rev. 1, (Ref. 2). Based on this consistency with the approved methodology and on the above evaluation, we find these TS changes acceptable. The proposed associated Bases changes adequately describe these TS changes and are also acceptable.

The staff finds the criticality aspects of the proposed McGuire license amendment request are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The analysis assumed credit for soluble boron, as allowed by WCAP-14416-NP-A, and partial or no credit for the Boraflex neutron absorber panels. We find the proposed amounts of Boraflex remaining in the racks acceptably conservative. In addition, we find the proposed in-situ testing necessary and appropriate to confirm that the Boraflex levels remain bounding. Therefore, we conclude that the proposed amounts of Boraflex remaining in the racks are acceptable. The criticality analysis conformed to the NRC guidance on the regulatory requirements for criticality analysis of fuel storage at light-water reactor power plants (Ref. 5).

2.2 Boraflex Degradation

The licensee proposes to verify the amount of Boraflex in the SFP racks through periodic, quantitative, in-situ measurements. These measurements will be obtained through the BADGER system (Ref. 4). This system measures the B-10 areal density in the spent fuel racks. In addition, the computer code, RACKLIFE (Ref. 6), will be used to estimate the future condition of the Boraflex through 2003 and to determine which Boraflex panels will be in-situ tested every three years.

The proposed assumptions of Boraflex panel losses for use in the McGuire criticality calculations are summarized in the following table. The assumed Boraflex panel losses are expressed as a percentage of the minimum as-built B-10 areal density for the subregions.

Proposed Assumptions for Boraflex Panel Loss	
Region 1A	75%
Region 1B	100%
Region 2A	50%
Region 2B	100%

The system consists of a source head containing Cf-252 and a detector head containing four BF-3 detectors. The source and detector heads are constructed from aluminum boxes with tapered lead-ins at the bottom. The aluminum source tube and the encapsulated BF-3 detectors are water-tight.

The BADGER system is based on the principle of attenuation of neutrons through the Boraflex panel between the source and the detectors. A portion of the fission neutrons produced by the source is thermalized in a media of high density polyethylene. The number of thermal neutrons reaching the detectors is inversely proportional to the amount of B-10 atoms (B-10 areal density) in the Boraflex panels; i.e., the higher the detector signal, the lower the B-10 areal density. The system is calibrated at the start of a shift by measuring signals through a cell containing Boraflex sections of known B-10 areal densities.

Signals are obtained after the source and detector heads are lowered to the bottom of the cell. The heads are moved in increments equal to two inches. Counts are taken for 10 or 15 seconds at each elevation with a longer interval for Boraflex panels with higher areal densities. The movement of the probes, the counting, and the recording are fully automated and controlled by computer.

RACKLIFE is a PC-based computer code that computes the in-service degradation of Boraflex panels. Developed by NETCO in collaboration with the nuclear utilities' Boraflex Users' Group (BUG) now known as the Wet Storage Users' Group and sponsored by EPRI, RACKLIFE models the service life of up to 10,000 Boraflex panels and inventories the storage of up to 5000 fuel assemblies.

Factors affecting Boraflex degradation include: water temperature, absorbed gamma dose, pool pH, rack design, cleanup and make-up system operation. The RACKLIFE program allows for the input of these factors and models the following phenomena: silica kinetics and pool transport, Boraflex panel absorbed gamma dose, boron carbide loss from Boraflex panels, silica source term, polymerization of silica, panel cavity to pool volume exchange, and clean-up systems. Calculations based on this information give: time dependent pool silica concentration, individual panel gamma exposure, and individual panel boron carbide loss.

RACKLIFE performs a mass balance of SiO_2 in the pool and within the wrapper plate plenum that encapsulates the Boraflex panel. All factors being equal, Boraflex panels with higher gamma exposure have higher SiO_2 releases. For Boraflex panels with equal doses, a higher

SiO_2 release is expected from those panels exposed early in life. RACKLIFE calculates the amount of boron carbide released based on the fixed ratio of boron carbide to SiO_2 in the panels.

A key input into RACKLIFE is the escape coefficient. This coefficient is associated with the rate at which the SFP water moves through the wrapper that encapsulates the Boraflex panel. A higher escape coefficient indicates a more "open" wrapper to the pool water. This results in a greater degradation rate of the Boraflex panel.

In January 1997, BADGER testing was demonstrated in the McGuire Unit 2 SFP. The spent fuel storage racks tested had previously housed fuel assemblies. The assemblies were relocated prior to testing. In addition, a RACKLIFE model of the Unit 2 racks was established based on plant data. An actual service history was generated for each Boraflex panel in the Unit 2 SFP. The racks with the highest dose were located in Region 1 close to the fuel transfer canal. These racks receive fuel first during core offload operations. Region 2 racks generally receive low doses due to their location and have not been used repeatedly to store freshly discharged fuel. A total of 33 panels were tested.

The results of the BADGER test are as follows:

- For Region 1, the 15 panels tested had a range of Boraflex loss from 0 (un-irradiated panel) to 33.33 percent. There was a clear trend for greater loss with increasing gamma exposure.

- For Region 2, the 18 panels tested had a range of Boraflex loss from 0 (un-irradiated panel) to 15.85 percent. There was no clear association between Boraflex loss and gamma exposure.

- Although not a primary function of BADGER, gap measurements were taken. Region 1 panels were found to have one to two gaps per panel. Region 2 panels were found to have three to four gaps per panel. The gaps did not exceed four inches and were randomly distributed with no preferential elevation for formation.

The RACKLIFE results for McGuire Unit 2 are presented below. These are the worst Boraflex panel losses and are expressed as a percentage of the minimum as-built B-10 areal density for the sub-regions. The following table summarizes the RACKLIFE prediction results for each sub-region for the given projected dates.

RACKLIFE Worst Boraflex Panel Loss for McGuire Unit 2			
	January 8, 1997	December 31, 1999	December 31, 2003
Region 1A	34%	50%	81%
Region 1B	41%	60%	97%
Region 2A	19%	21%	22%

These results were obtained using the following escape coefficients: 1.25 for Region 1 and 0.05 for Region 2. These coefficients provide the best match between the BADGER results taken in January 1997 and the RACKLIFE predictions for that time.

In-situ testing with BADGER was not performed for McGuire Unit 1. Therefore, BADGER results for comparison with RACKLIFE predictions are not available. However, since the design and construction of the Unit 1 storage racks are identical to those of Unit 2, the licensee applied the escape coefficients and adjustments for the Unit 2 model to the Unit 1 model. The following table summarizes the results.

RACKLIFE Worst Boraflex Panel Loss for McGuire Unit 1 December 31, 2003	
Region 1A	64%
Region 2A	21%

The licensee requests changes to their operating licenses and TSs to credit the remaining Boraflex in the SFP racks. These changes allow for flexibility in maintaining subcriticality in the McGuire SFPs. Specifically, the licensee proposes to credit the following amounts of Boraflex remaining in the SFP racks through 2003:

Amount of Boraflex Remaining through 2003 McGuire Units 1 and 2		
	Percentage of Boraflex Remaining	Minimum B-10 areal density, gm B-10/cm
Region 1A	25%	0.005
Region 1B	0%	0
Region 2A	50%	0.003
Region 2B	0%	0

The licensee's plans to verify the RACKLIFE prediction after December 31, 1999, will help establish the reliability of the projected amounts of Boraflex degradation and the validity of the escape coefficients used. The staff has determined that the licensee's proposed revision to the McGuire UFSAR Chapter 16 is appropriate and acceptable in establishing a conservative approach to verifying projected Boraflex degradation. This revision in Section 16.9-9, "Spent Fuel Pool Storage Rack Poison Material," includes a testing requirement which verifies, every three years, that the panel average spent fuel pool storage rack poison is within the proposed limits. Verification of the poison material is conducted through the BADGER system.

The staff notes that the RACKLIFE December 31, 2003 prediction of 19% Boraflex remaining for Unit 2 Region 1A is below the proposed 25% Boraflex remaining used in the criticality analysis. However, this prediction is for, at most, one panel in Region 1A. The staff has determined that the proposed 25% Boraflex remaining in Unit 2 Region 1A is still conservative in light of the 19% prediction because of the vast number of Boraflex panels present in this region of the spent fuel pool. In addition, the licensee will have the opportunity to verify this prediction before December 31, 2003. In the event that the proposed amounts of Boraflex remaining no longer support the McGuire criticality analysis, the licensee has stated that a future amendment proposing additional TS changes will be submitted to maintain acceptable levels of subcriticality in the McGuire spent fuel storage pools (Ref. 1).

The staff finds the proposed amounts of Boraflex remaining in the racks acceptably conservative. In addition, the staff finds the proposed in-situ testing necessary and appropriate to confirm that the Boraflex levels remain bounding. The staff concludes, therefore, that the proposed amounts of Boraflex remaining in the racks at McGuire Units 1 and 2, are acceptable.

2.3 Boron Dilution Event Analysis

Deterministic dilution event calculations were performed by the licensee for McGuire Units 1 and 2 to define the dilution times and volumes necessary to dilute the SFP from an initial boron concentration of 2,475 ppm to a minimum soluble boron concentration (730 ppm) required to maintain the SFP at k_{eff} less than 0.95. The initial boron concentration of 2,475 ppm corresponds to the core operating limit report (COLR) for Unit 1 Cycle 12, which is the lowest limit currently in use at McGuire. However, the COLR for Unit 1 Cycle 13 is to be raised to 2,675 ppm to match that of McGuire Unit 2. Therefore, the proposed TS limit for the minimum boron concentration of 2,675 ppm is to be applied for McGuire Units 1 and 2. For the purpose of this submittal, DEC conservatively chose the initial boron concentration of 2,475 ppm for the boron dilution event.

Systems that interface with the refueling cooling system may be misaligned because of operator errors, or component malfunction could cause unborated water to be added to the SFP. These interfaced systems include the refueling water system, the boron recycle system, the liquid waste recycle system, the chemical and volume control system, the makeup demineralized water system, the filtered water system, the drinking water system, the nuclear service water system, and the component cooling water system. The licensee compiled and evaluated these potential dilution sources to identify the bounding dilution event. As a result, the credible worst-case event in this category involves the dilution path from the McGuire recycle holdup tanks (RHTs) through the recycle evaporator feed pumps to the reactor makeup water storage tank (RMWST), and "piggy backing" the reactor makeup pumps into the SFP. The combined total volume is 336,000 gallons (two RHTs and one RMWST).

The maximum dilution resulting from this event reduces boron concentration from 2,475 ppm to 1,068 ppm. However, the maximum flow rate is limited to 60 gpm by the two recycle evaporator feed pumps and is estimated to take more than 60 hours to dilute the SFP to 1,068 ppm. Also, the licensee analyzed the same case event during infrequent plant configurations in which the cask loading pit was isolated. The maximum dilution resulting from this event reduces boron concentration from 2,475 ppm to 937 ppm. This concentration level is still much higher than the minimum required level of 730 ppm.

Other events that may affect the boron concentration of the SFP such as pipe cracks, loss of offsite power, and infrequent SFP configurations, were also evaluated. Random pipe break sizes were considered using the method required in the plant final safety analysis report, Section 3.6.2.2, for high-energy and moderate-energy systems.

The licensee stated that both McGuire SFPs are located at an elevation above all adjacent buildings. Therefore, water cannot flow into the pool from pipe breaks in these buildings and due to this fact, pipe breaks in these buildings are excluded from the boron dilution evaluation. However, through the review of plant drawings and from a plant walkdown inspection, the following piping systems in the SFP area, including the fire protection system, the demineralized water supply system, and the drinking water supply system, if broken during normal plant configurations, could introduce flow into the SFP. Of these three systems, a pipe break from the non-seismic fire protection system is considered a worst-case line break during infrequent plant configurations with a spent fuel pit isolated, resulting in a flow rate of approximately 700 gpm of non-borated water into the SFP. It is estimated that the SFP would be diluted to 730 ppm in more than 11 hours, and more than 550,000 gallons of water would be added to the pool. We performed an independent verification calculation of the event and confirmed the licensee's results.

The licensee evaluated other dilution events resulting either from normal plant configurations or infrequent plant configurations and determined that they take a much longer time to reach the minimum boron concentration. The licensee stated that all dilution events would be readily detected by normal operator rounds through the SFP area, by level alarms, and by flooding in the auxiliary building. In the event of a fire protection system pipe break, control room alarms would provide an indication that one or more fire protection pumps had started. Also, flow alarms on the fire protection pump headers would indicate to the operators that the flow was going into the auxiliary building. To detect low flow, long-term dilution events, plant TSs require that the SFP be sampled every seven days. This frequency is consistent with the standard TSs for Westinghouse plants and is considered appropriate for the McGuire plant.

The licensee concluded that an unplanned or an inadvertent event during infrequent plant configurations that dilute the SFP boron concentration from 2,475 ppm to less than 730 ppm is not a credible event because of the very low frequency of the configuration. The event would be readily detected by plant personnel through alarms, flooding in the auxiliary buildings, or by normal operator rounds through the SFP area. In addition, the licensee committed to raise the Unit 1 SFP boron concentration to 2,675 ppm to match that of McGuire Unit 2.

Considering the higher boron concentration, the large volume of water required for a dilution event, alarms, TS-controlled SFP concentration and the 7-day sampling requirement, and plant personnel rounds, the staff finds that there is adequate assurance that a dilution event would be detected before k_{eff} of 0.95 (730 ppm) is reached. Therefore, the analysis and the proposed TS controls are acceptable for the boron dilution aspects of the request.

Additionally, the criticality analysis for the SFP demonstrated that k_{eff} remains less than 1.0 at a 95/95 probability/confidence level, even with non-borated water in the SFP. Therefore, even if the spent fuel pool was diluted to approximately 0 ppm, the spent fuel in the racks would remain subcritical, in conformance with General Design Criterion 62.

The staff finds that the boron dilution aspects of the proposed McGuire license amendment request are acceptable. The TS boron concentration of 2,675 ppm and the 7-day surveillance requirements are acceptable for ensuring that sufficient time is available to detect and mitigate a dilution event before the design basis k_{eff} of 0.95 is exceeded.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (65 FR 62385). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from H.B. Barron, Duke, to U.S. Nuclear Regulatory Commission, Proposed Technical Specification (TS) Amendments, TS 3.7.15-Spent Fuel Assembly Storage, TS 4.3-Fuel Storage, August 1, 2000.

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

3. NUREG-1431, Vol. 1, Rev. 1, "Standard Technical Specifications Westinghouse Plants," April 1995.

4. EPRI TR-107335, "BADGER, a Probe for Nondestructive Testing of Residual Boron-10 Absorber Density in Spent-Fuel Storage Racks: Development and Demonstration," October 1997.

5. U.S. Nuclear Regulatory Commission, Memorandum to Timothy Collins from Laurence Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," August 19, 1998.

6. EPRI TR-109926, “The Boraflex Rack Life Extension Computer Code - RACKLIFE,” March 1999.

Principal Contributors: L. Kopp
D. Diec
C. Lauron
F. Rinaldi

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McGuire Nuclear Station

cc:

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

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

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

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

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

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

Mr. Steven P. Shaver
Senior Sales Engineer
Westinshouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

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

L. A. Keller
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

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

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

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