Duke Power®
A Duke Energy Company

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September 29, 2003

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

SUBJECT: Duke Energy Corporation (Duke)
McGuire Nuclear Station Units 1 and 2
Docket Nos. 50-369 / 50-370
Proposed Technical Specification (TS) Amendments
TS 3.7.15 - Spent Fuel Assembly Storage, and TS 4.3 - Fuel Storage

Reference: NRC letter to Duke dated January 31, 2003

Pursuant to 10 CFR 50.90 and 10 CFR 50.4, this letter submits a license amendment request (LAR) for the McGuire Nuclear Station Facility Operating Licenses (FOL) and TSs. Duke met with the NRC in White Flint on December 10, 2002, March 6, 2003 and May 20, 2003 to discuss the basis for this LAR, and also Duke's corrective actions to address the spent fuel storage issues at McGuire.

This LAR will change the McGuire TS 3.7.15 to provide revised spent fuel pool storage criteria based upon fuel type, fuel enrichment, burnup, cooling time and partial credit for soluble boron in the spent fuel pool water. This LAR also allows for the safe storage of fuel assemblies with a nominal enrichment of U-235 up to 5.00 weight percent. In addition, this LAR decreases the required soluble boron credit from 850 ppm boron to 800 ppm boron in McGuire TS 4.3.1, which continues to provide an acceptable margin of subcriticality in the McGuire spent fuel storage pools. This proposed amendment is applicable to Facility Operating Licenses NPF-9 and NPF-17 for the McGuire Nuclear Station.

As discussed previously with the NRC Staff, Region 1 of both units has been reracked with new racks of equivalent design from Holtec. These new Region 1 racks utilize Boral neutron poison

A001

material instead of Boraflex. Region 1 was reracked without prior NRC approval in accordance with the stipulations as set forth in 10 CFR 50.59. Region 2 will continue to use the Westinghouse racks that utilizes Boraflex neutron poison material. This submittal assumes full credit for the Boral neutron poison material for Region 1 and no credit for any remaining Boraflex in Region 2. Upon approval of this LAR, the commitment to perform "blackness testing" of the Boraflex panels as stipulated in Selected Licensees Commitment 16.9.24, "Spent Fuel Pool Storage Rack Poison Material", will no longer be performed.

The letter referenced above issued a temporary exemption to 10 CFR 70.24, "Criticality Accident Requirements", which expires December 31, 2005. However, in accordance with 10 CFR 50.68, Duke will comply with the requirements of 50.68(b) in lieu of maintaining a monitoring system capable of detecting a criticality event as described in 10 CFR 70.24. Upon approval of this submittal an exemption to 10 CFR 70.24 will no longer be necessary.

Attachment 1 provides marked up pages of the existing McGuire TSs showing the proposed changes. Attachment 2 contains the new McGuire TS pages. The Description of Proposed Changes and Technical Justification is provided in Attachment 3. Pursuant to 10 CFR 50.92, Attachment 4 documents the determination that this proposed amendment contains no significant hazards considerations. Pursuant to 10 CFR 51.22 (c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment or Impact Statement. A summary of the McGuire Spent Fuel Pool Criticality Analysis is shown in Attachment 6. Attachments 7 and 8 show the proposed and revised BASES for TS 3.7.14 and 3.7.15.

Implementation of this amendment to the McGuire FOLs and TSs will impact the station's UFSAR. Consequently, upon approval of this LAR, the applicable revisions will be included in a McGuire UFSAR update.

In accordance with Duke internal procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the McGuire Station's Plant Operations Review Committee and the Duke Nuclear Safety Review Board. Pursuant to 10 CFR 50.91, a copy of this LAR is being forwarded to the appropriate North Carolina state officials.

Consequently, Duke requests approval of this LAR by October 1, 2004. As indicated in the attached "No Significant Hazards Consideration Evaluation" the proposed changes in this LAR will not result in a significant reduction in the facility's margin of safety.

Please contact Norman T. Simms of Regulatory Compliance at 704-875-4685 with any questions regarding this LAR.

Very truly yours,

Gary R. Peterson

Attachments

xc: (w/attachments)

L.A. Reyes
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U.S. Nuclear Regulatory Commission
Atlanta Federal Center
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J.B. Brady
NRC Senior Resident Inspector
McGuire Nuclear Station

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Beverly O. Hall, Section Chief
Radiation Protection Section
1645 Mail Service Center
Raleigh, N.C. 27699-1645

G.R. Peterson, being duly sworn, states that he is Vice President of McGuire Nuclear Station; that he is authorized on the part of Duke Energy Corporation to sign and file with the U.S. Nuclear Regulatory Commission these revisions to the McGuire Nuclear Station Facility Operating Licenses Nos. NPF-9 and NPF-17; and, that all statements and matters set forth therein are true and correct to the best of his knowledge.

G.R. Peterson, Vice President
McGuire Nuclear Station
Duke Energy Corporation

Subscribed and sworn to before me on September 29, 2003.

Notary Public

My Commission Expires: August 17, 2006

bxc: (w/ attachments)

S.W. Moser (MG05EE)
K.L. Crane (MG01RC)
G.D. Gilbert (CN01RC)
L.E. Nicholson (ON03RC)
P.F. Guill (MG05EE)
J.I. Glenn (MG05EE)
D.C. Jones (EC08F)
J.P. Coletta (EC08F)
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G.A. Copp (EC09A)
C.J. Thomas (MG01RC)
N.T. Simms (MG01RC)
ELL (EC050)
NSRB Support Staff (EC05N)
Masterfile 1.3.2.9

ATTACHMENT 1

MARKUPS TO THE MCGUIRE TECHNICAL SPECIFICATIONS

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment, burnup and cooling time for ~~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:

See Insert (a)

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.~~

See Insert (b)

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 Ta~~ble ~~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

Insert ⓐ

New or irradiated fuel may be allowed for unrestricted storage in **Region 1** of the spent fuel pool provided the maximum initial U-235 enrichment of the fuel is $\leq$ 5.0 weight percent; or

Insert ⓑ

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

1. Unrestricted storage of fuel meeting the criteria of Table 3.7.15-1; or
2. Restricted storage in accordance with Figure 3.7.15-1 of fuel meeting the criteria of Table 3.7.15-2 (Restricted Fuel assembly) and Table 3.7.15-3 (Filler Fuel assembly); or
3. Checkerboard storage in accordance with Figure 3.7.15-2 of fuel meeting the criteria of Table 3.7.15-4.

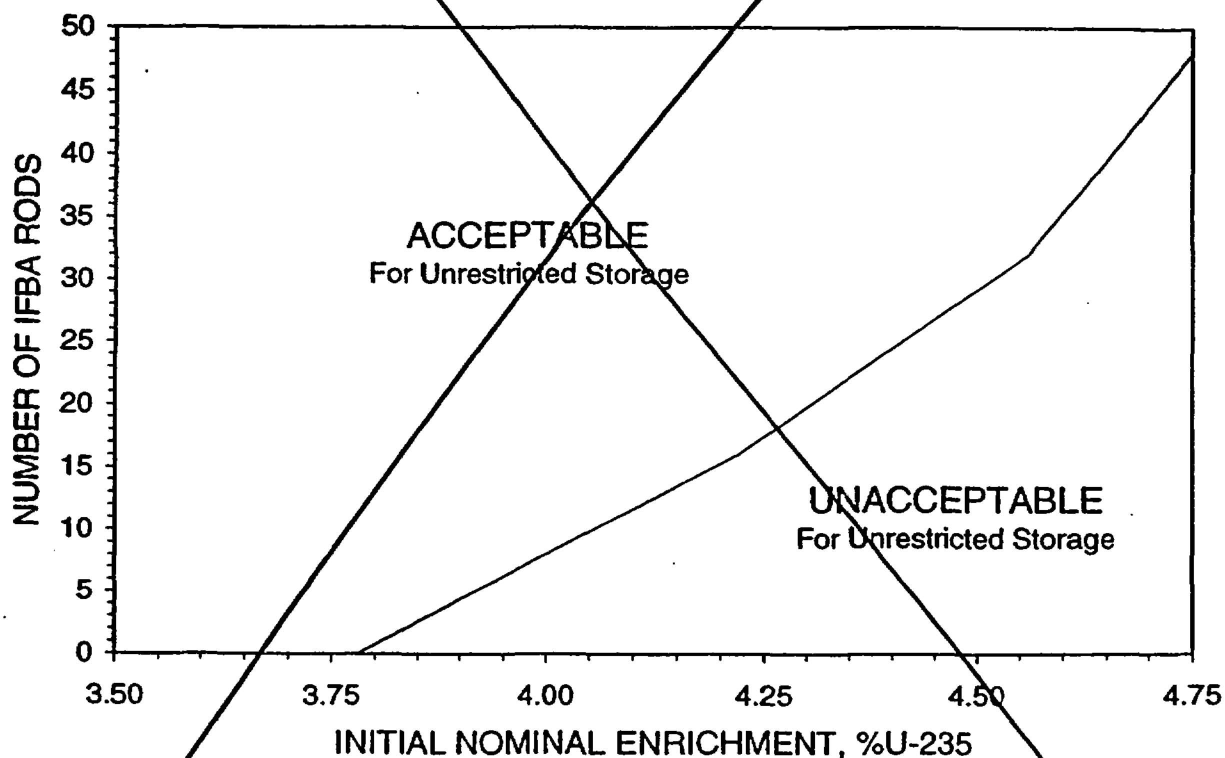
Delete all existing Tables 3.7.15-1 through 3.7.15-12 and replace with new Tables 3.7.15-1 through 3.7.15-4 (see Attachment 2)

Delete all existing Figures 3.7.15-1 through 3.7.15-7 and replace with new Figures 3.7.15-1 through 3.7.15-2 (see Attachment 2)

Delete this Table

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.

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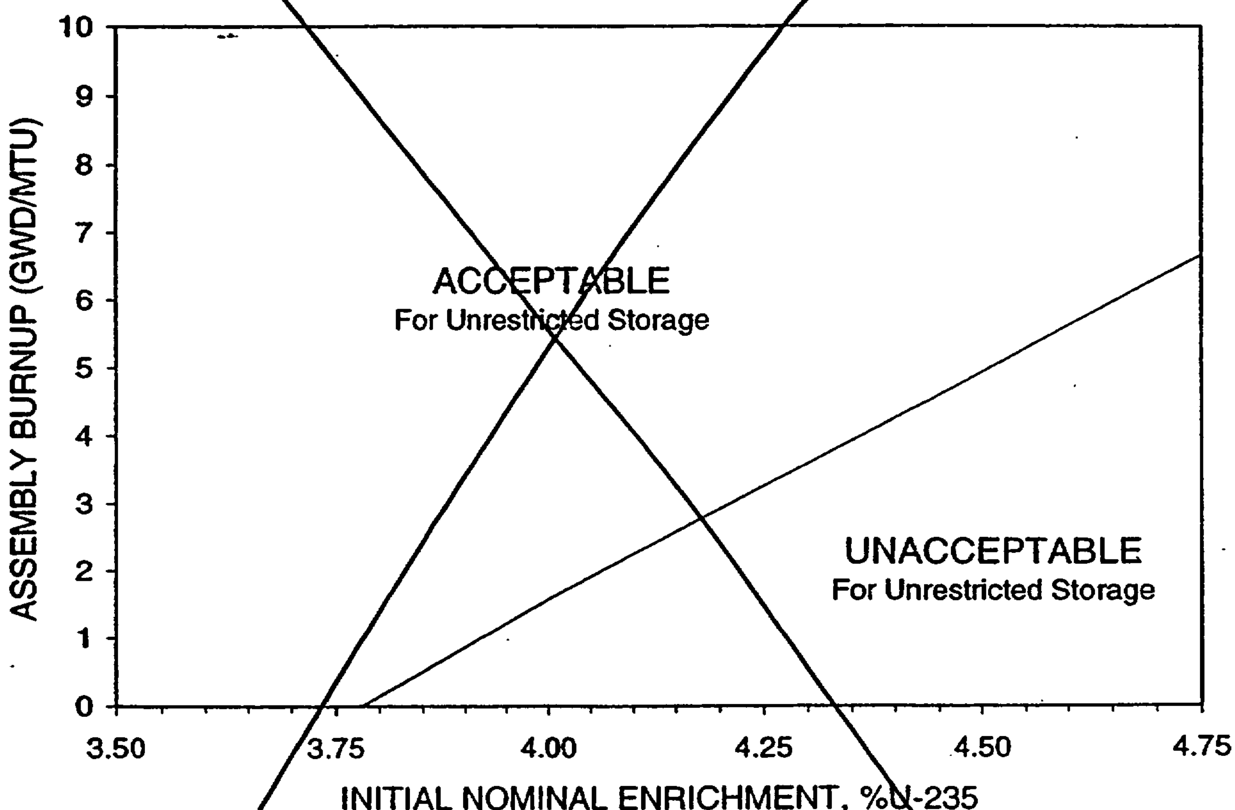
Table 3.7.15-2 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 1A Storage

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.

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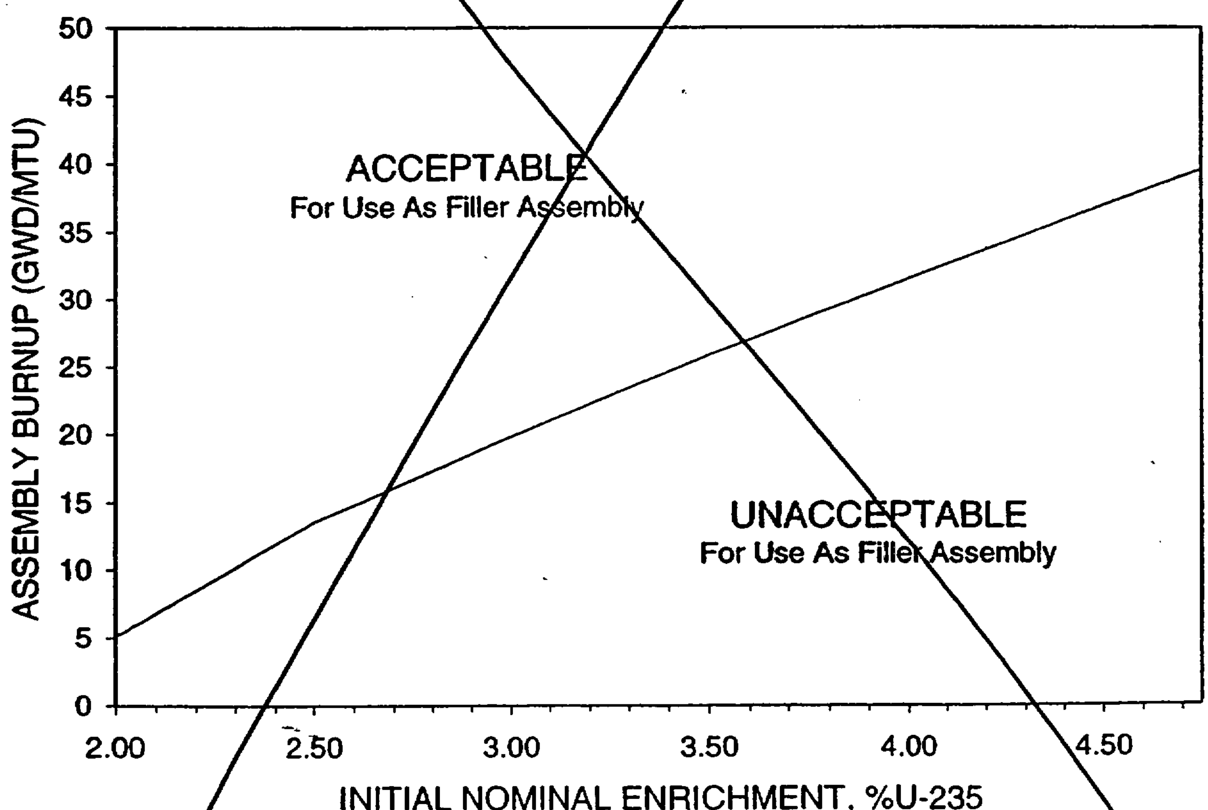
Table 3.7.15-3 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 1A Filler Assemblies

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.

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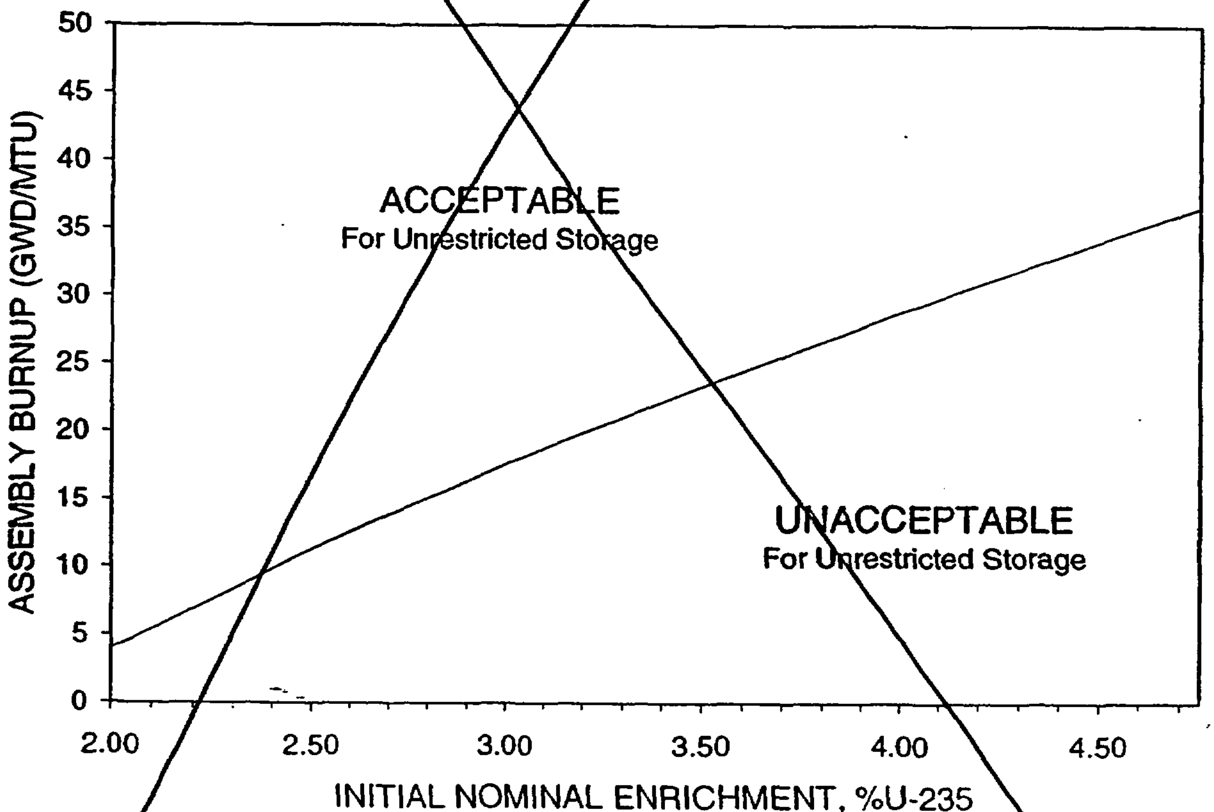
Table 3.7.15-4 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 1B Storage

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.

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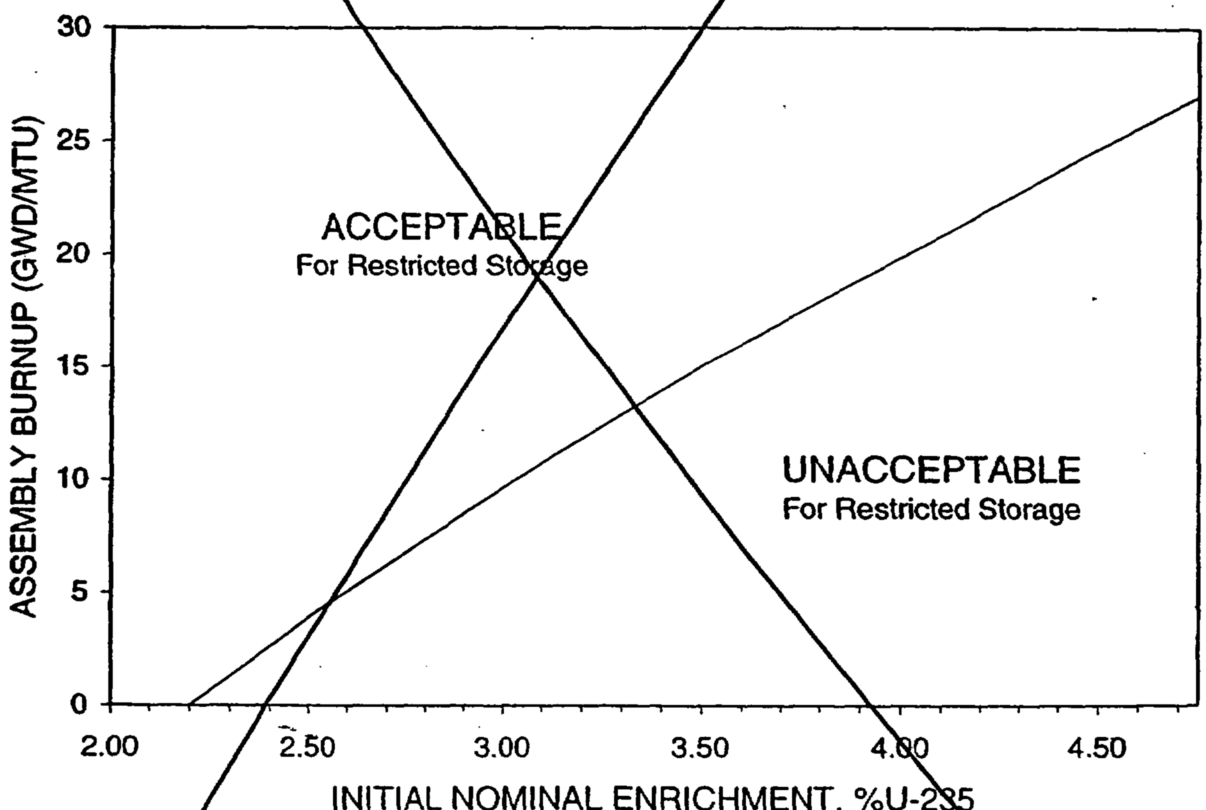
Table 3.7.15-5 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Restricted Region 1B Storage with Fillers

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.

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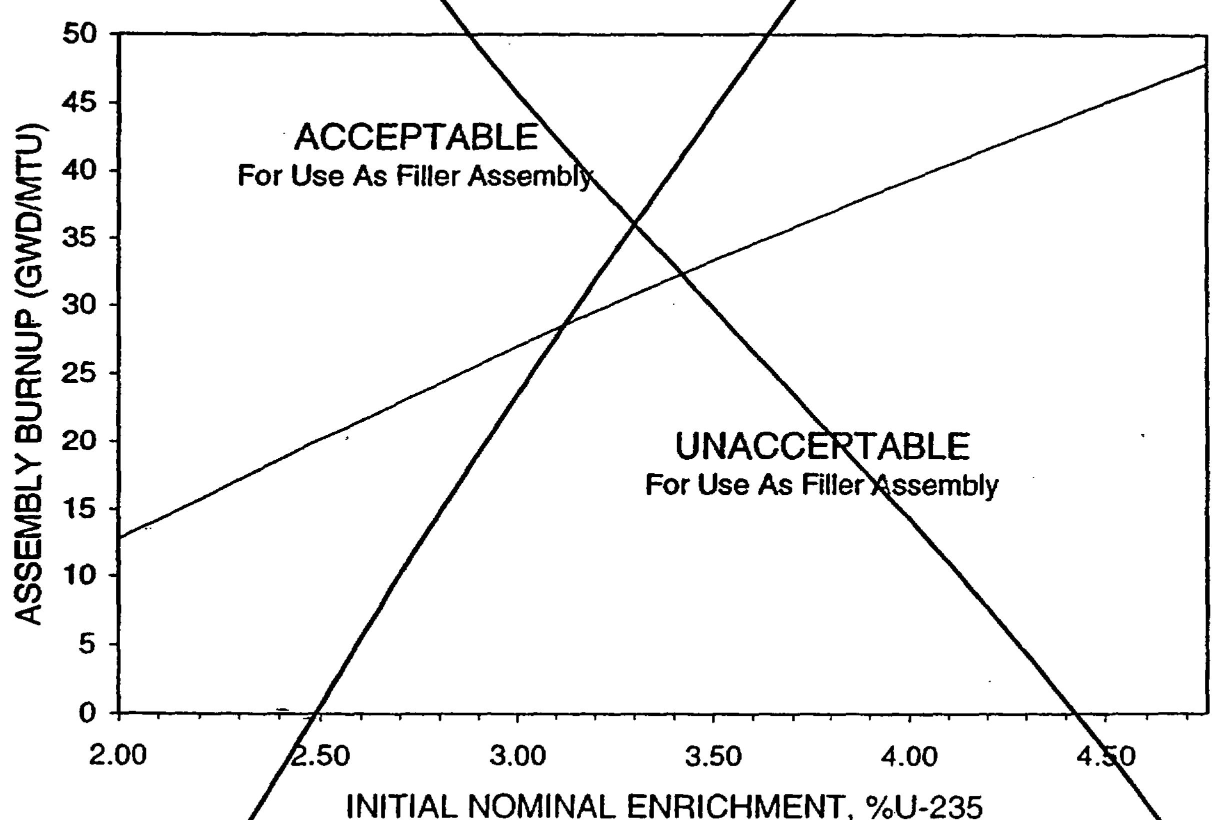
Table 3.7.15-6 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 1B Filler Assemblies

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.

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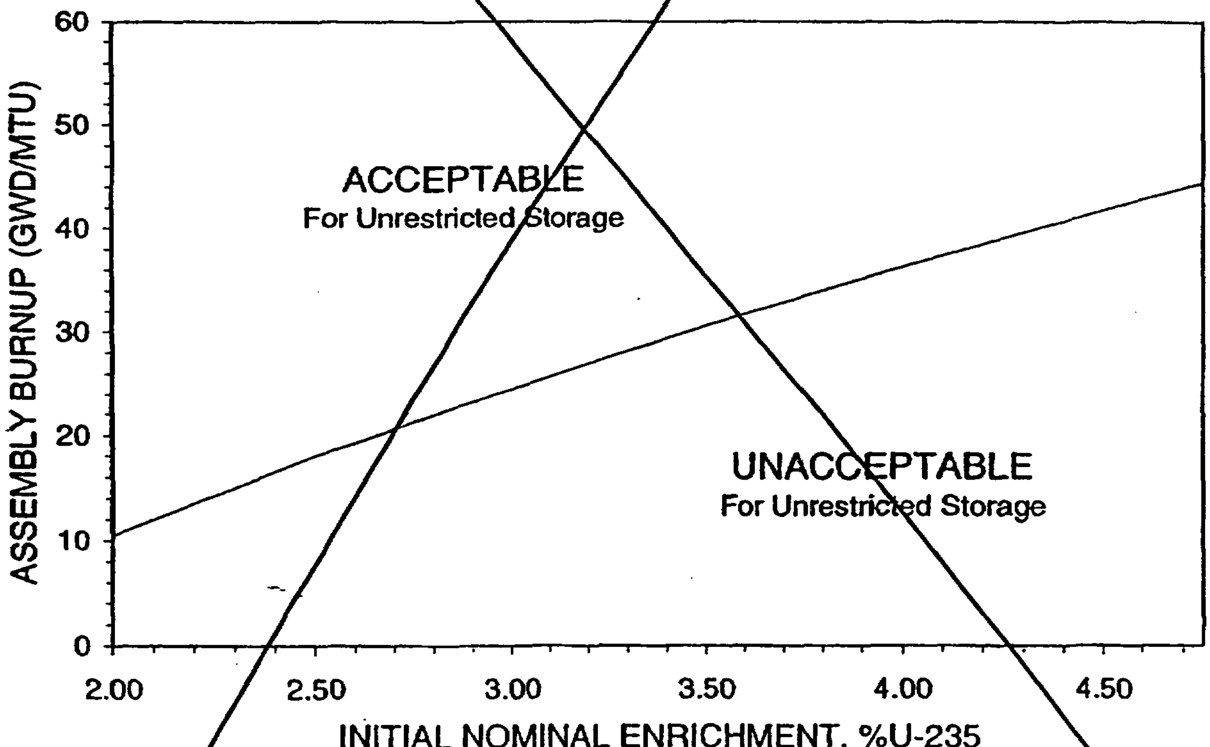
Table 3.7.15-7 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 2A Storage

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.50 (or less)	0.00
2.00	10.50
2.50	17.97
3.00	24.49
3.50	30.55
4.00	36.25
4.50	41.71
4.75	44.39



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.

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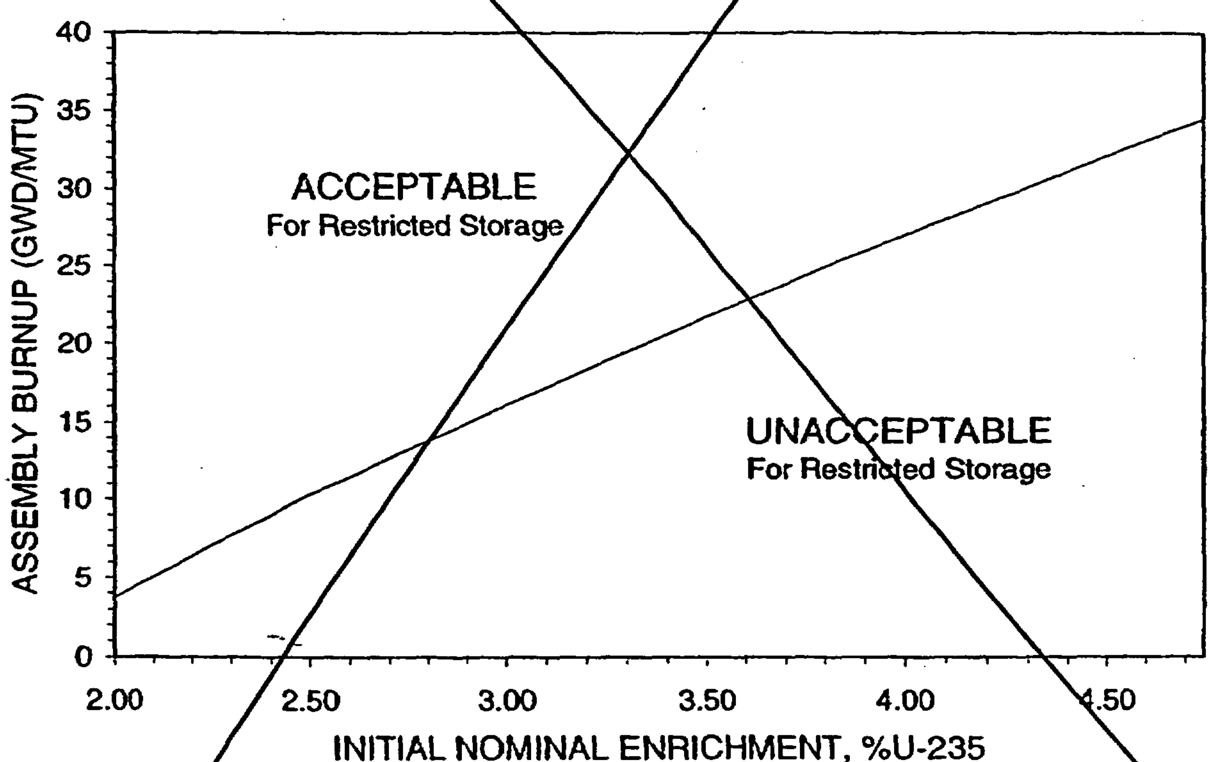
Table 3.7.15-8 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Restricted Region 2A Storage with Fillers

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.80 (or less)	0.00
2.00	3.70
2.50	10.30
3.00	16.10
3.50	21.70
4.00	27.00
4.50	32.10
4.75	34.50



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.

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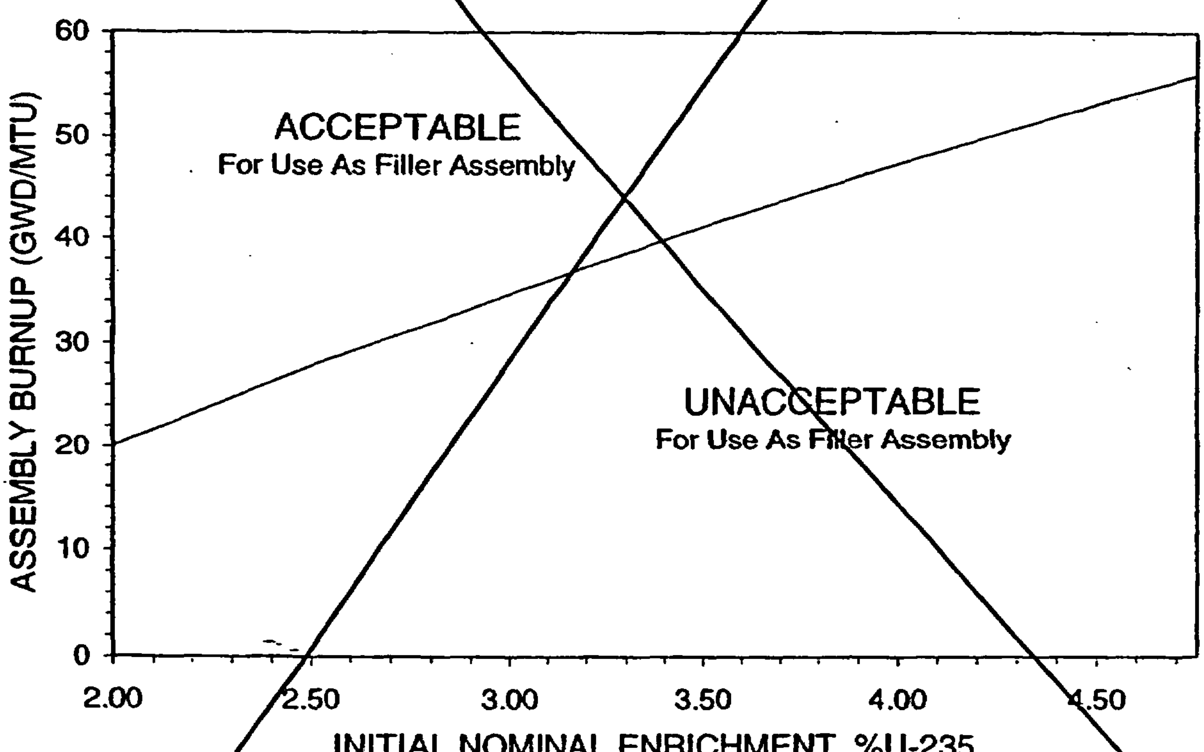
Table 3.7.15-9 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 2A Filler Assemblies

Initial Nominal Enrichment (% U-235)	Assembly Burnup (GWD/MTU)
1.15 (or less)	0.00
2.00	20.00
2.50	27.80
3.00	34.60
3.50	41.10
4.00	47.20
4.50	53.10
4.75	55.90



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.

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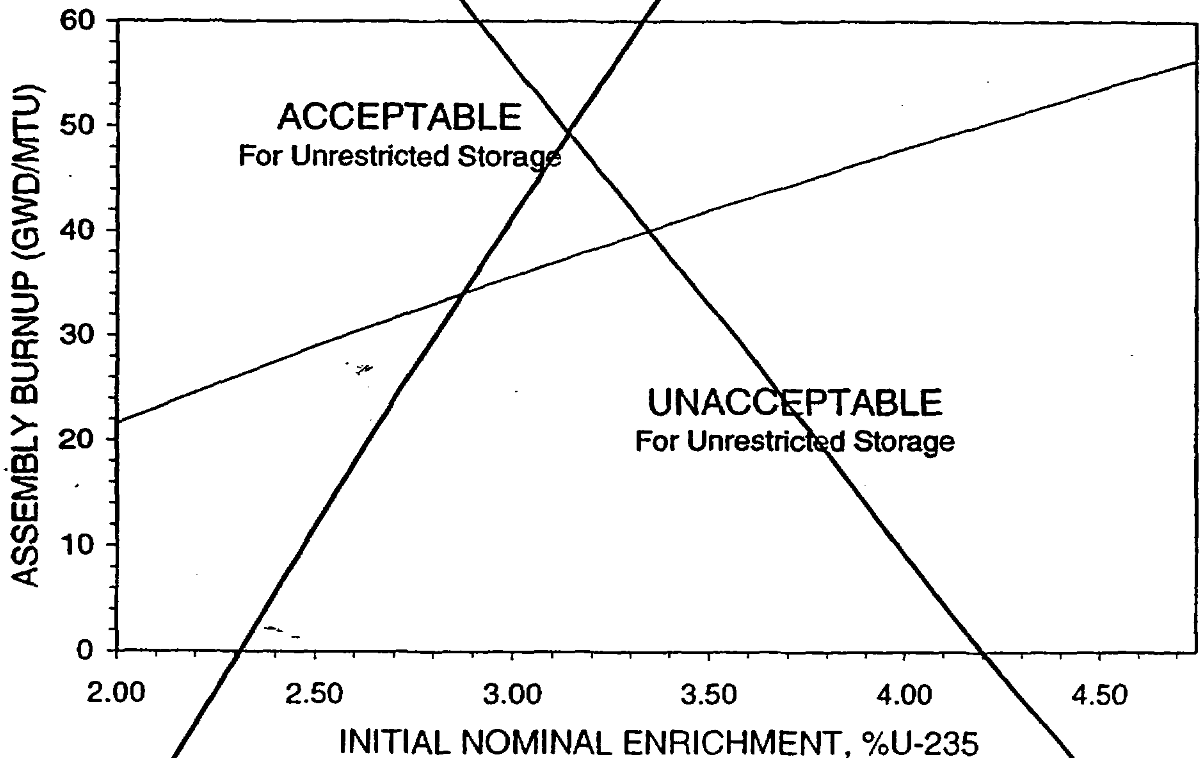
Table 3.7.15-10 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Region 2B Storage

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.

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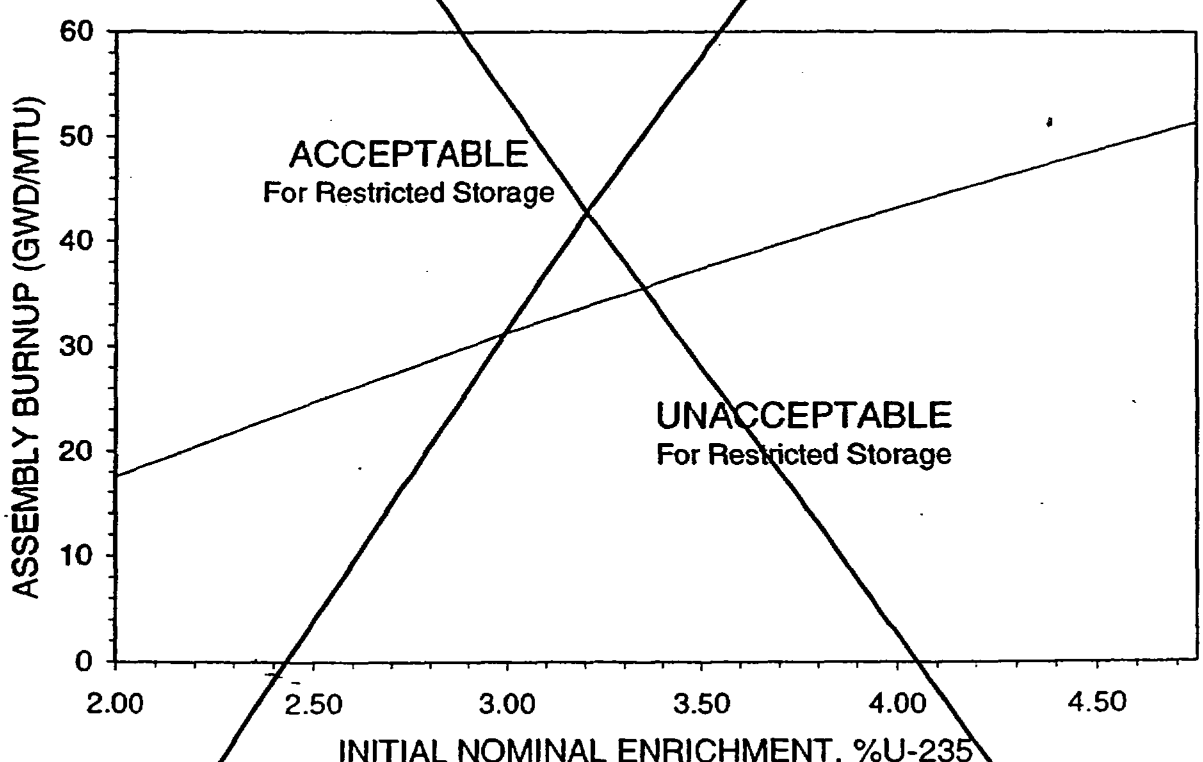
Table 3.7.15-11 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Restricted Region 2B Storage with Fillers

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	51.83



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.

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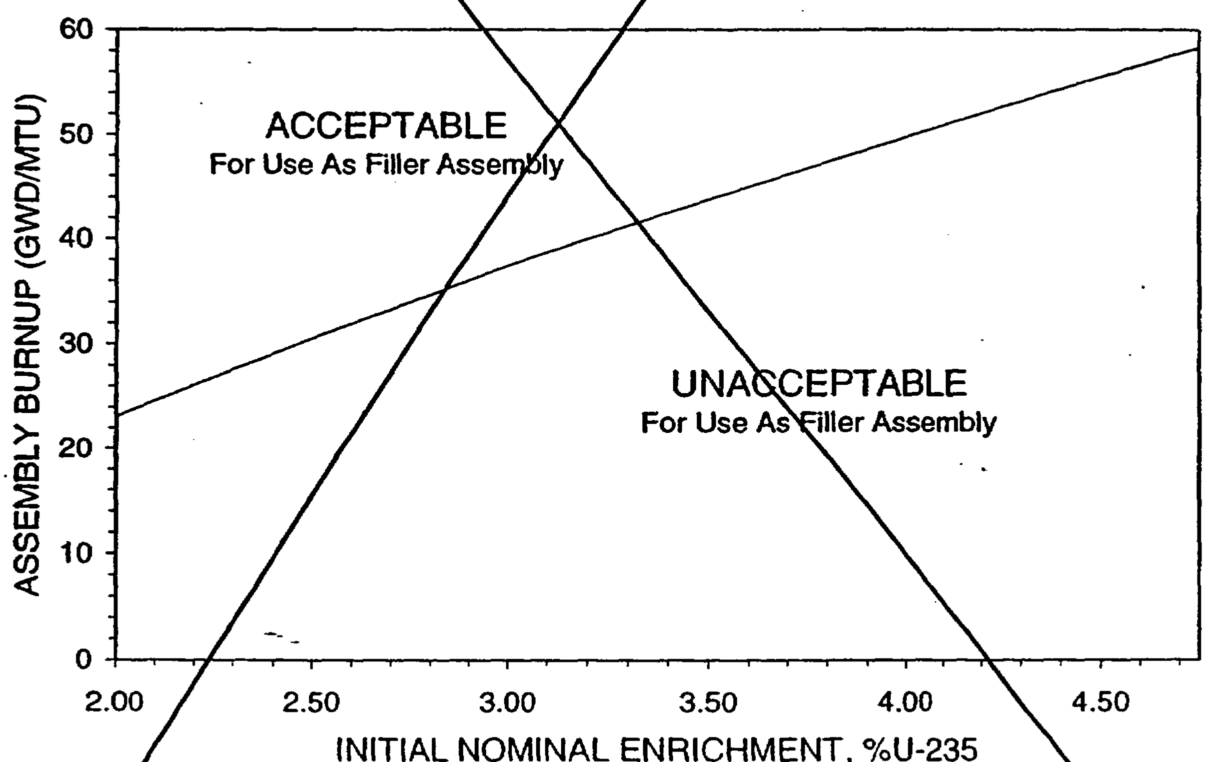
Table 3.7.15-12 (page 1 of 1)
Minimum Qualifying Burnup Versus Initial Enrichment
for Region 2B Filler Assemblies

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.

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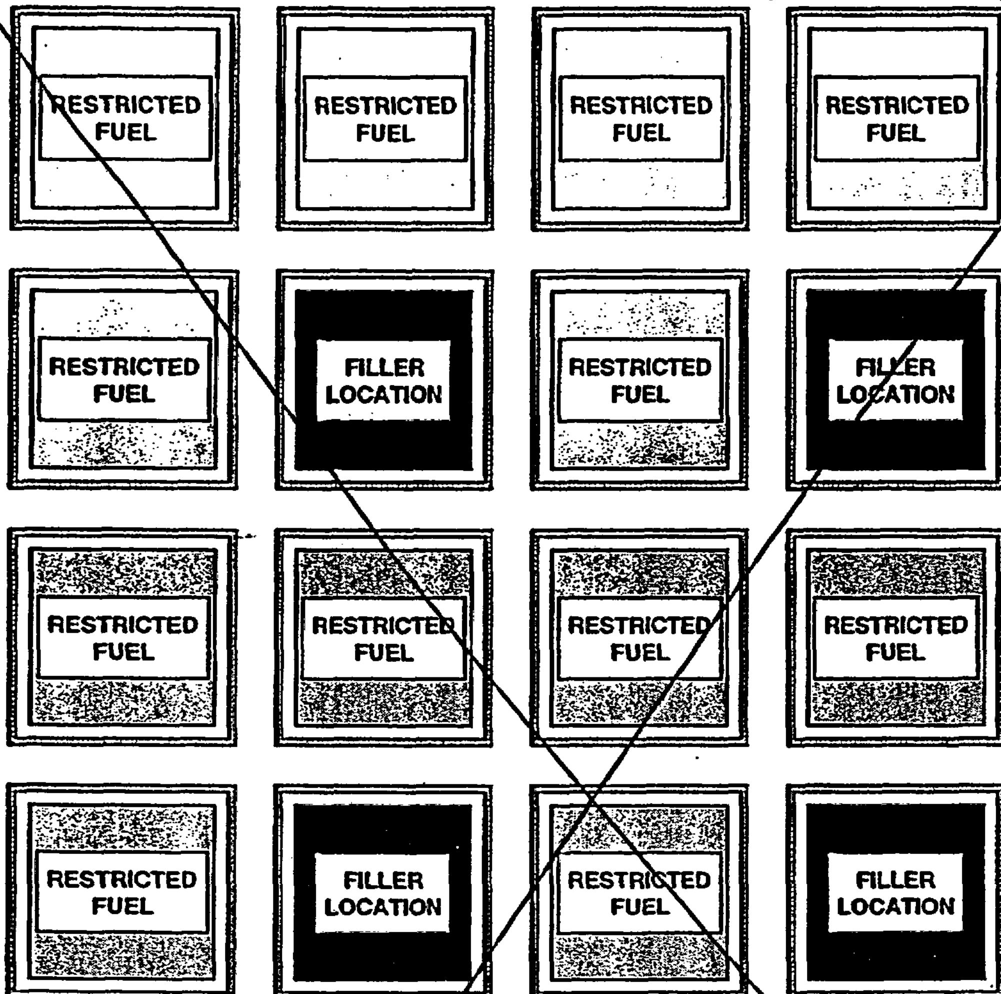

Restricted Fuel:	Fuel which does <u>not</u> meet the minimum burnup requirements of either Table 3.7.15-1 or Table 3.7.15.2. (Fuel which does meet the requirements of Table 3.7.15-1 or Table 3.7.15-2, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed).
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 <u>not</u> 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

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RESTRICTED FUEL	FILLER LOCATION	RESTRICTED FUEL	FILLER LOCATION
FILLER LOCATION	RESTRICTED FUEL	FILLER LOCATION	RESTRICTED FUEL
RESTRICTED FUEL	FILLER LOCATION	RESTRICTED FUEL	FILLER LOCATION
FILLER LOCATION	RESTRICTED FUEL	FILLER LOCATION	RESTRICTED FUEL

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

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

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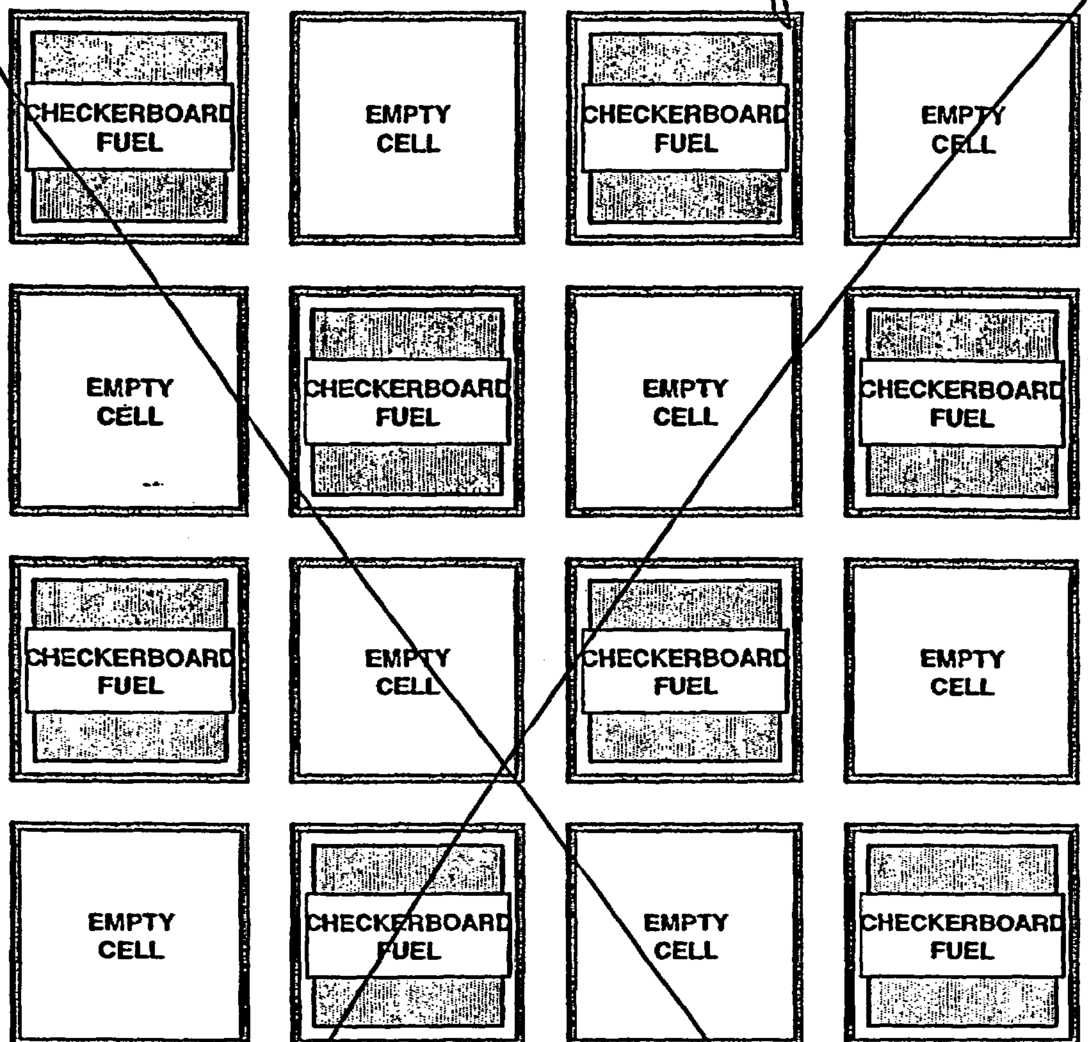

Checkerboard Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.7.15-5. (Fuel which does meet the requirements of Table 3.7.15-5, or non-fuel components, or an empty location may be placed in checkerboard fuel locations as needed)

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.

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

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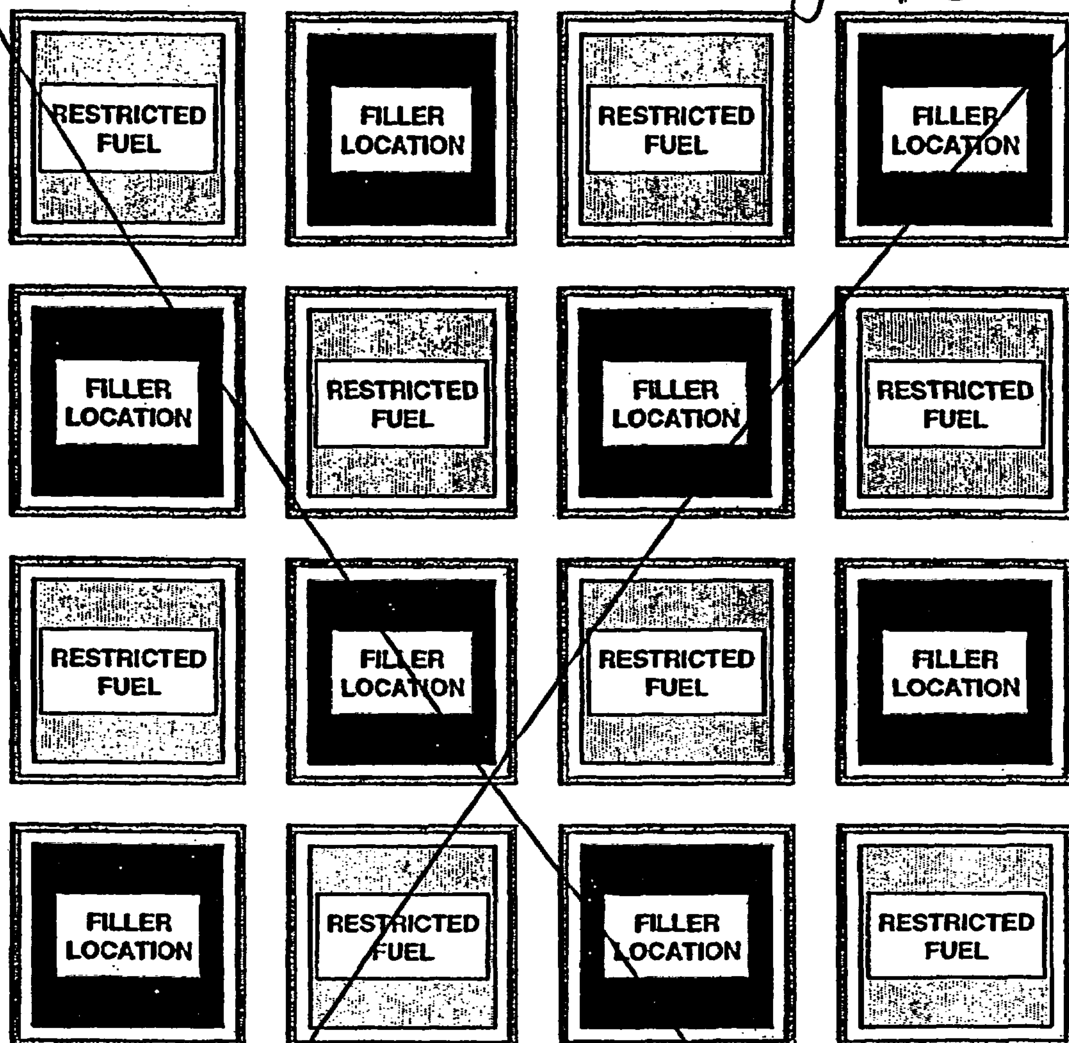

Restricted Fuel: Fuel which meets the minimum burnup requirements of Table 3.7.15-8, or non-fuel components, or an empty location.

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.7.15-9, or an empty cell.

Boundary Condition: At least three of the four faces of each 2A Restricted Fuel Assembly must be adjacent to a 2A Filler Location, an empty cell, or the pool wall, at all boundaries between storage regions.

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

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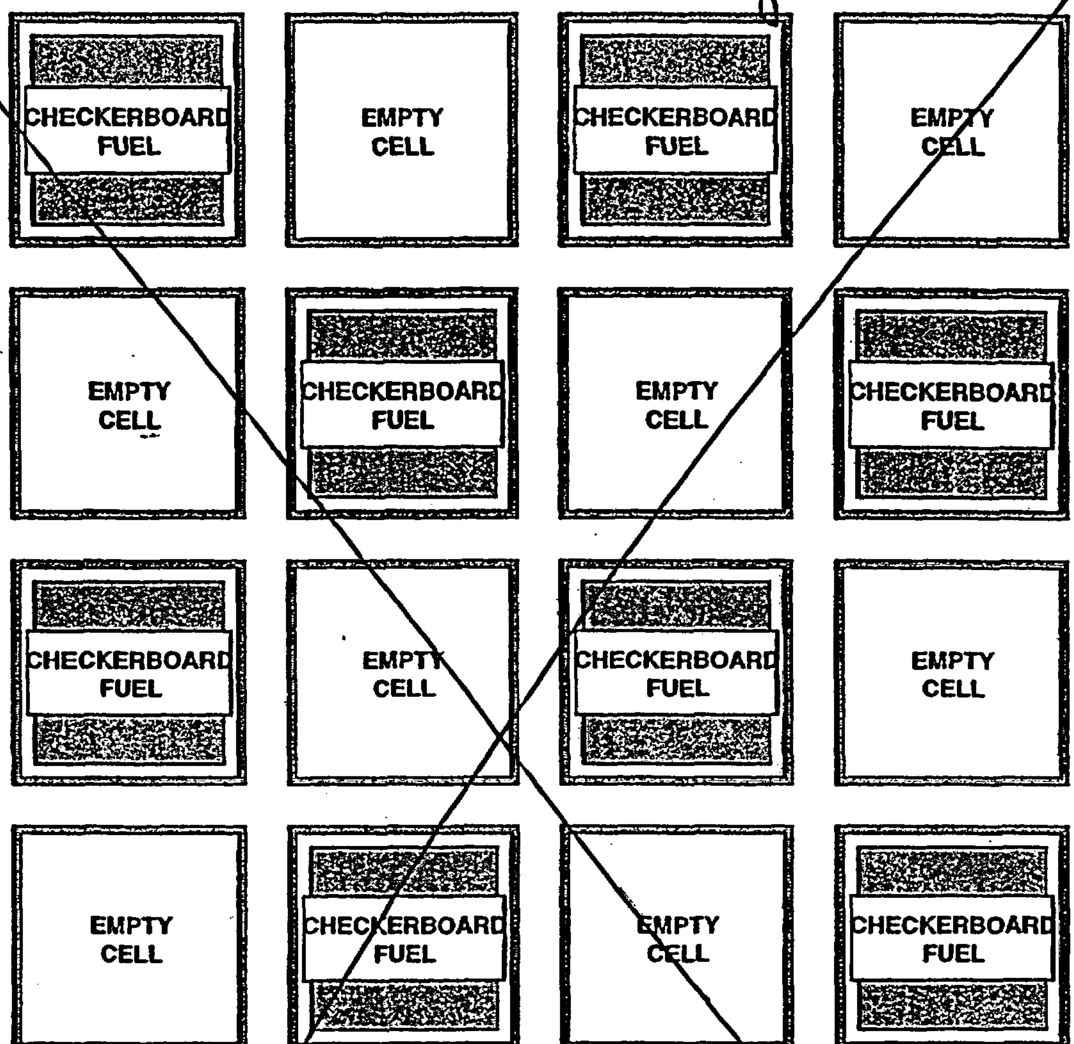

Checkerboard Fuel:	Fuel which does not meet the minimum burnup requirements of Table 3.7.15-8. (Fuel which does meet the requirements of Table 3.7.15-8, or non-fuel components, or an empty location may be placed in checkerboard fuel locations as needed)
Boundary Condition:	At least three of the four faces of each Checkerboard Fuel Assembly must be adjacent to an empty cell or the pool wall, at all boundaries between storage regions.

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

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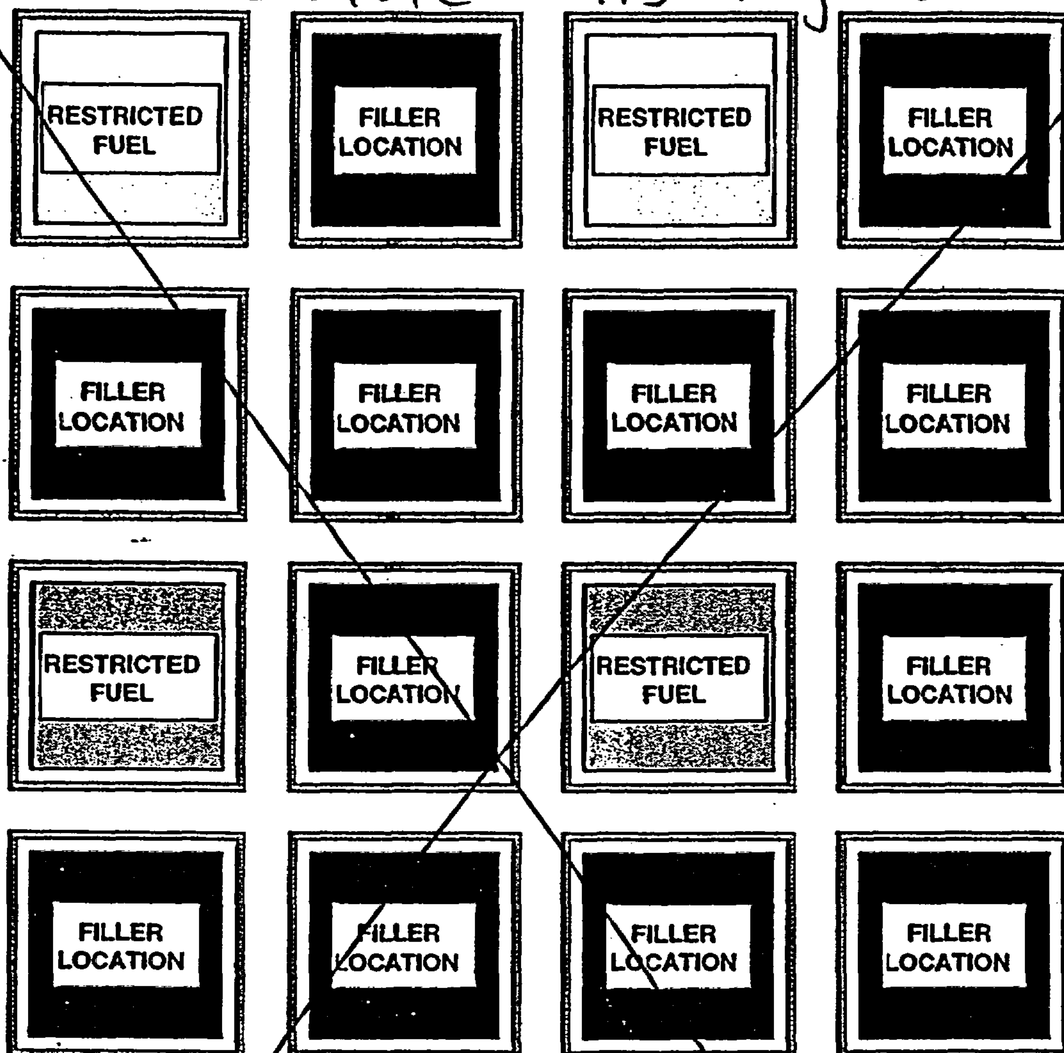

Restricted Fuel: Fuel which meets the minimum burnup requirements of Table 3.7.15-11, or non-fuel components, or an empty location.

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

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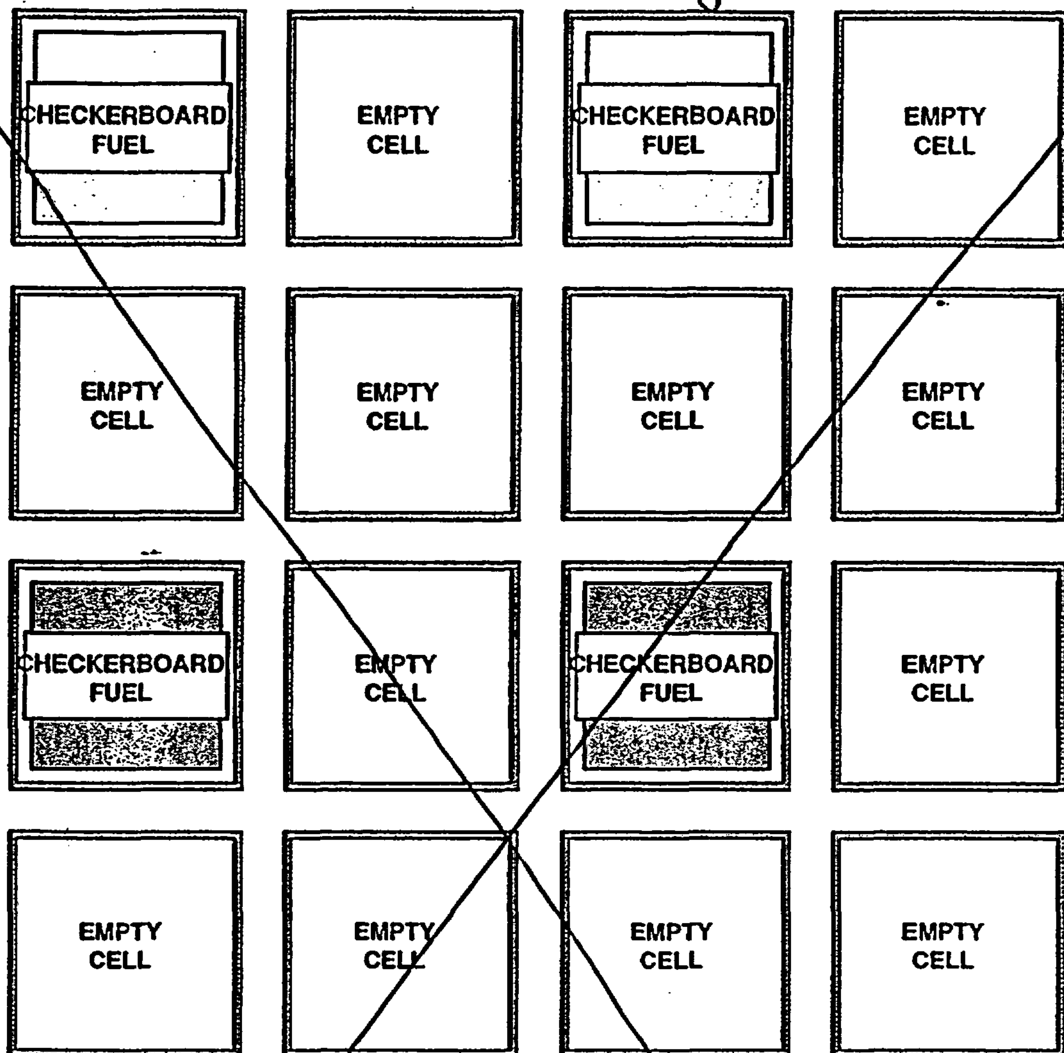


Checkerboard Fuel: Fuel which does <u>not</u> meet the minimum burnup requirements of Table 3.7.15-11. (Fuel which does meet the requirements of Table 3.7.15-11, or non-fuel components, or an empty location may be placed in checkerboard fuel locations as needed)

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 <u>not</u> 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~~ 5.00 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~~ 800 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 1~~A and 1B~~; and

e. A nominal 9.125 inch center to center distance between fuel assemblies placed in Regions 2~~A 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~~ 5.00 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 1~~A and 1B~~ and 1177 total spaces in Regions 2~~A and 2B~~).

ATTACHMENT 2

REVISED MCGUIRE TECHNICAL SPECIFICATIONS

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment, burnup and cooling time for 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 allowed for unrestricted storage in **Region 1** of the spent fuel pool provided the maximum initial U-235 enrichment of the fuel is $\leq$ 5.0 weight percent; or

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

1. Unrestricted storage of fuel meeting the criteria of Table 3.7.15-1; or
2. Restricted storage in accordance with Figure 3.7.15-1 of fuel meeting the criteria of Table 3.7.15-2 (Restricted Fuel assembly) and Table 3.7.15-3 (Filler Fuel assembly); or
3. Checkerboard storage in accordance with Figure 3.7.15-2 of fuel meeting the criteria of Table 3.7.15-4.

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

ACTIONS

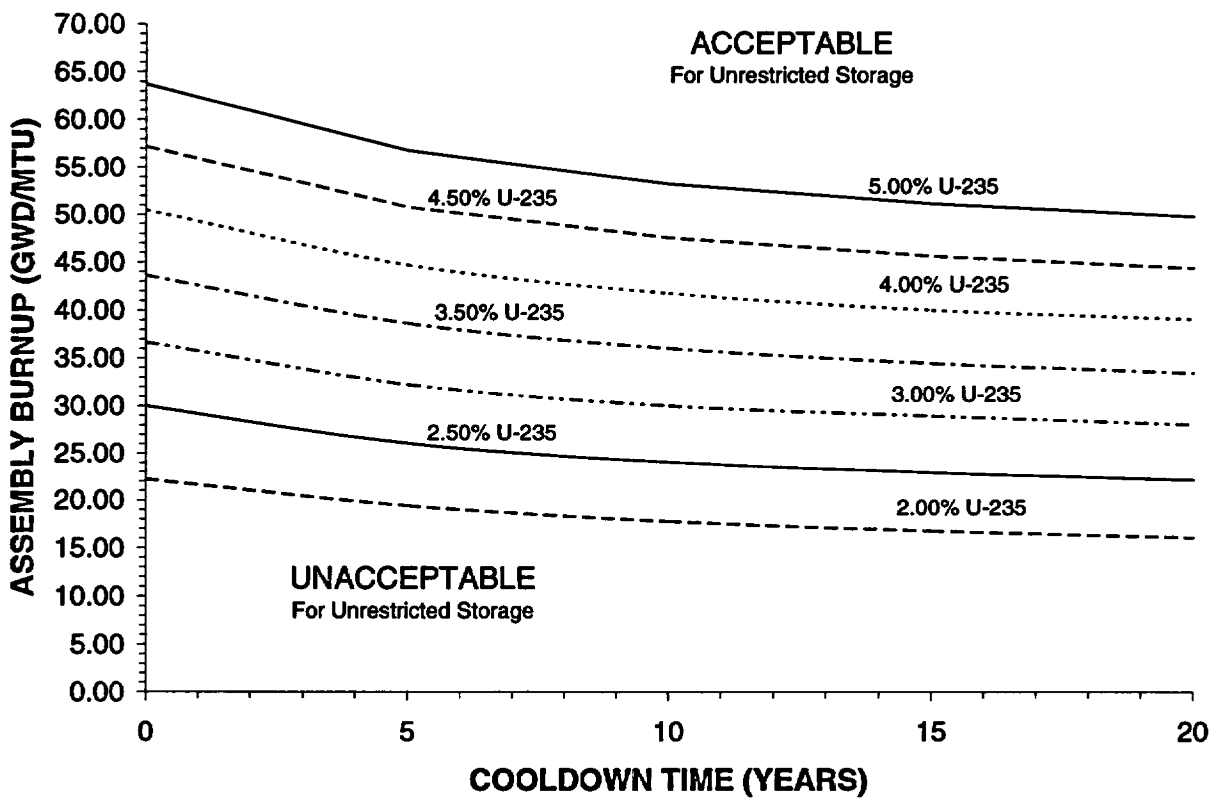
CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	-------------NOTE------------- LCO 3.0.3 is not applicable. ---------------------------------- Initiate action to move the noncomplying fuel assembly to the correct location.	Immediately

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

Table 3.7.15-1 (Page 1 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type MkBW

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	22.20	30.01	36.67	43.61	50.47	57.18	63.72
5	19.42	26.06	32.23	38.64	44.70	50.80	56.77
10	17.76	24.07	30.01	36.02	41.76	47.56	53.24
15	16.74	22.90	28.95	34.45	40.01	45.64	51.15
20	16.07	22.13	28.05	33.44	39.08	44.38	49.78

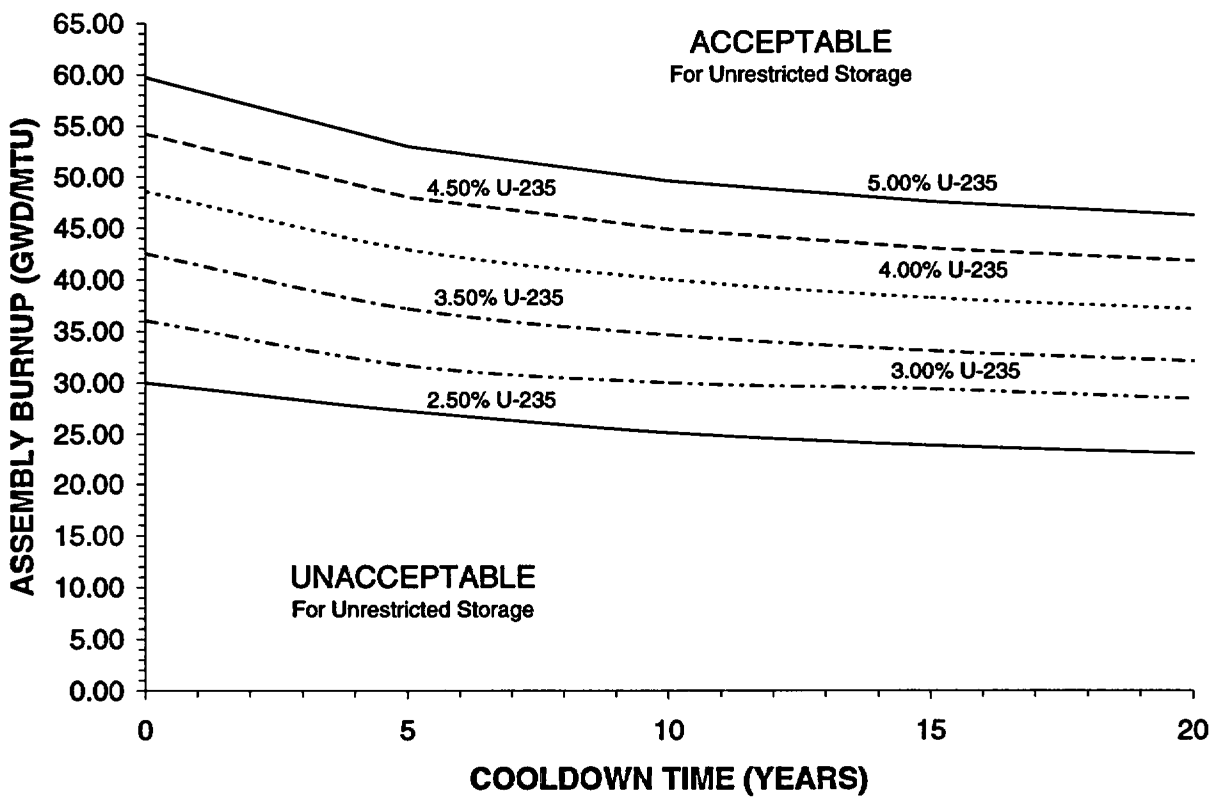


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-1 (Page 2 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type **MkBWb1**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time						
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)					
	2.50	3.00	3.50	4.00	4.50	5.00
0	30.01	36.05	42.52	48.57	54.24	59.74
5	27.27	31.69	37.20	42.92	48.03	53.01
10	25.15	30.01	34.63	40.01	44.85	49.58
15	23.89	29.37	33.09	38.21	43.00	47.57
20	23.09	28.43	32.09	37.13	41.78	46.26

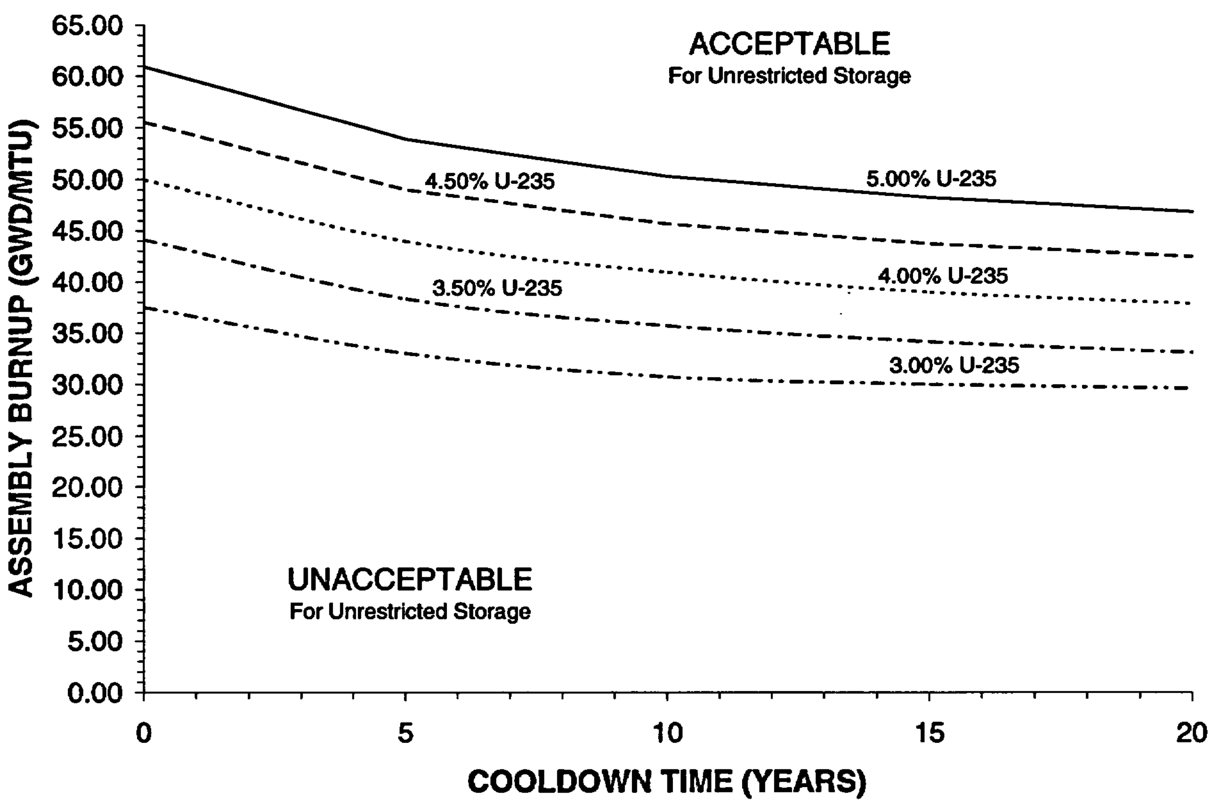


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-1 (Page 3 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type **MkBWb2**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	37.51	44.11	49.95	55.50	60.90
5	33.02	38.34	43.97	48.98	53.87
10	30.72	35.73	40.95	45.67	50.30
15	30.01	34.15	38.99	43.72	48.22
20	29.62	33.12	37.88	42.47	46.85

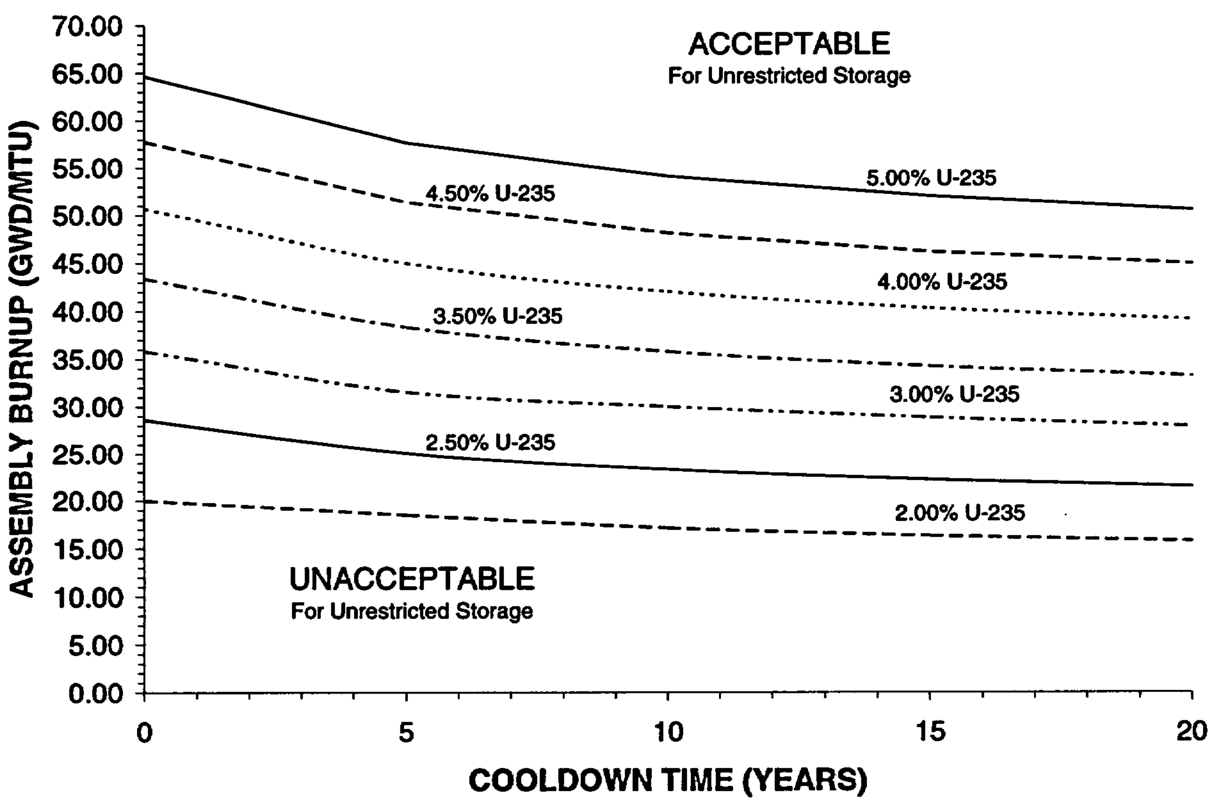


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-1 (Page 4 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type **W-STD**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	20.02	28.59	35.83	43.37	50.67	57.75	64.63
5	18.50	25.10	31.56	38.35	44.97	51.39	57.66
10	17.14	23.29	30.01	35.78	42.03	48.12	54.08
15	16.32	22.21	28.83	34.24	40.29	46.19	51.97
20	15.79	21.51	27.96	33.24	39.16	44.94	50.61

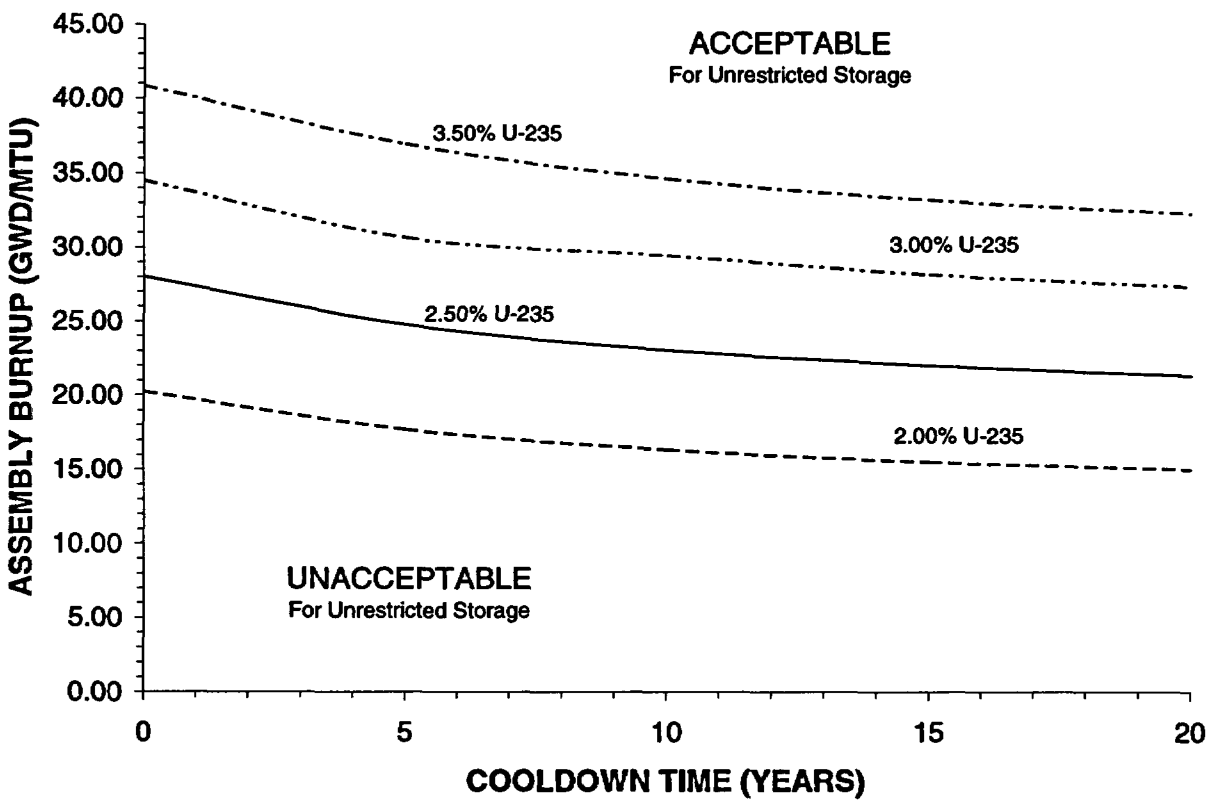


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-1 (Page 5 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type **MkBI**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time				
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)			
	2.00	2.50	3.00	3.50
0	20.21	28.01	34.47	40.82
5	17.71	24.76	30.66	36.92
10	16.35	23.04	29.42	34.60
15	15.53	22.01	28.16	33.19
20	15.00	21.33	27.34	32.27

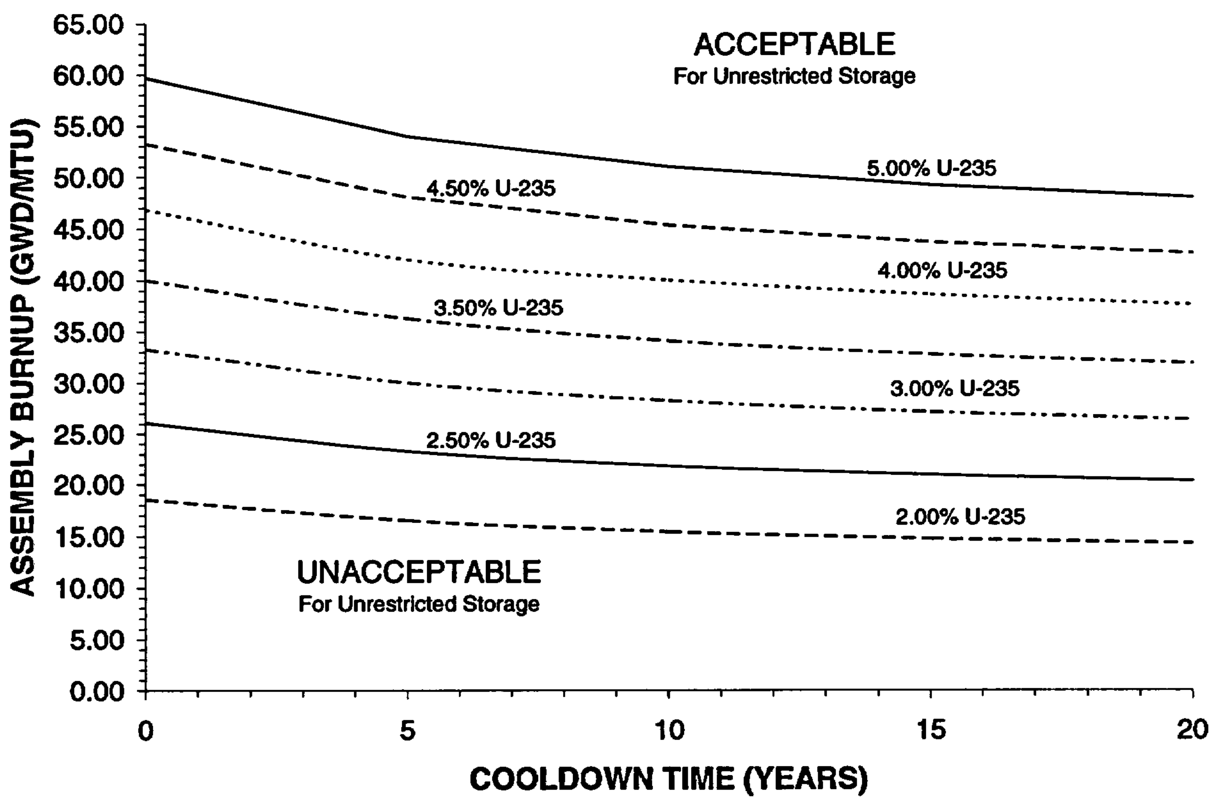


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-1 (Page 6 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type **W-OFA**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	18.55	26.08	33.28	40.01	46.83	53.25	59.71
5	16.53	23.30	30.01	36.27	42.01	48.05	53.98
10	15.43	21.83	28.25	34.10	40.01	45.34	50.99
15	14.75	20.92	27.12	32.78	38.60	43.68	49.19
20	14.32	20.33	26.40	31.91	37.62	42.62	48.02

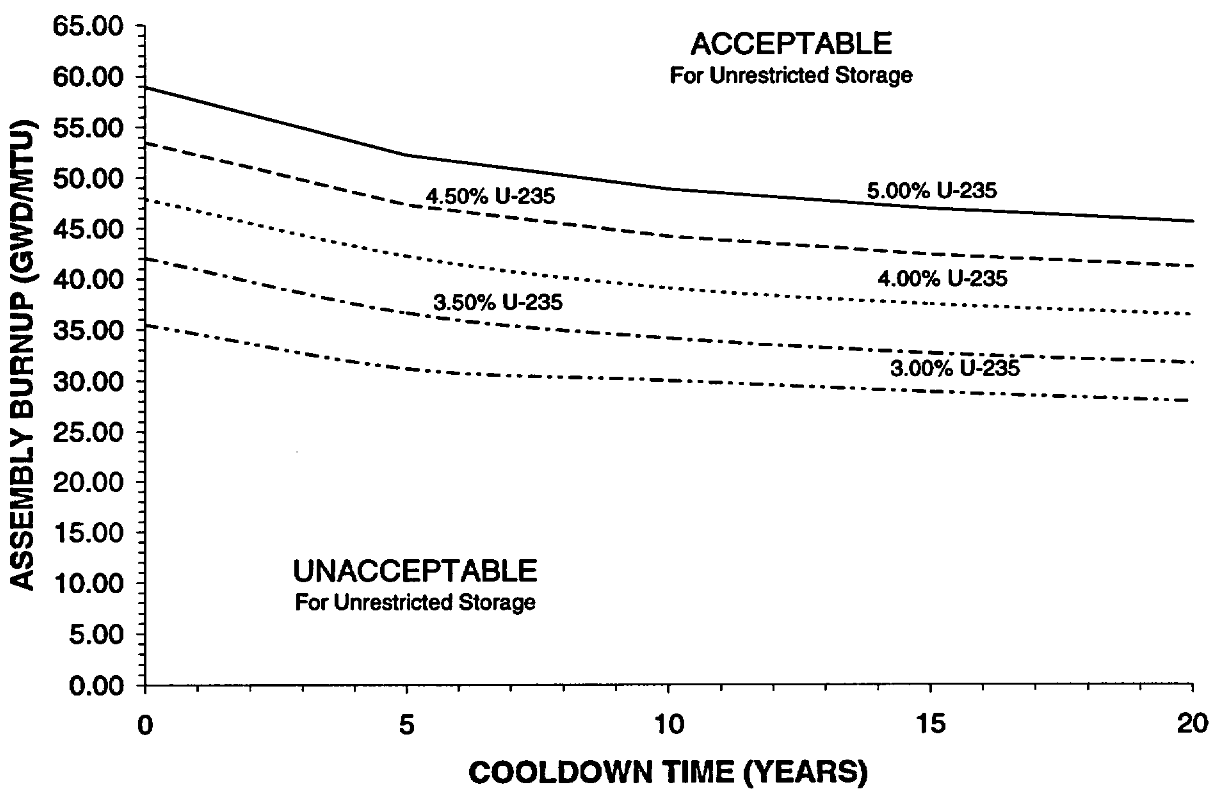


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-1 (Page 7 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Unrestricted Region 2 Storage
For Fuel Assembly Type W-RFA

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	35.46	42.04	47.88	53.50	58.94
5	31.19	36.62	42.23	47.30	52.23
10	30.01	34.11	39.05	44.15	48.85
15	28.85	32.63	37.41	42.31	46.87
20	27.93	31.67	36.35	41.12	45.57

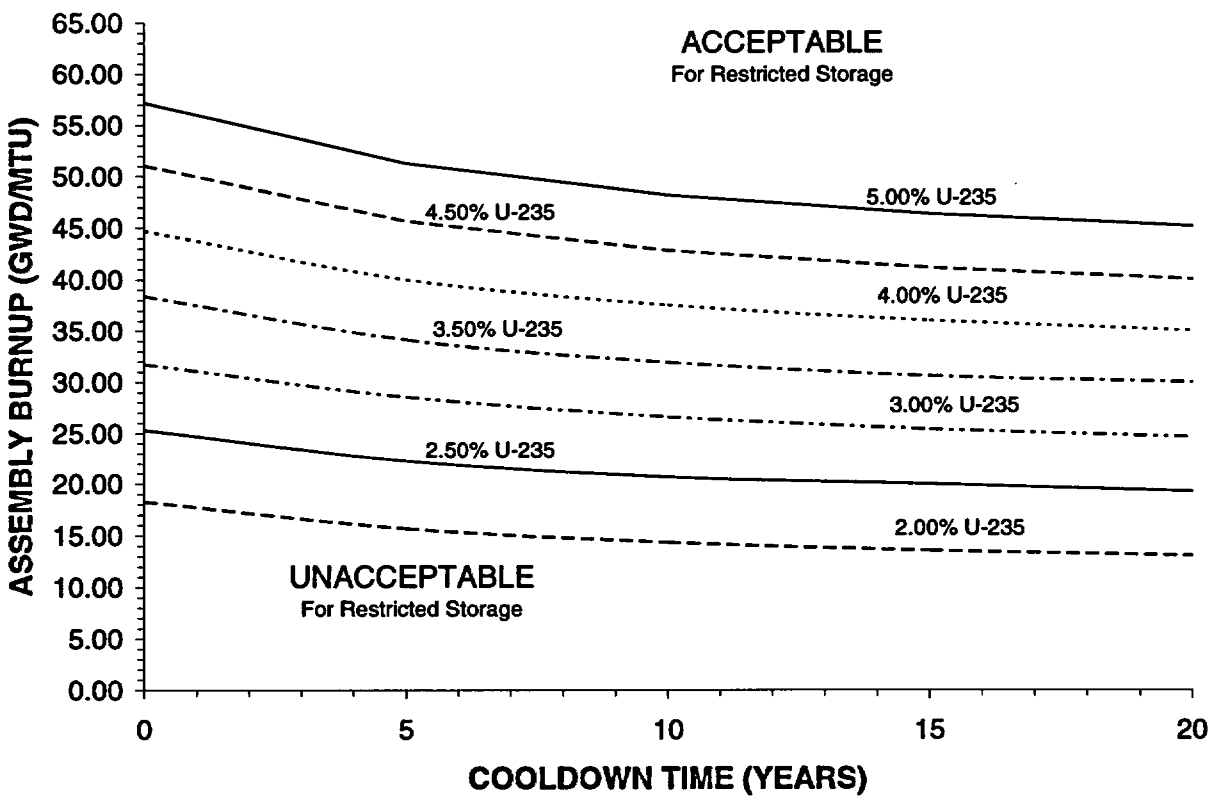


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-1 may be qualified for use as a Region 2 Unrestricted 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 3.7.15-2 (Page 1 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type MkBW

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	18.26	25.32	31.73	38.39	44.73	51.04	57.20
5	15.69	22.29	28.56	34.16	40.01	45.66	51.27
10	14.36	20.68	26.60	31.95	37.54	42.83	48.19
15	13.59	20.01	25.42	30.62	36.04	41.16	46.36
20	13.10	19.29	24.66	30.01	35.05	40.07	45.16

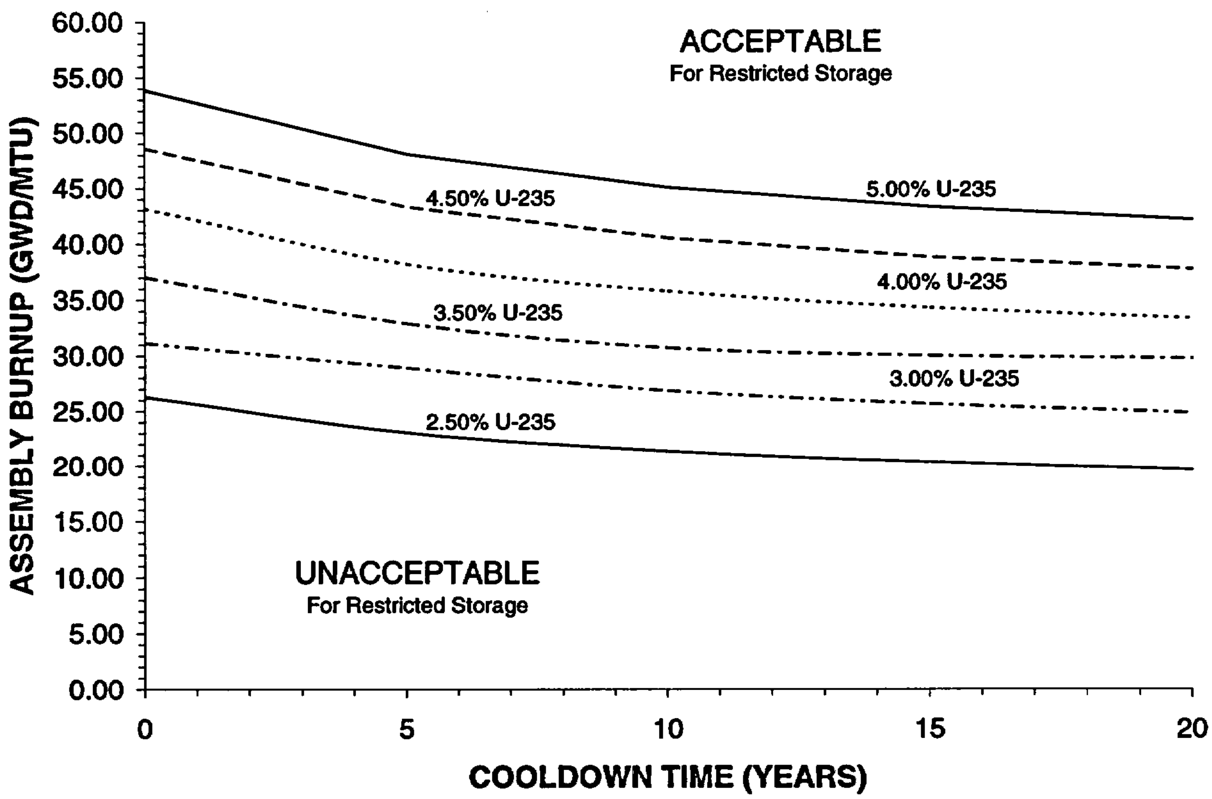


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-2 (Page 2 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type MkBWb1

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time						
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)					
	2.50	3.00	3.50	4.00	4.50	5.00
0	26.29	31.14	37.02	43.12	48.55	53.83
5	23.07	28.91	32.89	38.20	43.29	48.07
10	21.37	26.88	30.73	35.78	40.54	45.07
15	20.35	25.66	30.01	34.30	38.82	43.31
20	19.66	24.86	29.76	33.36	37.77	42.16

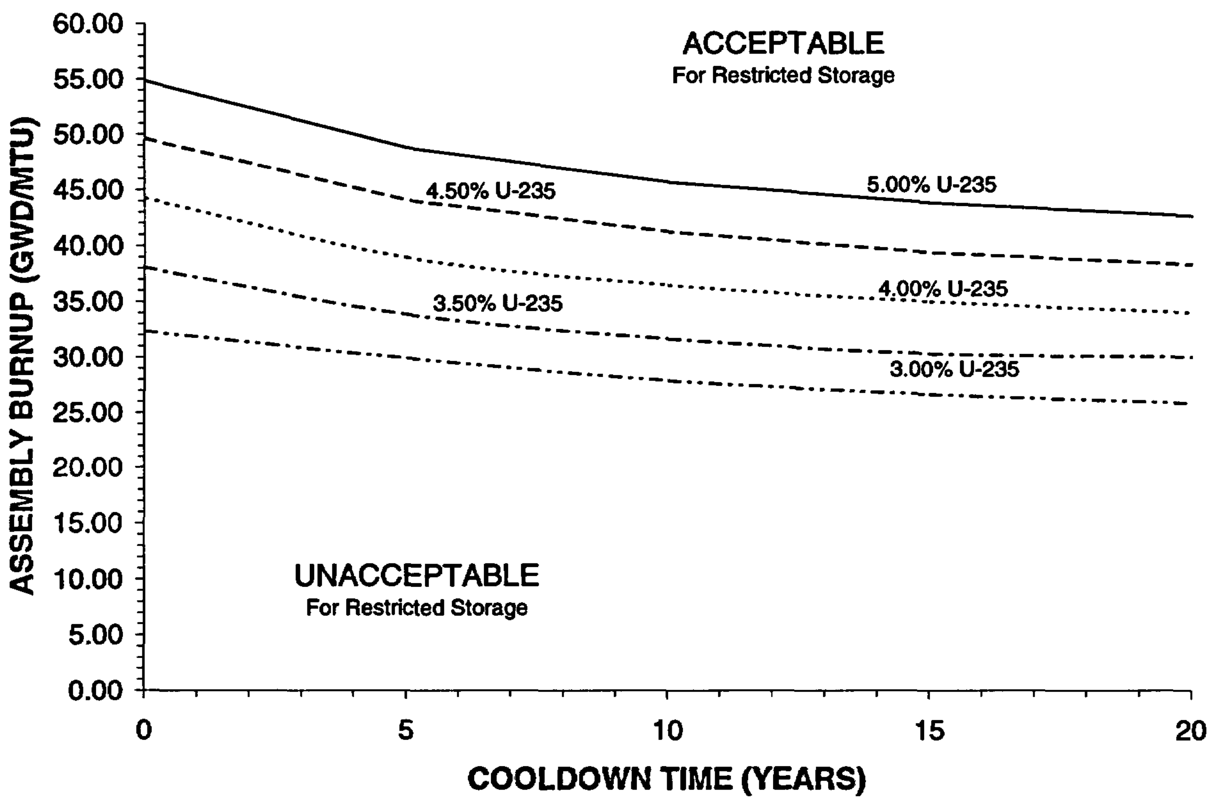


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-2 (Page 3 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type MkBWb2

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	32.33	38.06	44.25	49.61	54.82
5	29.93	33.86	38.94	44.09	48.80
10	27.89	31.65	36.48	41.24	45.69
15	26.66	30.32	34.99	39.40	43.85
20	25.85	30.01	34.01	38.34	42.67

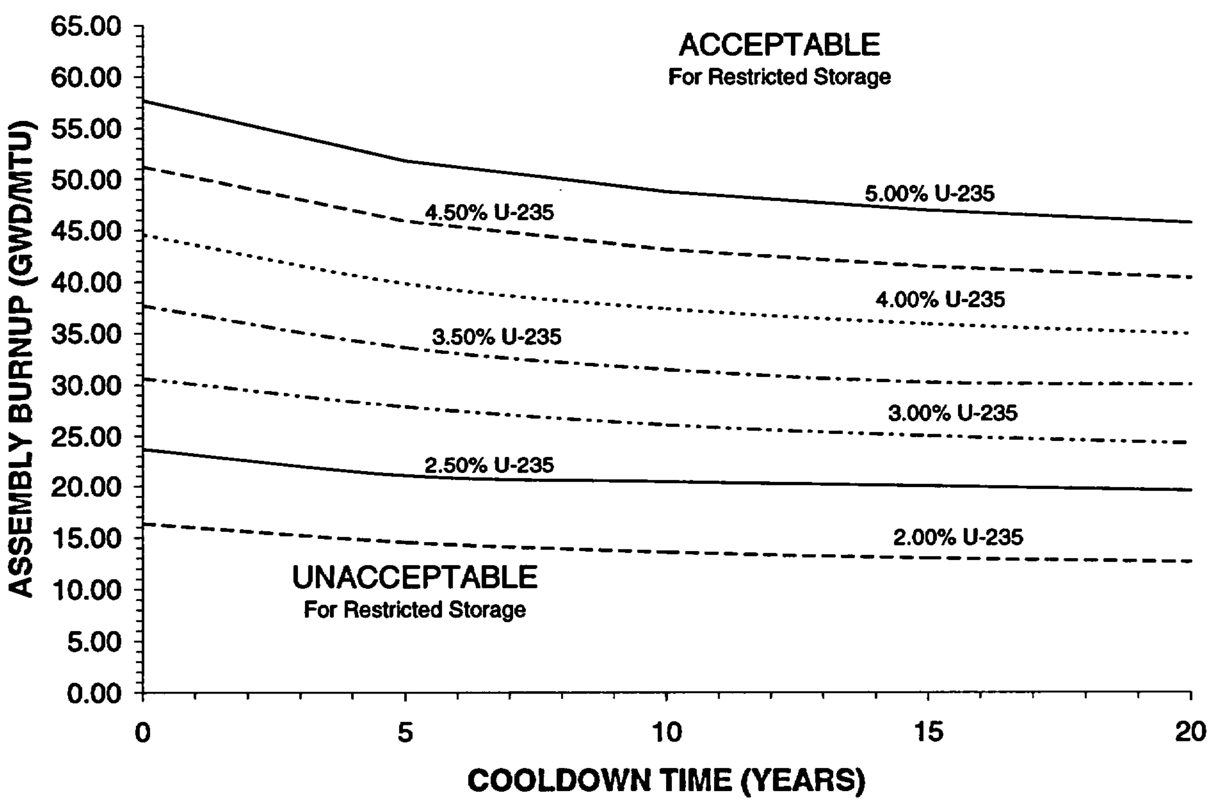


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-2 (Page 4 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type **W-STD**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	16.34	23.70	30.62	37.69	44.55	51.21	57.70
5	14.55	21.04	27.88	33.62	39.84	45.90	51.80
10	13.58	20.42	26.08	31.49	37.37	43.11	48.73
15	12.99	20.01	24.99	30.21	35.89	41.45	46.90
20	12.63	19.56	24.28	30.01	34.93	40.37	45.70

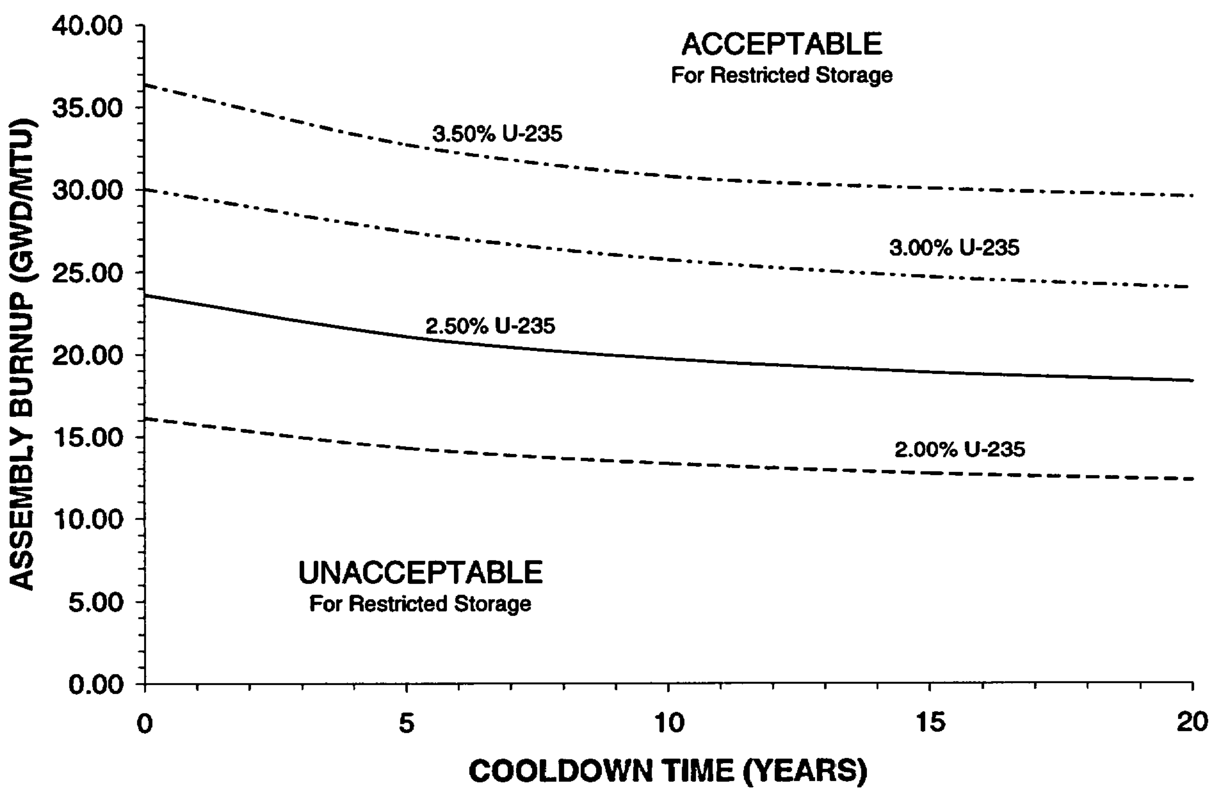


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-2 (Page 5 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type MkBI

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time				
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)			
	2.00	2.50	3.00	3.50
0	16.13	23.62	30.01	36.37
5	14.26	21.08	27.43	32.71
10	13.27	19.71	25.72	30.74
15	12.67	18.87	24.67	30.01
20	12.30	18.33	23.99	29.52

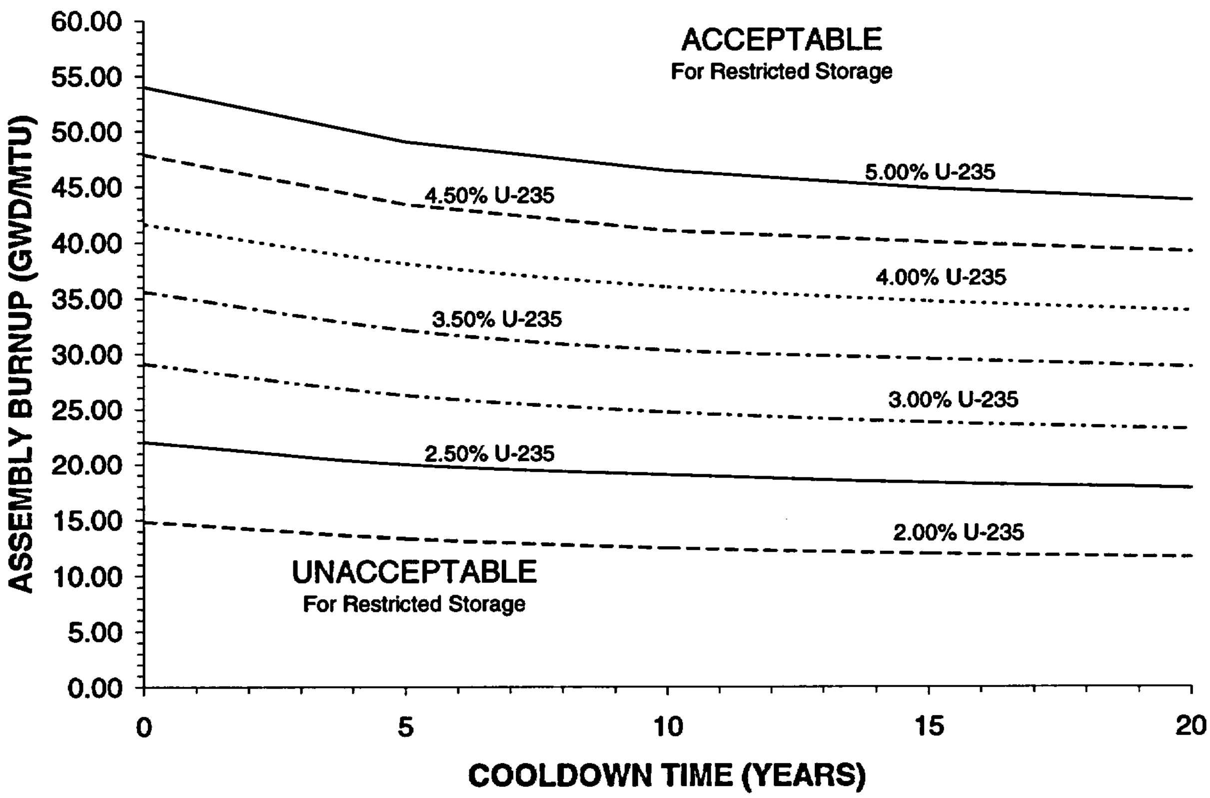


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-2 (Page 6 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type W-OFA

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	14.85	22.04	29.10	35.62	41.63	47.88	54.01
5	13.38	20.01	26.26	32.15	38.13	43.42	49.04
10	12.53	19.03	24.74	30.33	36.01	41.04	46.41
15	12.00	18.29	23.81	29.54	34.72	40.01	44.82
20	11.67	17.82	23.20	28.80	33.87	39.17	43.77

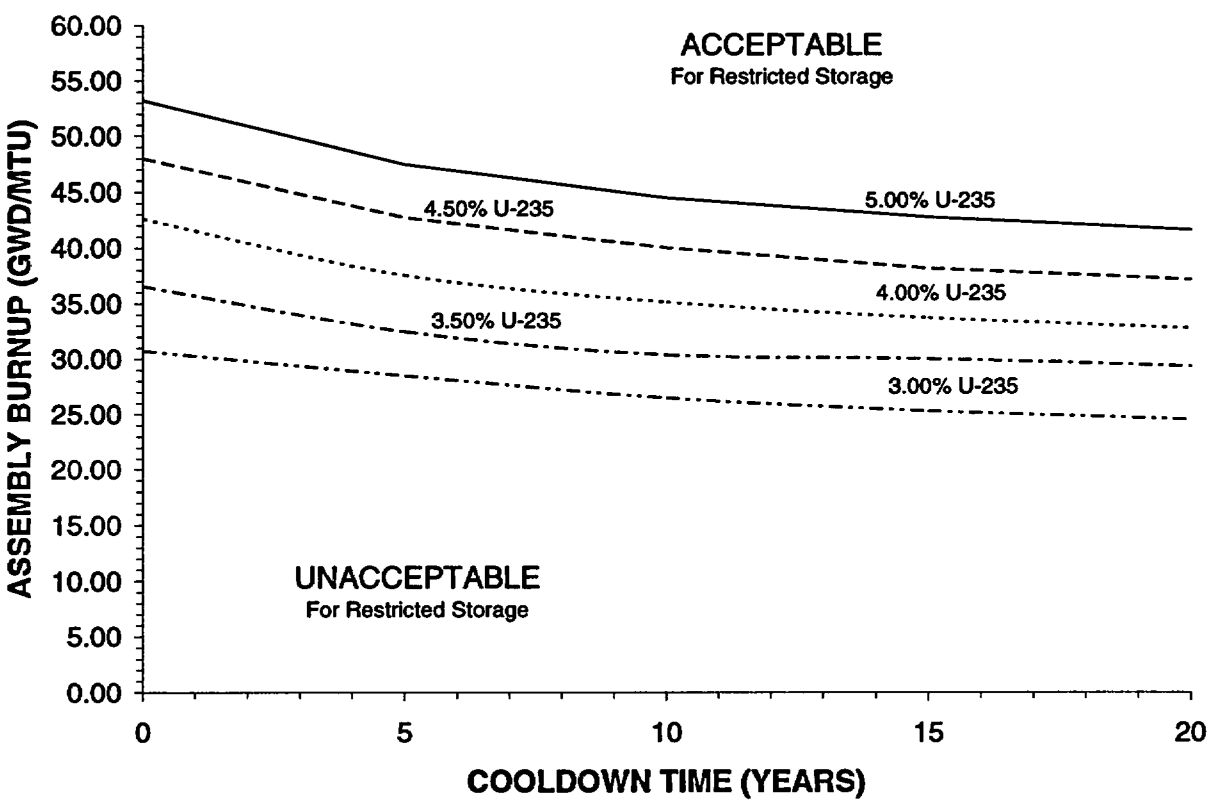


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-2 (Page 7 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted Region 2 Storage
For Fuel Assembly Type **W-RFA**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	30.73	36.55	42.59	47.99	53.22
5	28.49	32.49	37.54	42.73	47.47
10	26.49	30.38	35.15	40.01	44.50
15	25.30	30.01	33.71	38.18	42.75
20	24.53	29.33	32.78	37.16	41.61

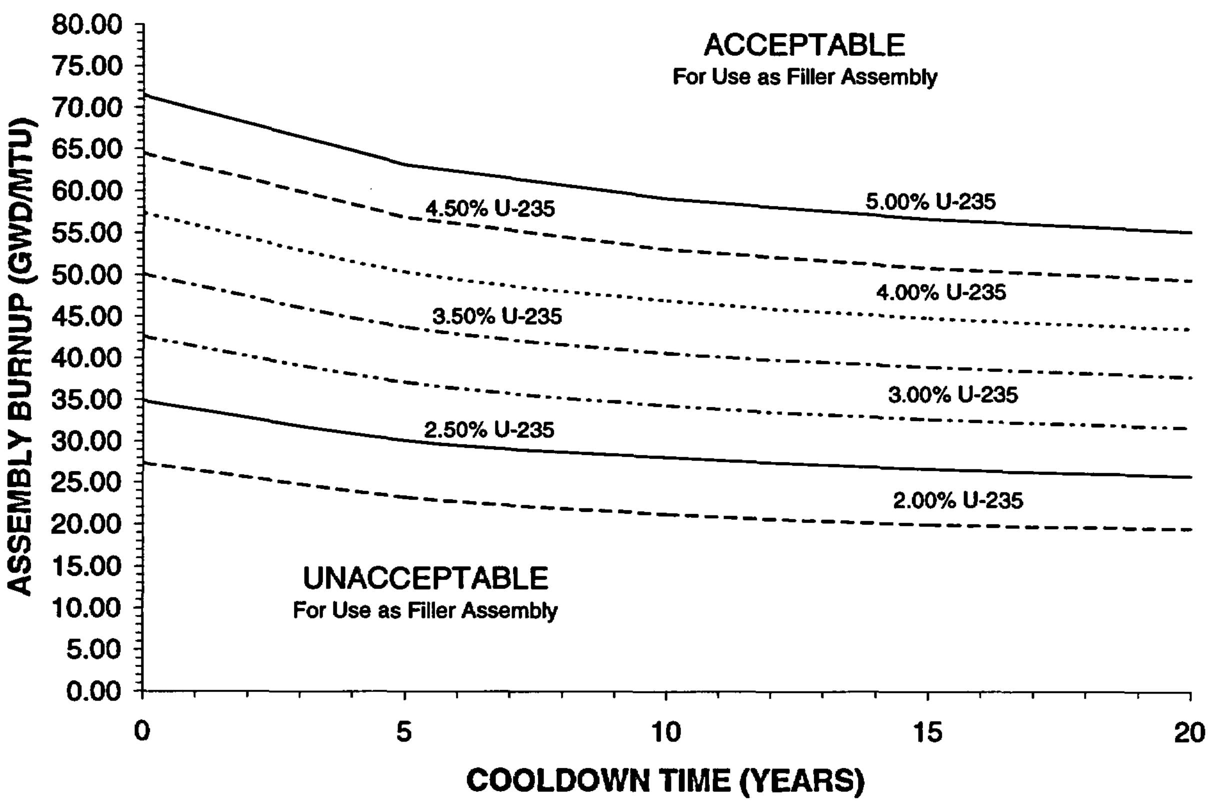


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-2 may be qualified for use as a Region 2 Restricted 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 3.7.15-3 (Page 1 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type MkBW

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	27.34	34.90	42.58	50.08	57.40	64.52	71.46
5	23.28	30.12	37.14	43.78	50.40	56.89	63.22
10	21.24	28.12	34.35	40.65	46.94	53.10	59.12
15	20.02	26.67	32.70	38.98	44.88	50.86	56.72
20	19.50	25.73	31.65	37.77	43.55	49.42	55.18

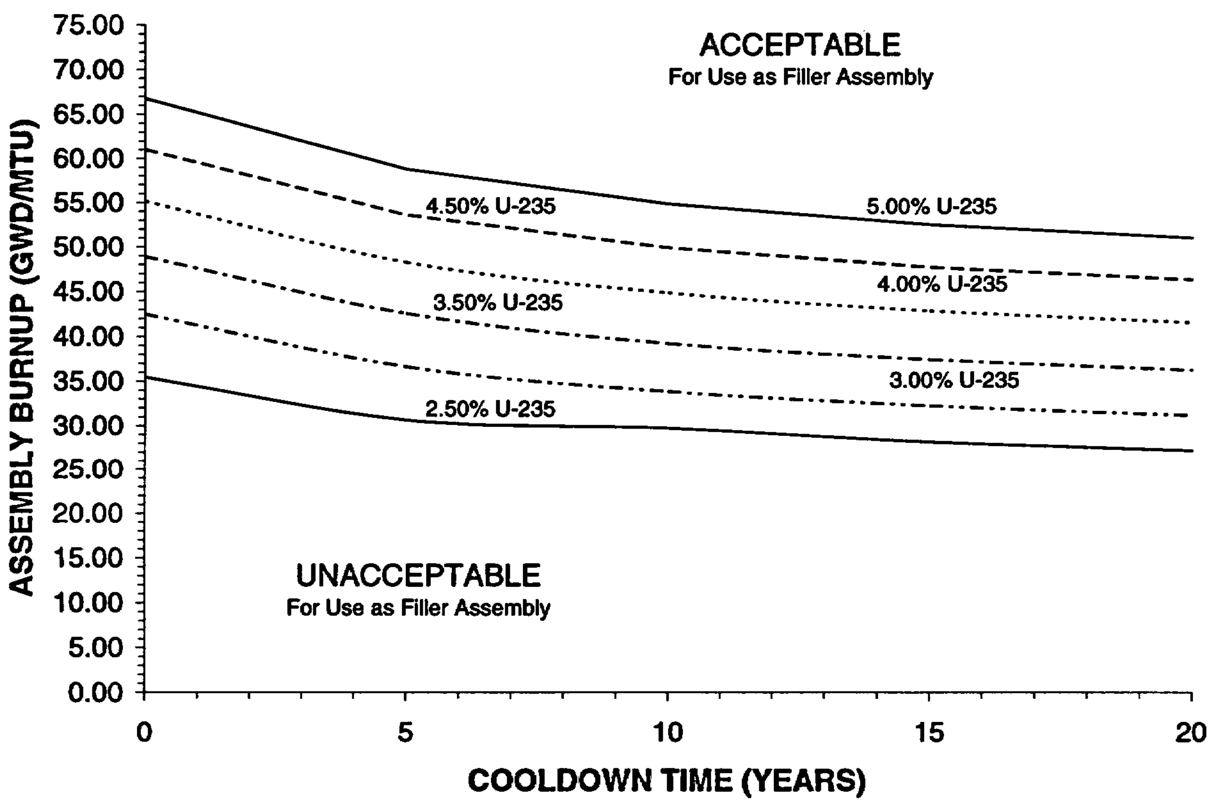


NOTES:

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 2 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 3.7.15-3 (Page 2 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type MkBWb1

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time						
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)					
	2.50	3.00	3.50	4.00	4.50	5.00
0	35.45	42.48	48.89	55.13	61.02	66.75
5	30.69	36.65	42.58	48.29	53.61	58.82
10	29.76	33.85	39.24	44.86	49.90	54.84
15	28.18	32.24	37.45	42.88	47.75	52.53
20	27.20	31.19	36.27	41.58	46.35	51.02

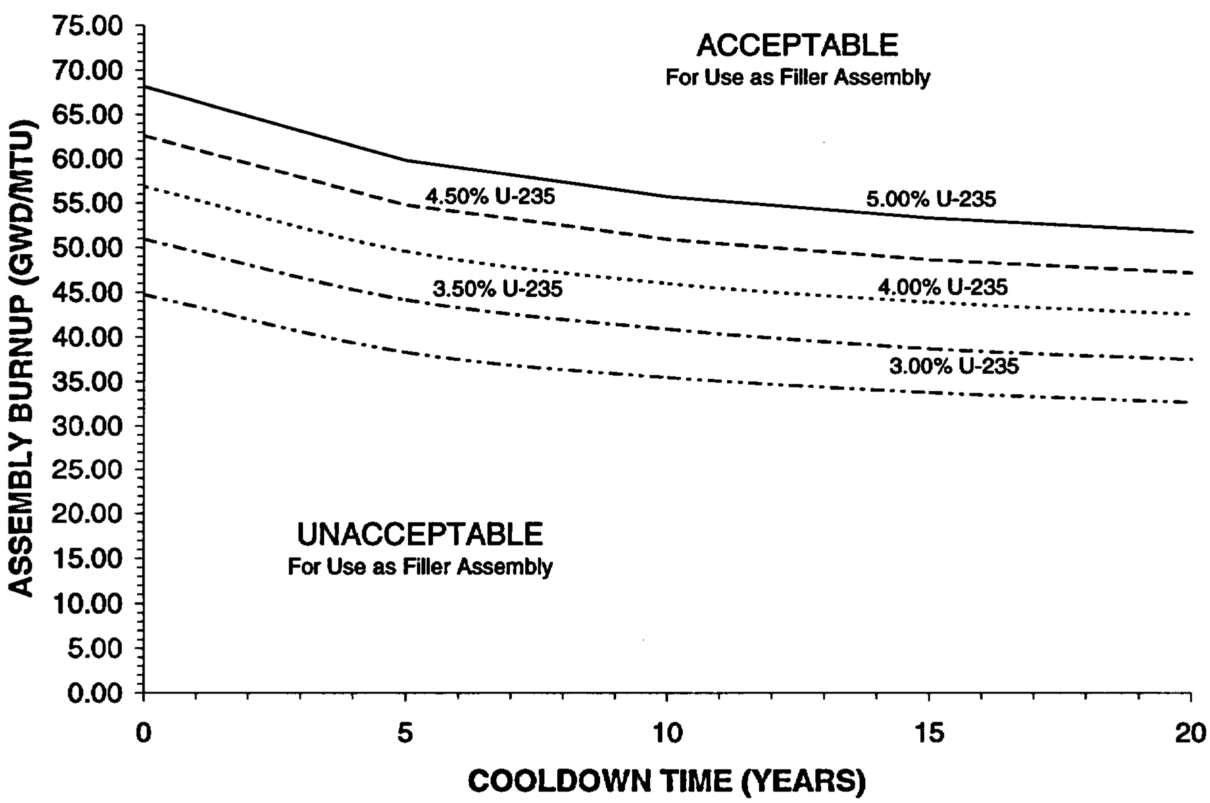


NOTES:

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 2 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 3.7.15-3 (Page 3 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type MkBWb2

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	44.74	50.89	56.84	62.57	68.14
5	38.31	44.17	49.58	54.76	59.81
10	35.48	40.89	45.99	50.89	55.68
15	33.78	38.71	43.90	48.63	53.27
20	32.70	37.52	42.56	47.17	51.72

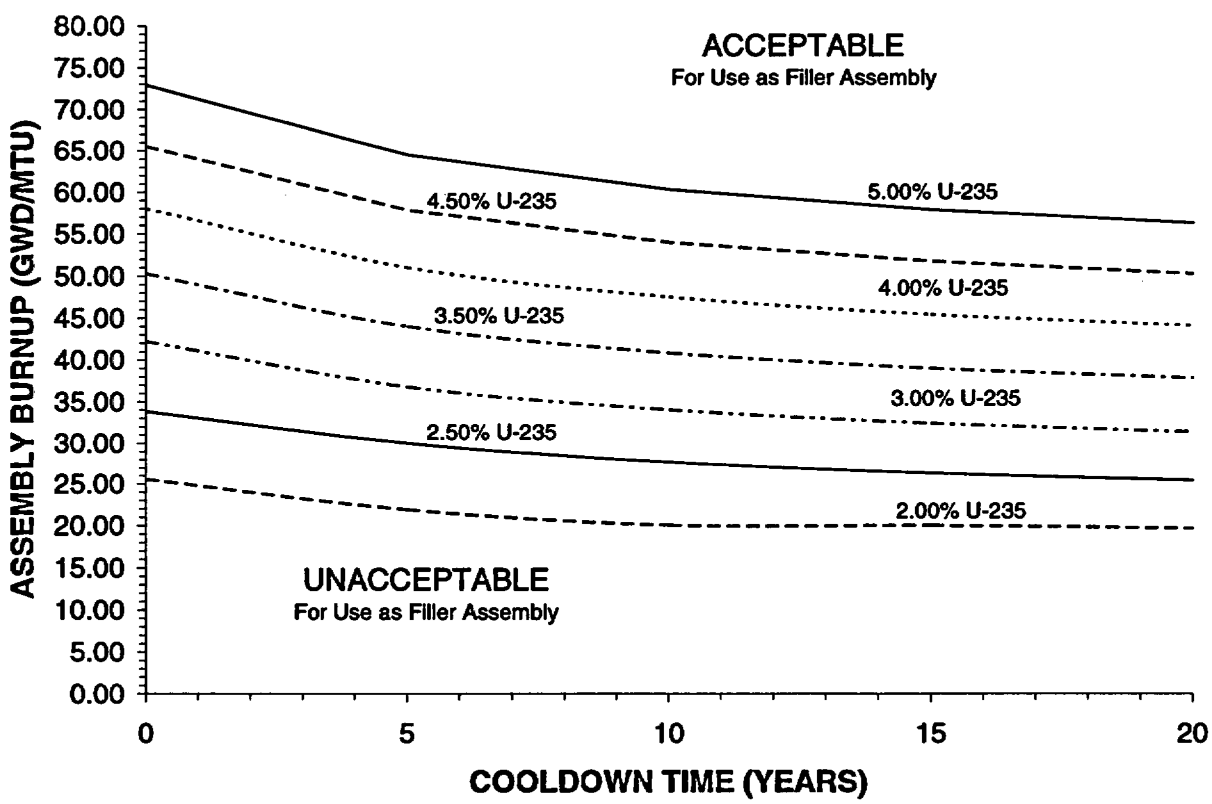


NOTES:

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 2 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 3.7.15-3 (Page 4 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type **W-STD**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	25.55	33.83	42.22	50.27	58.03	65.54	72.89
5	21.90	30.01	36.78	44.01	51.04	57.87	64.53
10	20.06	27.68	34.03	40.87	47.52	54.01	60.34
15	20.01	26.32	32.41	39.02	45.46	51.75	57.91
20	19.68	25.44	31.39	37.84	44.13	50.29	56.33

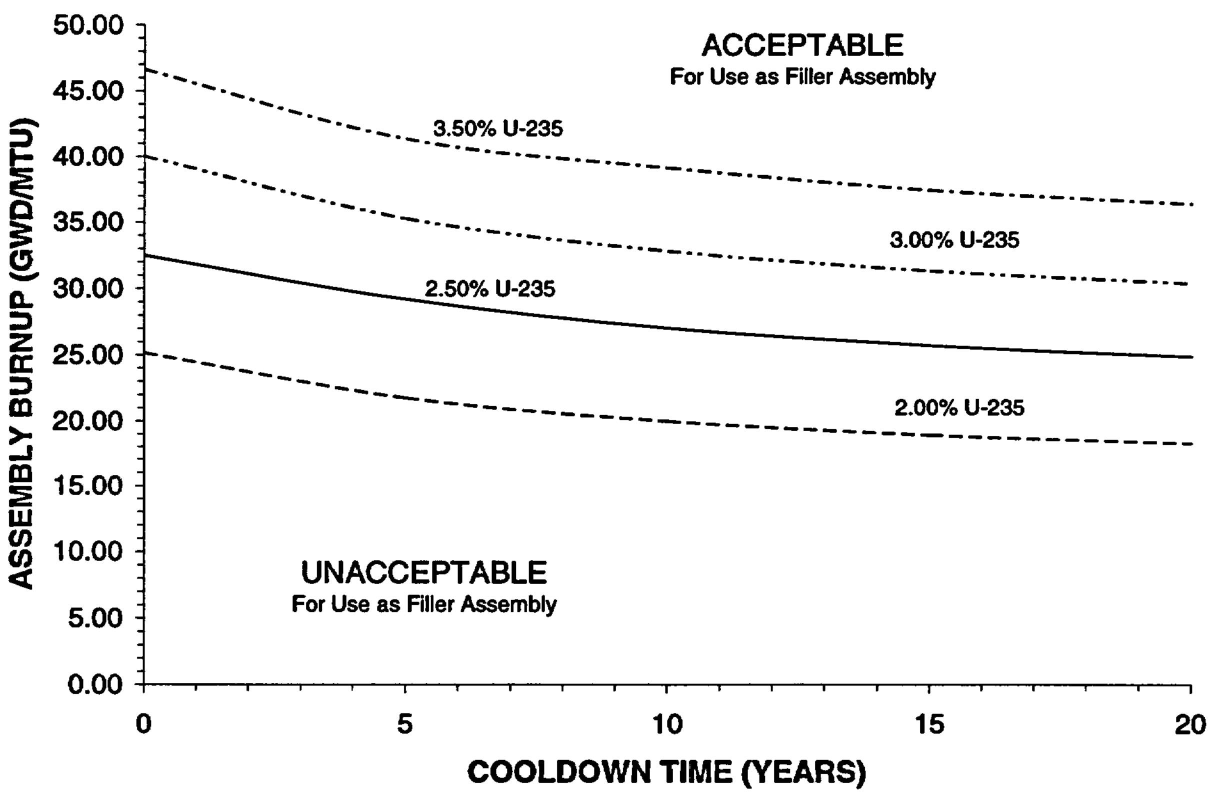


NOTES:

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 2 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 3.7.15-3 (Page 5 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type MkBI

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time				
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)			
	2.00	2.50	3.00	3.50
0	25.14	32.48	40.01	46.65
5	21.76	29.20	35.28	41.37
10	19.98	27.03	32.82	39.13
15	18.94	25.73	31.34	37.45
20	18.27	24.89	30.40	36.39

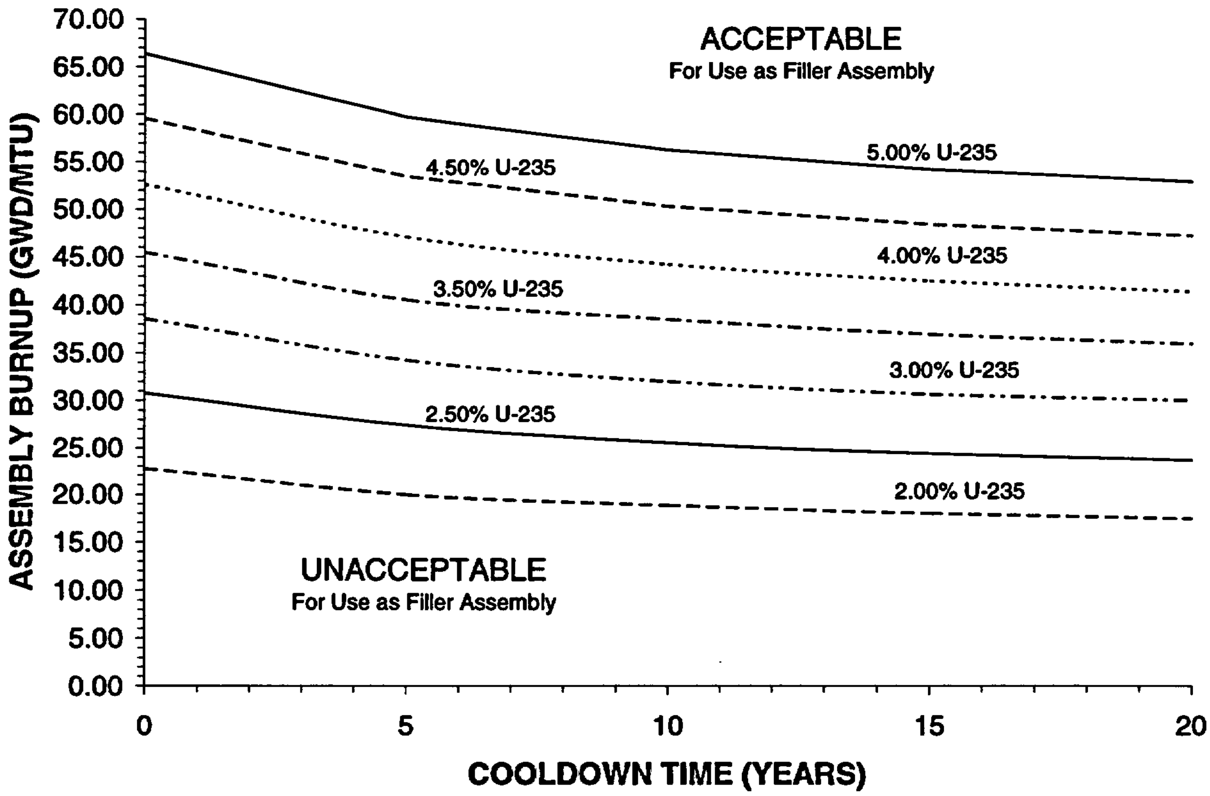


NOTES:

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 2 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 3.7.15-3 (Page 6 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type W-OFA

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	22.71	30.79	38.56	45.46	52.60	59.57	66.38
5	20.01	27.42	34.25	40.55	47.08	53.46	59.71
10	18.87	25.56	32.01	38.51	44.22	50.31	56.27
15	18.00	24.44	30.67	36.96	42.51	48.42	54.24
20	17.43	23.71	30.01	35.96	41.41	47.20	52.92

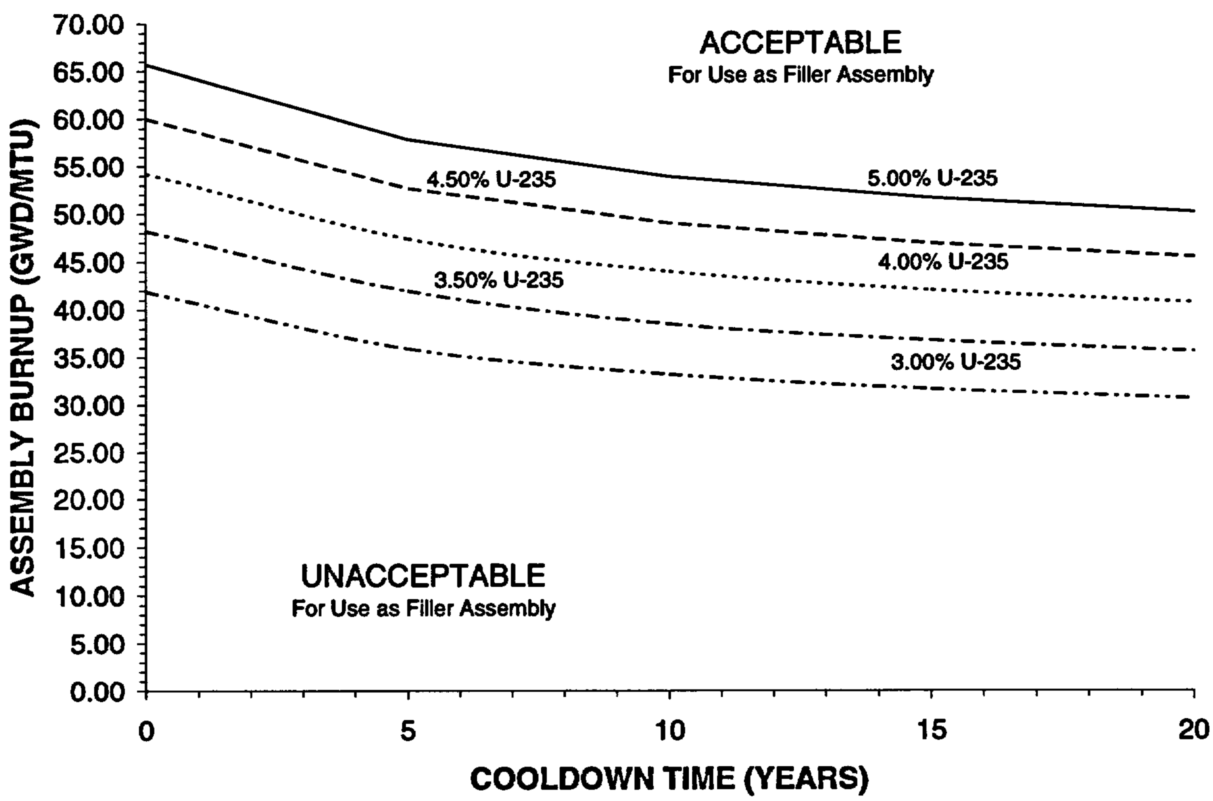


NOTES:

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 2 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 3.7.15-3 (Page 7 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Filler Region 2 Storage
For Fuel Assembly Type W-RFA

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	41.90	48.19	54.22	60.04	65.72
5	35.92	41.96	47.39	52.66	57.81
10	33.22	38.50	44.00	49.01	53.90
15	31.66	36.77	42.03	46.89	51.62
20	30.66	35.65	40.76	45.49	50.14

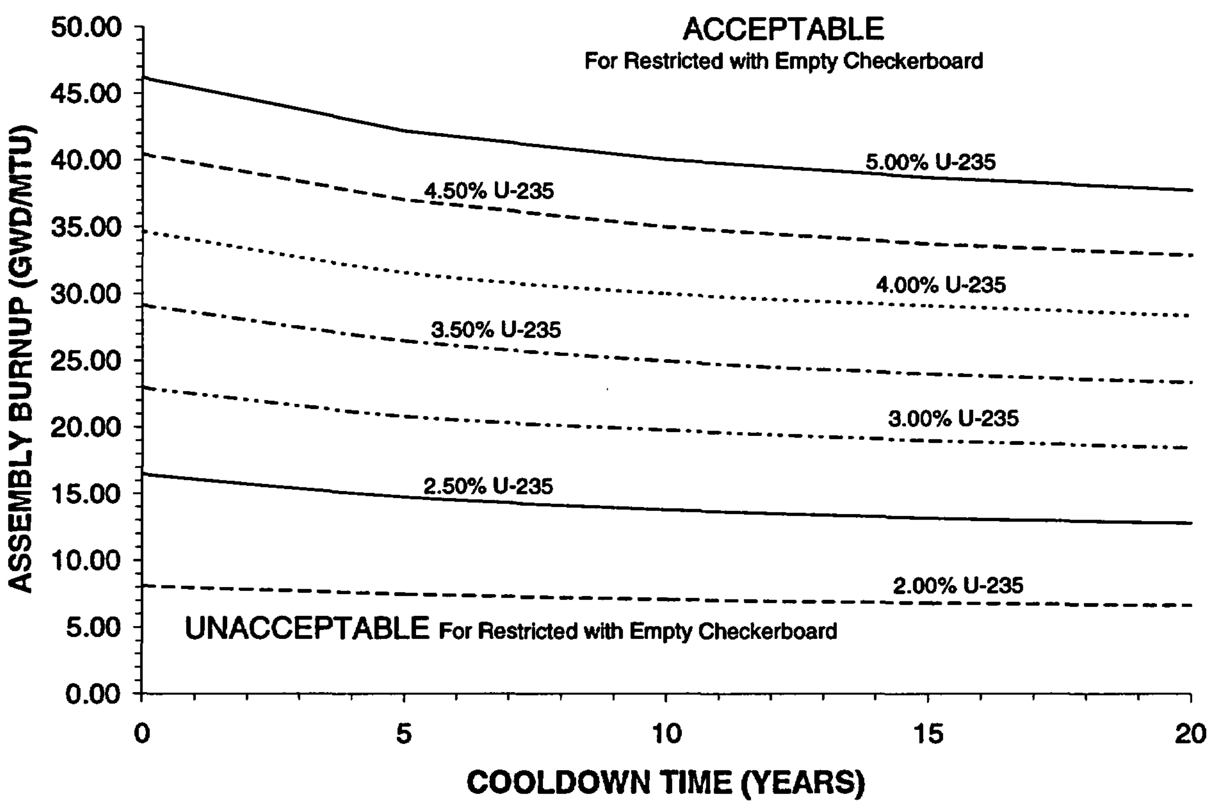


NOTES:

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 2 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 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type MkBW

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	8.12	16.50	22.94	29.15	34.67	40.43	46.20
5	7.49	14.77	20.81	26.50	31.60	37.03	42.18
10	7.07	13.77	19.79	24.96	30.01	34.99	40.01
15	6.81	13.16	18.98	24.00	29.10	33.73	38.67
20	6.64	12.78	18.45	23.37	28.35	32.90	37.75

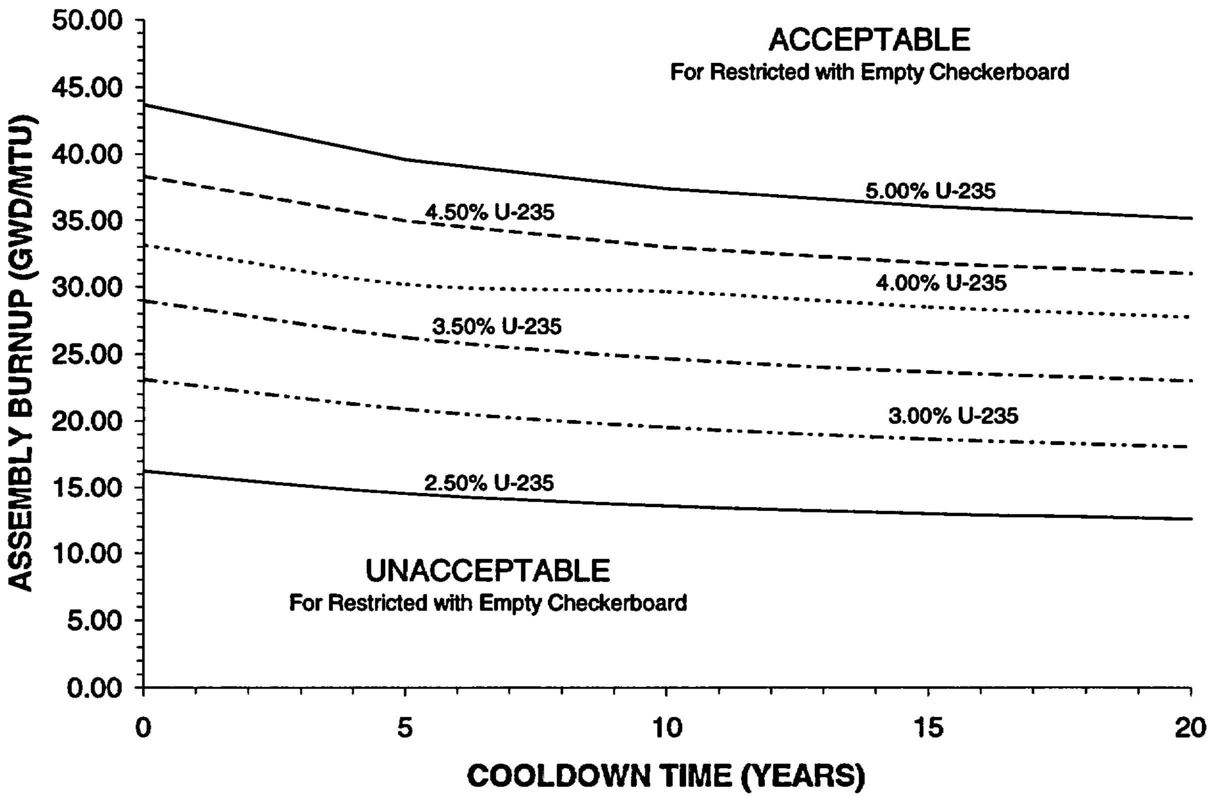


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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 2 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type **MkBWb1**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time						
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)					
	2.50	3.00	3.50	4.00	4.50	5.00
0	16.23	23.10	28.95	33.14	38.33	43.68
5	14.53	20.89	26.24	30.19	34.95	39.57
10	13.55	19.52	24.65	29.63	32.99	37.40
15	12.95	18.65	23.66	28.48	31.78	36.06
20	12.58	18.08	23.01	27.73	30.99	35.17

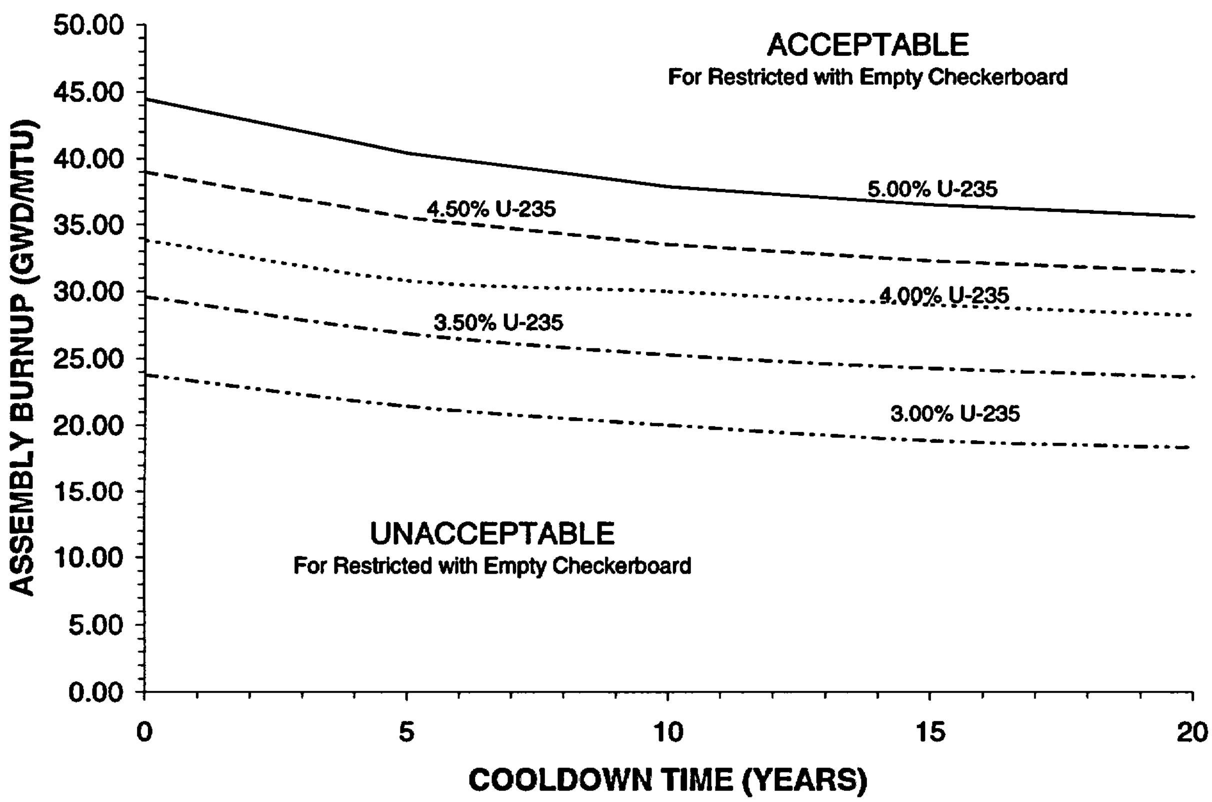


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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 3 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type MkBWb2

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	23.77	29.59	33.83	38.97	44.48
5	21.44	26.88	30.81	35.52	40.41
10	20.03	25.29	30.01	33.52	37.89
15	18.88	24.29	29.01	32.29	36.53
20	18.35	23.64	28.25	31.48	35.62

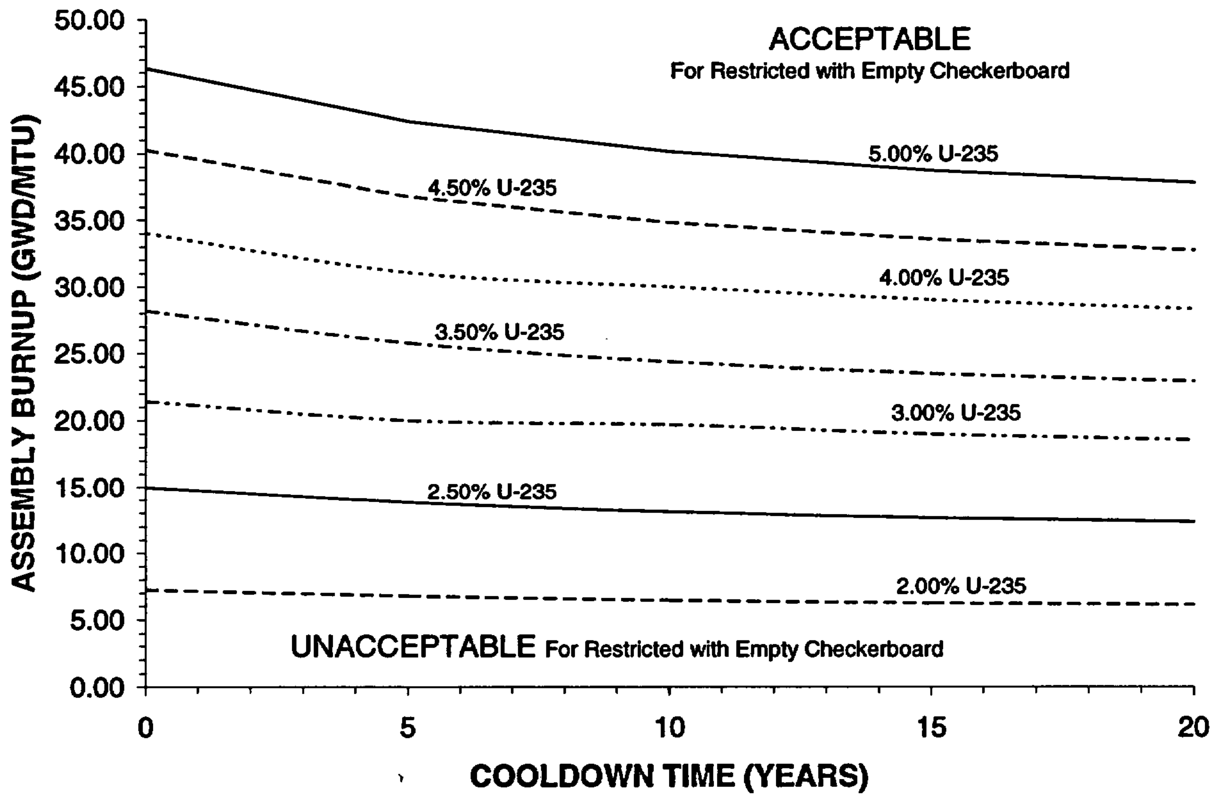


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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 4 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type **W-STD**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	7.24	14.97	21.42	28.18	34.01	40.25	46.34
5	6.79	13.83	20.01	25.78	31.08	36.80	42.40
10	6.46	13.11	19.69	24.37	30.01	34.81	40.14
15	6.24	12.65	18.99	23.48	29.01	33.57	38.74
20	6.11	12.37	18.54	22.91	28.31	32.76	37.83

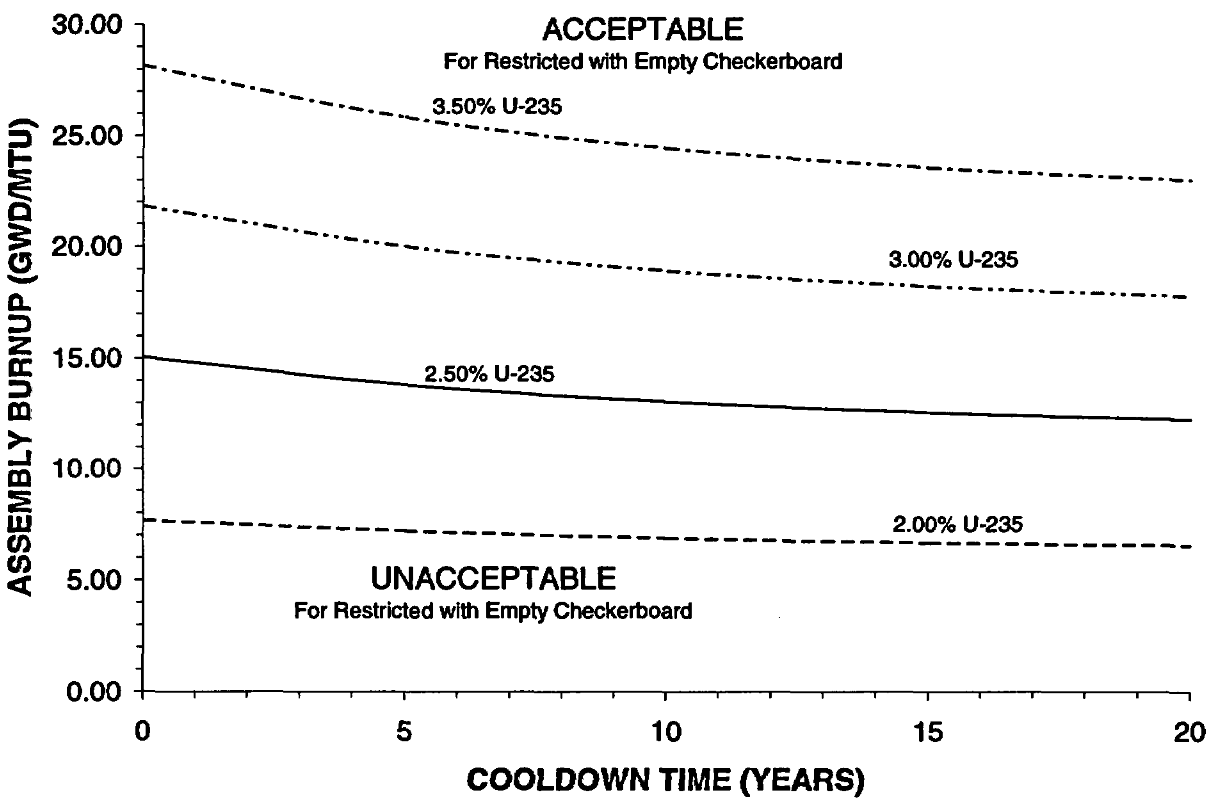


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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 5 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type MkBI

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time				
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)			
	2.00	2.50	3.00	3.50
0	7.67	15.06	21.81	28.16
5	7.20	13.82	20.01	25.83
10	6.86	13.05	18.92	24.44
15	6.66	12.57	18.23	23.56
20	6.53	12.26	17.78	22.99

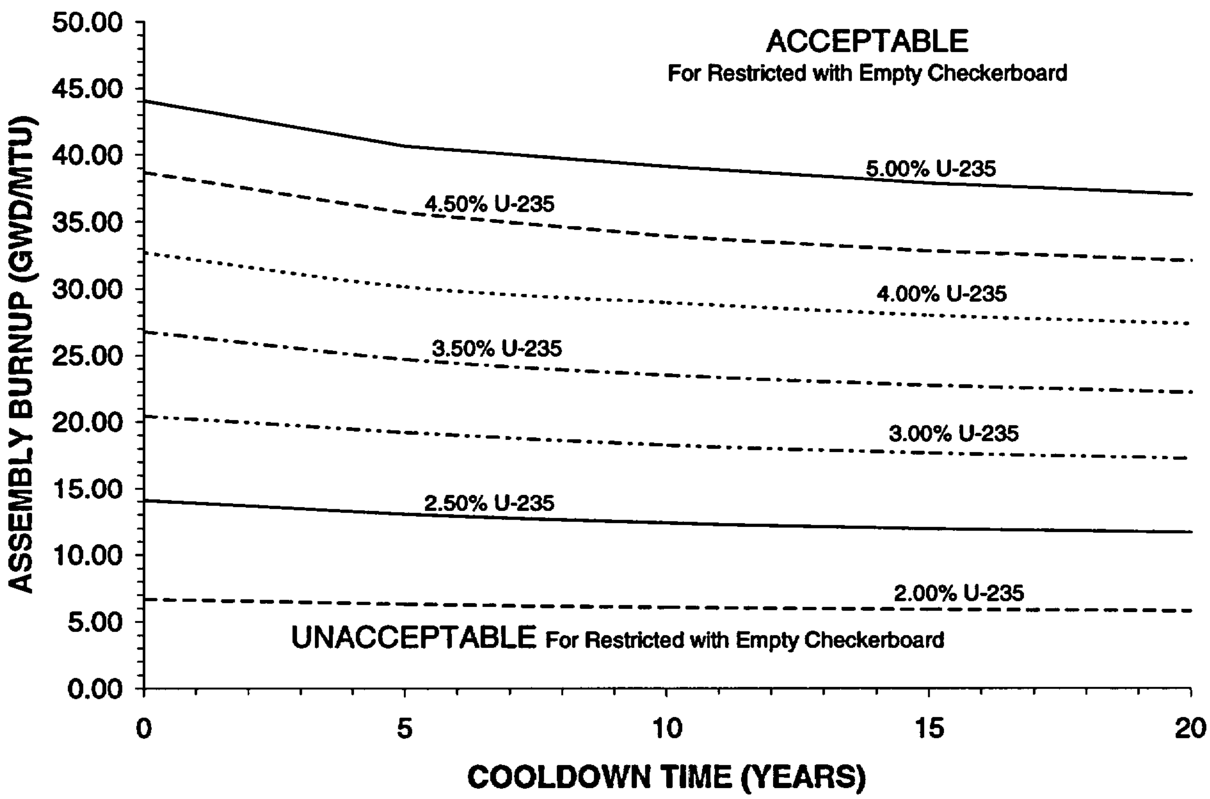


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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 6 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type **W-OFA**

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time							
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)						
	2.00	2.50	3.00	3.50	4.00	4.50	5.00
0	6.69	14.08	20.44	26.78	32.70	38.68	44.03
5	6.32	13.06	19.21	24.72	30.15	35.68	40.65
10	6.07	12.40	18.26	23.50	28.94	33.93	39.12
15	5.91	11.98	17.66	22.73	28.00	32.83	37.87
20	5.82	11.71	17.27	22.22	27.38	32.10	37.05

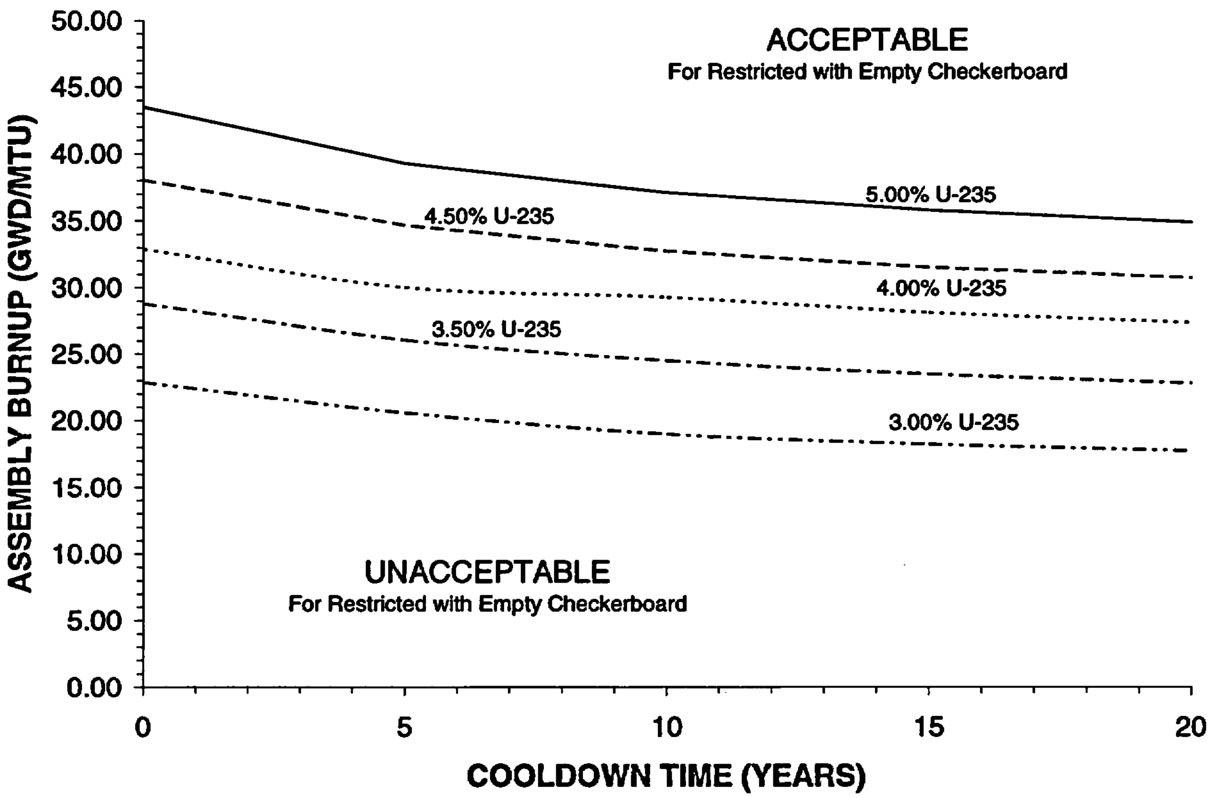


NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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.

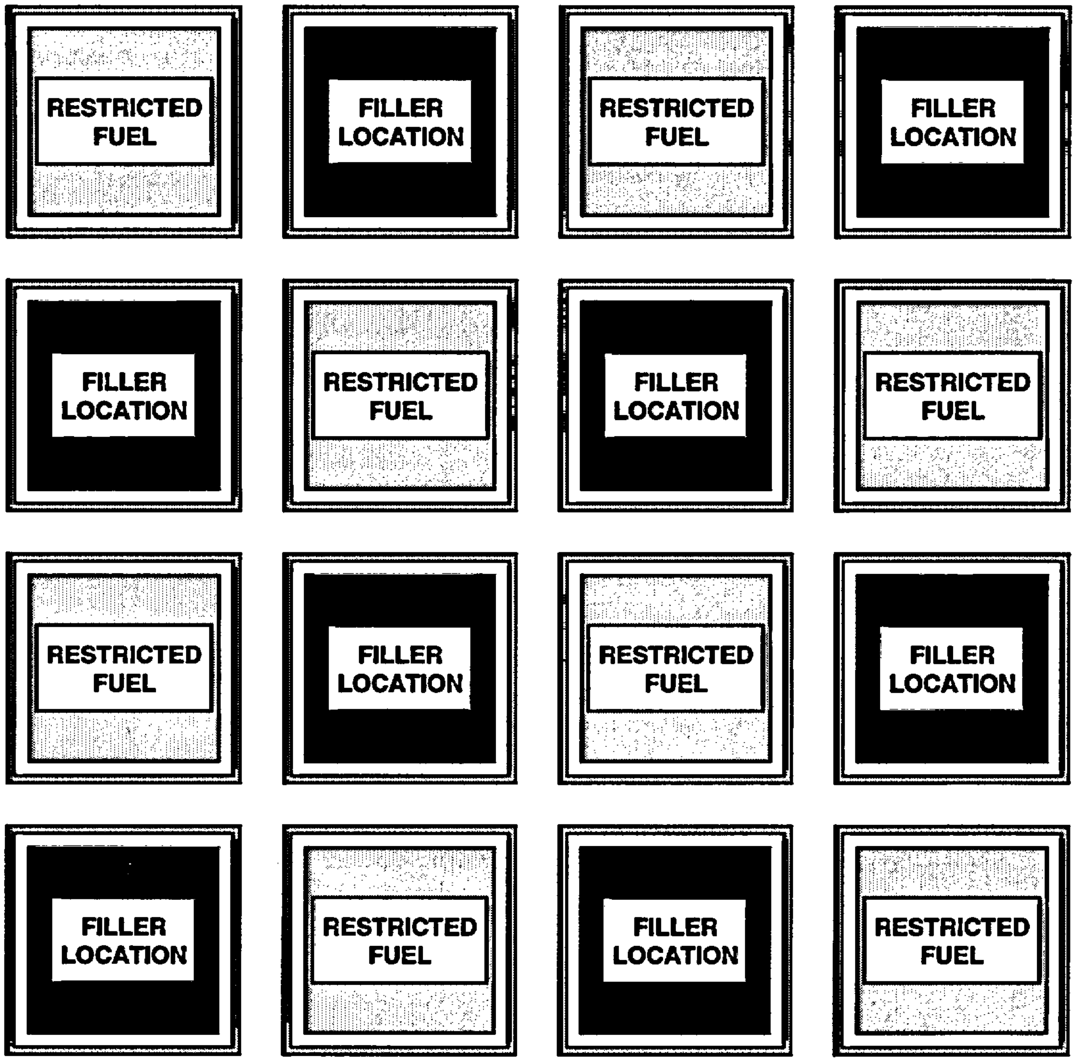
Table 3.7.15-4 (Page 7 of 7)
Minimum Qualifying Burnup Versus Initial Enrichment and Cooldown Time
For Restricted with Empty Checkerboard Region 2 Storage
For Fuel Assembly Type W-RFA

Burnup (GWD/MTU) versus Initial Nominal Enrichment and Cooldown Time					
Cooldown Time (years)	Initial Nominal Enrichment (% U-235)				
	3.00	3.50	4.00	4.50	5.00
0	22.87	28.80	32.88	38.06	43.49
5	20.56	26.06	30.01	34.66	39.27
10	18.99	24.48	29.27	32.73	37.11
15	18.23	23.47	28.12	31.52	35.77
20	17.74	22.81	27.37	30.73	34.89



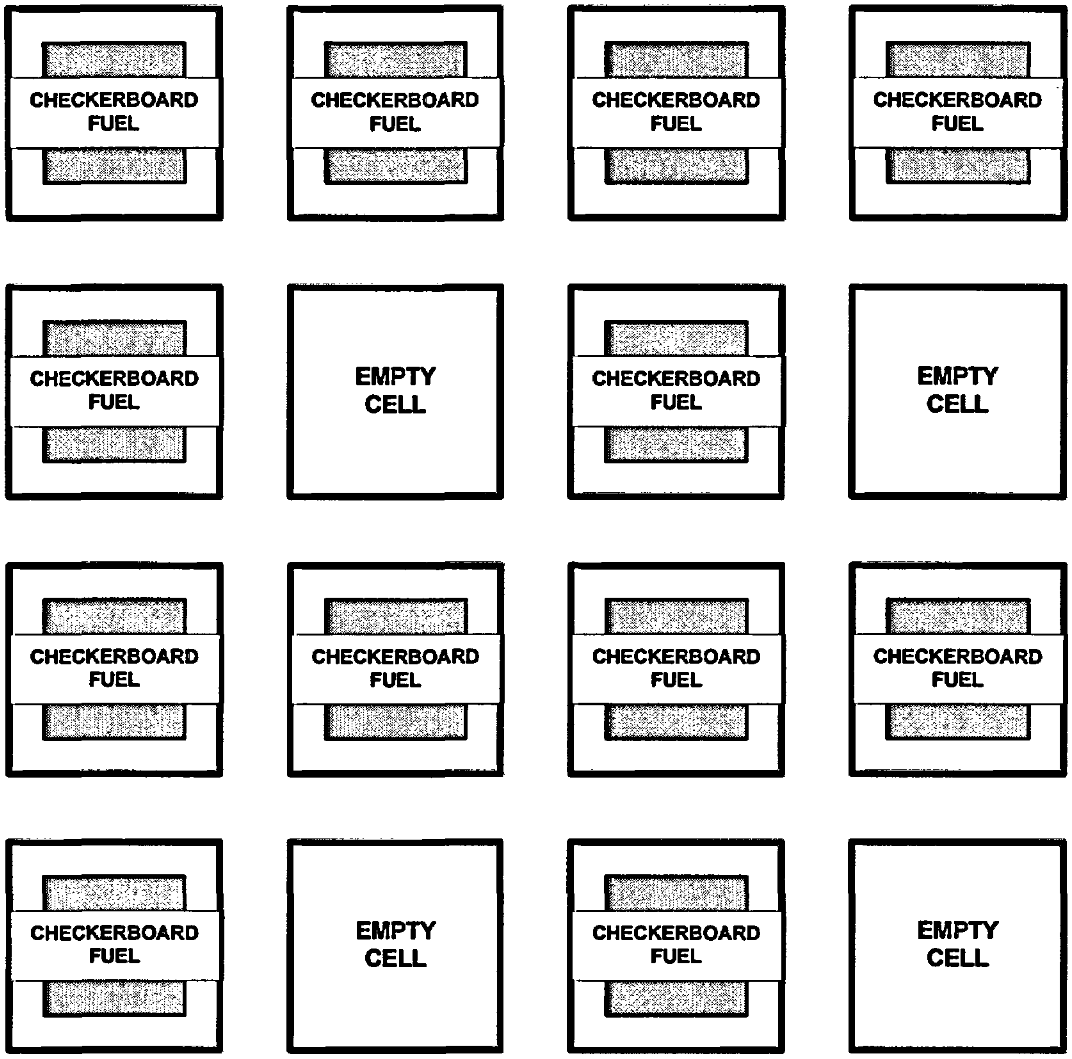
NOTES:

Fuel which differs from those designs used to determine the requirements of Table 3.7.15-4 may be qualified for use as a Region 2 Restricted with Empty Checkerboard 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 meets the minimum burnup requirements of Table 3.7.15-2, or non-fuel components, or an empty cell.
Filler Location:	Either fuel which meets the minimum burnup requirements of Table 3.7.15-3, or an empty cell.
Boundary Condition:	None.

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



Checkerboard Fuel: Fuel which meets the minimum burnup requirements of Table 3.7.15-4, or non-fuel components, or an empty cell.

Boundary Condition: Row or Column of only Checkerboard Fuel (Example: Row 1 or Column 1) shall be bounded by either: a) Alternating pattern of Checkerboard Fuel and empty cell, b) String of empty cells, or c) Spent fuel pool wall. No boundary conditions for a row or column of alternating pattern of Checkerboard Fuel and empty cell (Example: Row 4 or Column 4)

Figure 3.7.15-2 (page 1 of 1)
Required 3 out of 4 Loading Pattern for Checkerboard Region 2 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 5.00 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 800 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 Region 1 and

e. A nominal 9.125 inch center to center distance between fuel assemblies placed in Region 2.

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 5.00 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 Region 1 and 1177 total spaces in Region 2).

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

Description of Proposed Changes

Duke Power Company proposes to modify the McGuire Nuclear Station (MNS) Technical Specifications (TS) Sections 3.7.15, Spent Fuel Assembly Storage, and 4.3, Design Features – Fuel Storage. A markup of the specific changes is shown in Attachment 1. This License Amendment Request (LAR) provides revised spent fuel storage criteria based upon fuel type, fuel enrichment, burnup, cooling time and partial credit for soluble boron. In addition, this amendment allows for the safe storage of fuel assemblies with a nominal enrichment of U-235 up to 5.0 weight percent. Finally, this LAR reduces the required soluble boron credit from 850 ppm to 800 ppm.

The proposed TS changes are based upon the new McGuire Fuel Storage Criticality Analysis (Attachment 6). The criticality analysis was performed in accordance with the regulatory criteria of 10 CFR 50.68(b). The TS changes in this LAR include the following:

a) **LCO 3.7.15:** This LCO is modified by deleting the reference to Integral Fuel Burnable Absorber (IFBA) rods and replacing it with a reference to cooling time. The criticality analysis, as discussed in Attachment 6, is performed in accordance with the requirements of 10 CFR 50.68(b). The evaluation takes credit for Boral in the new Region 1 spent fuel storage racks and no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks or for IFBA rods that may be present. Credit is taken for burnup, and cooling time for the Region 2 spent fuel storage racks. In addition, the analysis utilizes partial credit for the soluble boron in the spent fuel pool water.

b) **LCO 3.7.15a:** This LCO Section defined the fuel limits and acceptable storage configurations for new or irradiated fuel to be stored within Region 1A of the spent fuel pool. The entire LCO Section is to be deleted and replaced with a new LCO Section that defines the requirements for the safe storage of new or irradiated fuel within Region 1 of the spent fuel pool. This LCO Section will specify that the maximum initial U-235 enrichment of fuel stored in Region 1 be equal to or less than 5.00 weight percent. There will be no other restrictions for the safe storage of new or irradiated fuel within Region 1.

c) **LCO 3.7.15b:** This LCO Section defined the fuel limits and acceptable storage configurations for new or irradiated fuel to be located in Region 1B of the spent fuel pool. The entire LCO Section is to be deleted and replaced with a new LCO Section that defines the requirements for the safe storage of new or irradiated fuel within Region 2 of the spent fuel pool. This LCO Section establishes three fuel storage configurations for Region 2; 1) unrestricted storage; 2) restricted storage or 3) checkerboard storage. The new TS Tables define the fuel limits for the three storage configurations and are referred within this LCO Section.

d) **LCO 3.7.15c:** This LCO Section defined the fuel limits and acceptable storage configurations for new or irradiated fuel to be located in Region 2A of the spent fuel pool. This entire LCO Section is to be deleted, since the criticality analysis no longer takes credit for Boraflex as a neutron absorbing material.

e) **LCO 3.7.15d:** This LCO Section defined the fuel limits and acceptable storage configurations for new or irradiated fuel to be located in Region 2B of the spent fuel pool. This entire LCO Section is to be deleted since the criticality analysis no longer takes credit for Boraflex as a neutron absorbing material.

f) **Tables 3.7.15-1 through 3.7.15-12:** These TS Tables specify the burnup and enrichment limits for fuel stored in the spent fuel racks. The data for these tables were derived from criticality analysis that relied on partial credit for Boraflex. These TS Tables are deleted since credit for Boraflex as a neutron absorber has been eliminated.

g) **Figures 3.7.15-1 through 3.7.15-7:** These TS Figures illustrate acceptable loading patterns and define the boundary conditions for various storage configurations. The loading patterns were based on the criticality analysis that relied on partial credit for Boraflex. These TS Figures are deleted since credit for Boraflex as a neutron absorber has been eliminated.

h) **New Table 3.7.15-1:** This TS Table specifies the minimum burnup requirements as a function of initial enrichment, fuel assembly design type and post-irradiation cooling time to be stored as unrestricted fuel in Region 2 of the spent fuel racks. The data for this table is based on the criticality analysis discussed in Attachment 6. The analysis no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

i) **New Table 3.7.15-2:** This TS Table specifies the minimum burnup requirements as a function of initial enrichment, fuel assembly design type and post-irradiation cooling time to be stored in Region 2 of the spent fuel racks as a restricted fuel assembly for the 2 out of 4 restricted/filler storage configuration. The data for this table is based on the criticality analysis discussed in Attachment 6. The analysis no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

j) **New Table 3.7.15-3:** This TS Table specifies the minimum burnup requirements as a function of initial enrichment, fuel assembly design type and post-irradiation cooling time to be stored in Region 2 of the spent fuel racks as a filler fuel assembly for the 2 out of 4 restricted/filler storage configuration. The data for this table is based on the criticality analysis discussed in Attachment 6. The analysis no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

k) **New Table 3.7.15-4:** This TS Table specifies the minimum burnup requirements as a function of initial enrichment, fuel assembly design type and post-irradiation cooling time to be stored in Region 2 of the spent fuel racks as a checkerboard fuel assembly for the 3 out of 4 checkerboard/empty storage configuration. The data for this table is based on the criticality analysis discussed in Attachment 6. The analysis no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

l) **New Figure 3.7.15-1:** This TS Figure illustrates the loading pattern to be employed in the Region 2 spent fuel storage racks for the 2 out of 4 restricted/filler storage configuration. There are no boundary conditions specified for this storage configuration. The loading pattern illustrated by this figure is based on the criticality analysis discussed in Attachment 6. The analysis no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

m) **New Figure 3.7.15-2:** This TS Figure illustrates the loading pattern to be employed in the Region 2 spent fuel storage racks for the 3 out of 4 checkerboard/empty cell storage configuration. Boundary conditions for this storage configuration are specified. For this configuration, a string of checkerboard fuel is to be bounded by either; 1) an alternating pattern of checkerboard fuel and empty cell, 2) string of empty cells, or 3) spent fuel pool wall. The loading pattern illustrated by this figure is based on the criticality analysis discussed in Attachment 6. The analysis no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

n) **TS 4.3.1.1a:** The maximum nominal U-235 enrichment of fuel to be stored in the spent fuel storage racks is increased from 4.75 weight percent to 5.00 weight percent. The criticality analysis discussed in Attachment 6 is performed in accordance with the requirements of 10 CFR 50.68(b). The analysis takes credit for the Boral in the new Region 1 spent fuel storage racks and no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water. In addition, the analysis utilizes partial credit for the soluble boron in the spent fuel pool water.

o) **TS 4.3.1.1c:** The required soluble boron concentration necessary to maintain k_{eff} less than 0.95 is reduced from 850 ppm to 800 ppm. The criticality analysis confirms that 800 ppm of partial soluble boron credit is sufficient to maintain k_{eff} less than 0.95. The criticality analysis as discussed in Attachment 6 is performed in accordance with the requirements of 10 CFR 50.68(b). The analysis takes credit for the Boral in the new Region 1 spent fuel storage racks and no longer takes credit for any remaining Boraflex in the Region 2 spent fuel storage racks. For each fuel assembly design type, credit is taken for burnup, cooling time and partial credit for the soluble boron in the spent fuel pool water.

p) **TS 4.3.1.1d:** The TS specifies the nominal center to center spacing between fuel assemblies stored within Regions 1A and 1B. The subregion designations of 1A and 1B are deleted, since the criticality analysis no longer takes credit for Boraflex as a neutron absorbing material in Region l. The new designation stated in the TS is Region 1.

q) **TS 4.3.1.1e:** The TS specifies the nominal center to center spacing between fuel assemblies stored within Regions 2A and 2B. The subregion designations of 2A and 2B are deleted, since the criticality analysis no longer takes credit for Boraflex as a neutron absorbing material in Region 2. The new designation stated in the TS is Region 2.

r) **TS 4.3.1.2a:** The maximum nominal U-235 enrichment of fuel to be stored in the new fuel storage racks is increased from 4.75 weight percent to 5.0 weight percent. The criticality analysis discussed in Attachment 6 is performed in accordance with the requirements of 10 CFR 50.68(b)(2) & 10 CFR 50.68(b)(3). For the new fuel storage racks, all fuel is considered to be unirradiated within the analysis. The analysis takes no credit for spacer grids or other neutron poisons that may be inserted in the fuel assembly.

s) **TS 4.3.3:** The TS specifies the total storage capacity of the spent fuel storage pool. The subregion designations of 1A, 1B, 2A and 2B are deleted since the criticality analysis no longer takes credit for Boraflex as a neutron absorbing material. The new designations stated are Region 1 and Region 2.

Technical Justification

This section provides the technical justification for the proposed modifications to the MNS Technical Specifications. These changes address revised spent fuel storage criteria based upon fuel type, fuel enrichment, burnup, cooling time and partial credit for soluble boron. In addition, the nominal fuel enrichment that can be stored within the spent fuel racks is increased. Finally, this proposed amendment reduces the required soluble boron credit. These changes allow for the storage of fuel without the need to credit Boraflex for reactivity control in the MNS spent fuel pool. These changes, also, increase design and operational flexibility, while at the same time maintaining acceptable criticality safety margins and decay heat removal capabilities.

The existing design basis for preventing criticality in the McGuire spent fuel storage pools is that, including uncertainties, there is a 95% probability at a 95% confidence level that k_{eff} of the fuel storage assembly array will be less than 1.0 if fully flooded with unborated water, and k_{eff} will be equal to or less than 0.95 if fully flooded with water borated to 850 ppm, with credit for the presence of IFBA rods where applicable, and reduced credit for the degraded spent fuel rack Boraflex neutron absorber panels.

Each spent fuel pool contains a two region rack design. Region 1 racks (286 storage locations) have a fuel assembly spacing of 10.4 inches, utilizing a neutron absorbing material. These racks (2 modules/pool) are typically reserved for temporary core off loading and storage of non-irradiated fully enriched fuel. The Region 1 racks are composed of individual storage cells made of stainless steel that utilize Boral as the neutron absorbing material. The Region 1 racks had utilized Boraflex as the neutron absorbing material. In July 2003, the Region 1 racks were replaced with a similar designed rack, except for the neutron absorber material being Boral. The replacement of the Region 1 racks was performed per the provisions of 10 CFR 50.59.

Region 2 (1177 storage locations) has a fuel assembly spacing of 9.125 inches and utilizes Boraflex as a neutron absorbing material. The Region 2 racks provide normal long term storage for irradiated fuel assemblies and can be used for restricted storage of new fuel.

Currently, each region is further subdivided into two subregions based on the amount of remaining Boraflex. Placement of fuel into a given subregion without restriction is limited to assemblies meeting a certain minimum required assembly burnup versus enrichment. In the event that fuel assemblies do not meet the minimum requirements for unrestricted storage, a restricted storage configuration must be utilized. In the event that fuel assemblies do not meet the minimum requirements for restricted storage, a checkerboard storage configuration must be utilized.

McGuire TS 3.7.15 will be amended to provide revised spent fuel pool storage configurations, and revised spent fuel pool storage criteria, specifying minimum burnup requirements as a function of initial fuel enrichment, post-irradiation cooling time, and fuel assembly design type. With the applicable minimum concentration of soluble boron present in the spent fuel pool, and credit for the Boral neutron absorber panels where applicable, these changes will ensure that the pool storage rack k_{eff} is ≤ 0.95 under non-accident conditions, and accident conditions (including the unlikely occurrence of a credible spent fuel pool dilution event with thorough mixing). The applicable minimum concentration of soluble boron is ensured by existing McGuire TS 3.7.14.

The new McGuire Fuel Storage Criticality Analysis evaluates the Region 1 and Region 2 storage racks in the McGuire spent fuel pools. These spent fuel storage racks originally contained Boraflex poison panels for reactivity holddown. However; ongoing degradation of the Boraflex material has limited the effectiveness of continuing to rely on this poison material in Region 1 and Region 2. To address the continuing degradation of the Boraflex panels, the McGuire criticality analysis considers "permanent solutions" to this issue. The permanent solutions for Region 1 and Region 2 include the following:

Region 1 Re-rack with Boral poison panels. The old Region 1 Boraflex racks were replaced with racks designed, fabricated and supplied by Holtec in mid-2003. The new Region 1 racks have the same dimensions as the old racks, and thus the same storage capacity (286 cells). The new Region 1 racks will allow unrestricted storage of fresh fuel, up to 5.00 weight percent U-235.

Region 2 Retain existing racks, but eliminate credit for any remaining Boraflex poison. Take credit for cooling time reactivity reduction (due primarily to Pu-241 decay and Gd-155 buildup following the end of reactor irradiation). Segregate storage burnup requirements by fuel assembly type to take advantage of lower reactivity associated with certain fuel designs (seven different types have been identified). Finally, take credit for burnup in storage arrays containing empty cells, such as 3 assemblies with one empty cell. The Region 2 criticality analysis employs specific 3-D calculations for the fuel storage configurations that take credit for burnup.

The criticality evaluation demonstrated that the new Region 1 Boral racks can store fresh McGuire reactor fuel of any type, up to 5.00 weight percent of U-235, with no restrictions. The minimum burnup requirements for Region 2 storage were developed for seven different fuel types, as a function of initial enrichment and post-irradiation cooling time. These burnup requirements were specified for three Region 2 storage configurations: Unrestricted, 2 out of 4 Restricted/Filler, and 3 out of 4 Checkerboard/Empty.

For the spent fuel pool storage rack criticality analyses, the maximum 95/95 k_{eff} is determined to be less than 1.00 with no boron in the spent fuel pool water for both Region 1 and 2 storage racks. These results meet the no-boron 95/95 k_{eff} criterion in 10 CFR 50.68(b)(4). Further, the criticality analysis confirmed that 800 ppm of partial soluble boron credit is sufficient to maintain the maximum 95/95 k_{eff} less than 0.95. A minimum boron concentration of 1600 ppm is adequate to maintain the maximum 95/95 k_{eff} below 0.95 for a worst-case misloading event in the McGuire spent fuel pool. Finally, for the worst-case weir gate drop on the new Region 1 Boral racks, the maximum achievable 95/95 k_{eff} is well below the 0.95 subcriticality criterion, when credit is taken for 2475 ppm soluble boron in the SFP.

The new McGuire Fuel Storage Criticality Analysis demonstrates that under non-accident conditions a spent fuel storage pool boron concentration of 800 ppm would be adequate to maintain the spent fuel storage rack $k_{eff} \leq 0.95$. Existing McGuire TS 3.7.14 states that the spent fuel pool storage boron concentrations shall be maintained within the limits specified in the McGuire Core Operating Limits Report (COLR). The spent fuel pool boron concentration limit currently specified in the COLR is 2675 ppm, which is well above the minimum required boron credit of 800 ppm for non-accident conditions.

A possibility does exist that the boron concentration in the spent fuel pool could be lowered below the COLR limit by a pool dilution event. Consequently, an analysis of a dilution event of the spent fuel pool boron concentration is necessary to ensure that acceptable levels of subcriticality are maintained during and following the event. As part of this analysis, calculations were performed to define the dilution time and volumes for the spent fuel pool. The dilution sources available at McGuire were compiled and evaluated against the calculated dilution volume to identify the bounding "continuous flow" dilution event. The McGuire dilution analysis concluded that the bounding event was a pipe break in the non-seismic fire protection system, as this could deliver the largest flow rate (700 gpm) of unborated water into the SFP. For this dilution event, in conjunction with an isolation of the cask loading pit, calculations determined that it would take at least 9.5 hours to dilute the SFP from an initial boron concentration of 2675 ppm to 800 ppm. Such a scenario would involve substantial overflow of the SFP in less than two hours, and it was deemed incredible, because numerous indicators such as level alarms, flooding in the auxiliary building, fire protection pump header flow alarms, etc., would alert Operations long before 9 hours had elapsed (Reference 4).

The above post-dilution event is based upon the assumption that all of the unborated water is thoroughly mixed with the water in the pool. Given the spent fuel storage pool cooling water flow and convection from the spent fuel decay heat, it is likely that this thorough mixing would occur. However, if mixing was not adequate, it is possible that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the calculation in Attachment 6 which shows that a spent fuel storage pool k_{eff} will still be less than 1.0 on a 95/95 basis with the spent fuel pool filled with unborated water. Thus, in the unlikely event that the worst case dilution event occurred and then a pocket of non-borated water formed in the spent fuel pool due to inadequate mixing, acceptable subcritical conditions would still be maintained in the McGuire spent fuel storage pools.

Many of the postulated spent fuel pool accidents at McGuire will not result in an increase in k_{eff} of the spent fuel racks. Such accidents are the drop of a fuel assembly on top of a rack, the drop of a fuel assembly between rack modules, and the drop of a fuel assembly between rack modules and the pool wall. At McGuire, the spent fuel assembly rack configuration is such that it precludes the insertion of a fuel assembly between rack modules. The placement of an assembly between the rack and the pool wall would result in a lower k_{eff} relative to the criticality analysis due to the increased neutron leakage at the spent fuel pool wall because the criticality analysis assumes an infinite array of fuel assemblies. In the case where a dropped fuel assembly in its most reactive condition is dropped onto the spent fuel racks, it is assumed that the rack structure pertinent for criticality is not excessively deformed. For this event, previous accident analysis with unborated water showed that a dropped fuel assembly resting horizontally on top of the spent fuel rack has sufficient separation from the active fuel height of stored fuel assemblies to preclude neutronic interaction.

However, three accidents can be postulated which could result in an increase in reactivity in the spent fuel storage pools. The first is the misloading of a fuel assembly. Another postulated accident to be addressed is a significant change in the spent fuel pool water temperature. The third event is a heavy load drop (limited to Region 1 racks).

A fuel assembly misload accident relates to the use of administratively controlled storage locations based on fuel assembly type, initial enrichment, burnup and cooling time. The misloading of a fuel assembly constitutes not meeting the enrichment, burnup or cooling time requirements for that administratively controlled location. The result of the misloading is to add positive reactivity, increasing k_{eff} toward 0.95. For Region 1, any type of McGuire reactor fuel, with any enrichment (up to 5.00 weight percent) and burnup, can be stored without restriction in the Region 1 racks. As such, there is no possibility of a misloading accident in Region 1. For Region 2, the worst-case misload event involves placing a fresh 5.00 weight percent W-OFA fuel assembly in an empty cell, within the 3 out of 4 Checkerboard/Empty storage configurations. The analysis of this event demonstrates that 1600 ppm is sufficient to ensure that the SFP Region 2 system k_{eff} remains below 0.95.

A significant change in the spent fuel pool water temperature can be caused by either the loss of normal cooling to the spent fuel pool water which causes an increase in the temperature of the water passing through the stored fuel assemblies or a large makeup to the pool with cold water which could happen if the spent fuel pool were used an as emergency source of borated water. Loss of spent fuel pool cooling causes water density to decrease, typically increases reactivity in the SFP. A decrease in pool temperature causes water density to increase, typically reduces SFP reactivity. However, this event is bounded by the misloading accident, which is much more severe, from a criticality perspective, than a change in SFP water temperature.

As far as loads heavier than a fuel assembly are concerned, the largest loads that may be moved over the Region 1 area of the McGuire SFPs are the weir gates. An analysis of the criticality consequences of a worst-case weir gate drop on the new Region 1 Boral racks demonstrates that even with up to 9 fuel assemblies crushed by the weir gate into an optimum-reactivity configuration, the maximum achievable 95/95 k_{eff} (0.874) is well below the 0.95 subcriticality criterion, when credit is taken for 2475 ppm soluble boron in the SFP. The heavy load drop accident does not need to be considered for Region 2, because the weir gate is not carried directly over Region 2, and thus an end-drop of the gate onto Region 2 – the only type of weir gate drop capable of deforming the storage racks – is not possible.

In summary, for each of the accidents evaluated, these analyses determined that the minimum boron concentration required to maintain k_{eff} less than or equal to 0.95 is well below the spent fuel pool storage boron concentrations specified in the McGuire Core Operating Limits Report (COLR). The spent fuel pool boron concentration limit currently specified in the COLR is 2675 ppm. Consequently, under the applicable accident conditions, maintaining spent fuel pool boron concentrations within the COLR limit will ensure that the spent fuel storage rack k_{eff} is $\leq$ 0.95 when fuel is stored in accordance with the revised spent fuel pool storage configurations and storage criteria (fuel enrichment limits, specified fuel assembly design types, post-irradiation cooling time and burnup requirements) in the proposed changes to TS 3.7.15.

The current TS 3.7.15 specifies the requirements for spent fuel pool storage configurations with fuel pool storage criteria involving fuel enrichment and fuel burnup. Consequently, plant operating procedures already include controls to ensure these existing requirements are satisfied. These procedural controls will be revised and maintained as needed under the revised TS 3.7.15. In addition, new controls necessary to ensure that independent administrative confirmation of fuel type and for determining cooling time achieved will be incorporated into plant operating procedures prior to implementation of the proposed TS changes. Note that existing McGuire spent fuel pool storage systems, spent fuel pool cooling systems, fuel handling systems instrumentation and other supporting systems are not modified as a result of this proposed LAR.

McGuire TS 4.3 will be revised to increase the maximum allowable U-235 enrichment from 4.75 to 5.00 weight percent that can be stored in the spent fuel storage racks and in the new fuel storage racks, to decrease the boron concentration required to maintain $k_{eff} \leq 0.95$ from 850 ppm to 800 ppm, and to eliminate the sub-region designation within Regions 1 and 2 (replace designation Region 1A, 1B, 2A, & 2B with Region 1 and Region 2).

The criticality analysis for the New Fuel Vault storage racks is performed in accordance with the requirements of 10 CFR 50.68(b). This analysis determined that the New Fuel Vault storage racks can store unirradiated MkBW (with or without axial blankets), W-RFA, and W-STD fuel up to 5.00 weight percent of U-235, with no location restrictions. Fresh W-OFA fuel up to 4.76 weight percent of U-235 may be stored with no location restrictions. The analysis determined that the maximum 95/95 k_{eff} if the New Fuel Vault area is flooded with full-density unborated water would be **0.9498** and if flooded with optimum-moderation unborated water, the maximum 95/95 k_{eff} would be **0.9618**. These results meet the requirements of 10 CFR 50.68(b)(2).

As noted in the criticality analysis for the New Fuel Vault storage racks, fuel design type W-OFA is limited to 4.76 weight percent of U-235. Note that only fresh, un-irradiated fuel can be stored in the New Fuel Vault storage racks. Fuel design type W-OFA was utilized in batches 4 through

9 for both McGuire Units. Further, the current operating cycles for both units do not contain this fuel design type. As such, all fuel design type W-OFA assemblies have been irradiated and, thus are stored in the spent fuel pools. The current fuel design type in use at McGuire is W-RFA. In addition, there are no plans to utilize the W-OFA fuel design type in future operating cycles at MNS. As such, storage of W-OFA assembly within the New Fuel Vault storage racks is highly unlikely. In addition, the design requirements specified by TS 4.3.1.2b and TS 4.3.1.2c provide the necessary regulatory control regarding the safe storage of fuel assemblies within the New Fuel Vault storage racks. These TS requirements will ensure that the nominal enrichment of a W-OFA assembly that would be stored within the New Fuel vault storage racks is equal to or less than 4.76 weight percent of U-235.

Conclusion

Revision of the McGuire TS's as proposed in this LAR will provide a level of safety comparable to the conservative criticality analysis methodology required by References 1, 2, and 3 of this attachment. Consequently, the health and safety of the public will not be adversely affected by the proposed Technical Specification changes. The bases for these conclusions are as follows:

1. Utilizing the revised spent fuel pool storage configurations and revised spent fuel pool storage criteria (fuel enrichment limits, identified fuel assembly design types, cooling time and burnup requirements) specified in the proposed change to TS 3.7.15, the new McGuire Fuel Storage Criticality Analysis demonstrates that a minimum spent fuel storage pool boron credit of 800 ppm would be adequate to maintain the spent fuel storage rack $k_{eff} \leq 0.95$. This minimum boron concentration is ensured by existing McGuire TS 3.7.14.

2. Utilizing the revised spent fuel pool storage configurations and revised spent fuel pool storage criteria (fuel enrichment limits, identified fuel assembly design types, cooling time and burnup requirements) specified in the proposed change to TS 3.7.15, the new McGuire Fuel Storage Criticality Analysis demonstrates that spent fuel storage rack k_{eff} would remain below 1.0 with the spent fuel pool fully flooded with unborated water.

3. The new McGuire Spent Fuel Pool Criticality Analysis demonstrates that the amount of soluble boron necessary to ensure that the spent fuel rack k_{eff} will be maintained less than or equal to 0.95 following a significant change in spent fuel pool temperature or the misloading of a fuel assembly is well below the spent fuel pool storage boron concentrations specified in TS 3.7.14 and in the McGuire Core Operating Limits Report (COLR). The analysis also demonstrates that for the worst-case weir gate drop on the new Region 1 Boral racks, the maximum achievable 95/95 k_{eff} is well below the 0.95 subcriticality criterion, when full credit is taken for the soluble boron concentration in the SFP.

References

1. Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, Memorandum from L. Kopp (NRC) to T. Collins (NRC), U.S. Nuclear Regulatory Commission, August 19, 1998.

2. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-O800, June 1987.

3. Title 10 of the Code of Federal Regulations Part 50 Section 68

4. Duke Power letter to the NRC, dated October 30, 2002, Response to NRC Request for Additional Information – Boron Dilution Analyses.

5. McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments for Spent Fuel Pool (TAC NOs. MB5014 and MB5015)," Letter from R. Martin (NRC) to D. Jamil (Duke), February 4, 2003.

6. McGuire Nuclear Station Re: Issuance of Exemption to 10 CFR 70.24, Criticality Accident Requirements (TAC NOs. M97863, M97864, MB5014 and MB5015), Letter from R. Martin (NRC) to D. Jamil (Duke), January 31, 2003.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

No Significant Hazards Consideration Evaluation

In accordance with 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: 1) Involve a significant increase in the probability or consequences of an accident previously evaluated; 2) Create the possibility of a new or different kind of accident from any previously evaluated, or; 3) Involve a significant reduction in a margin of safety. This proposed amendment provides revised spent fuel storage criteria based upon fuel type, fuel enrichment, burnup, cooling time and partial credit for soluble boron. In addition, this amendment also allows for storage of fuel assemblies with a nominal enrichment up to 5.0 weight percent of U-235. Finally, this proposed amendment reduces the required soluble boron credit from 850 ppm to 800 ppm. In accordance with the criteria set forth in 10 CFR 50.91 and 50.92, McGuire Nuclear Station has evaluated the proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Will the change involve a significant increase in the probability or consequence of an accident previously evaluated?**

   NO The change in the amount of soluble boron specified by Specification 4.3 has no impact on the likelihood or consequences of any previously evaluated accident. This decrease in the soluble boron specified is not considered to be an initiator of any accidents nor does it influence how previously evaluated accidents are mitigated.

   There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pools when allowing for credit to be taken for different fuel types, fuel enrichments, burnup, plutonium decay and soluble boron to maintain an acceptable margin of subcriticality in the spent fuel pool. The increase of the nominal fuel enrichment for storage within the spent fuel pool does not increase the likelihood of a fuel assembly drop accident. The method of handling fuel assemblies in the spent fuel pool is not affected by the changes made to the criticality analysis for the spent fuel pool or by the proposed TS changes. The handling of fuel assemblies during normal operation is unchanged, since the same equipment and procedures will be used.

   The radiological consequences of a fuel assembly drop accident will not be adversely impacted due to taking credit for different fuel types, fuel enrichments, burnup, plutonium decay and soluble boron for criticality control in the spent fuel pool in the criticality analysis. The fission product inventory of individual fuel assemblies will not change significantly as a result of an increase in the nominal fuel enrichment. The criticality analysis showed that the consequences of a fuel assembly drop accident in the spent fuel pools are not affected when allowing for credit to be taken for different fuel types, fuel enrichments, burnup, plutonium decay and soluble boron to maintain an acceptable margin of subcriticality in the spent fuel pool.

There is no significant increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel pool racks when allowing for credit to be taken for different fuel types, fuel enrichments, burnup, cooling time and soluble boron to maintain an acceptable margin of subcriticality in the spent fuel pool. Fuel assembly placement and storage will continue to be controlled pursuant to approved fuel handling procedures and other approved processes to ensure compliance with the Technical Specification requirements. These procedures and processes will be revised as needed to comply with the revised requirements which would be imposed by the proposed Technical Specification changes. The proposed amendment decreases the number of different storage configurations specified by Technical Specification 3.7.15, but the number of criteria to consider increases. However, the revised storage requirements and criteria are considered no more complicated then what is currently specified by Technical Specifications. In some ways, the proposed amendment simplifies the process for identifying the placement of fuel assemblies within appropriate locations in the spent fuel pool storage racks. For instance, boundary conditions between storage configurations are significantly simpler. As such, station procedures and processes for appropriate placement of fuel assemblies in the spent fuel pool storage rack will continue to provide additional assurance that an accidental misloading of a spent fuel assembly will not occur.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate soluble boron concentration. Current Technical Specification 3.7.14 ensures that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

The probabilities of a loss of spent fuel pool cooling or reduction of pool temperature are not influenced by the proposed amendment changes. Fuel storage requirements, nominal fuel enrichment, or the amount of soluble boron present in the spent fuel pool water are not initiators of a loss of spent fuel pool cooling accident or in events resulting in a decrease in the pool water temperature. The consequences of a loss of Spent Fuel Pool cooling is not affected by this change. The concern with this accident is a reduction of spent fuel pool water inventory from bulk pool boiling resulting in uncovering fuel assemblies. Loss of spent fuel pool cooling at McGuire is mitigated in the usual manner by ensuring that a sufficient time lapse exists between the loss of forced cooling and uncovering fuel. This period of time is compared against a reasonable period to re-establish cooling or supply an alternative water source. The heat up rate in the spent fuel pool is a nearly linear function of the fuel decay heat load. The fuel decay heat load will not increase subsequent to the proposed changes since the number of fuel assemblies and the fuel burnups are unchanged. In the unlikely event that all pool cooling is lost, sufficient time will still be available for the operators to provide alternate means of cooling before the onset of pool boiling. Therefore, the proposed changes represents no increase in the consequences of loss of pool cooling.

A decrease in pool water temperature from a large emergency makeup causes an increase in water density, increasing reactivity. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (800 ppm), will compensate for the increased reactivity which could result from a decrease in spent fuel pool water temperature. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a decrease in pool water temperature will not be increased. Current Technical Specification 3.7.14 ensures that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

2. Will the change create the possibility of a new or different kind of accident from any previously evaluated?

NO Criticality and other related accidents within the spent fuel pool are not new or different types of accidents. They have been analyzed in the Updated Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments. Specific accidents considered and evaluated include fuel assembly drop, accidental misloading of spent fuel assemblies into the spent fuel pool racks, and significant changes in spent fuel pool water temperature. The accident analysis in the Updated Final Safety Analysis Report remains bounding.

The possibility for creating a new or different kind of accident is not credible. In a previous amendment request, taking credit for the soluble boron in the spent fuel pool water for reactivity control in the spent fuel pool was approved by the NRC. For the proposed amendment, the spent fuel pool dilution evaluation demonstrates that a dilution of the boron concentration in the spent fuel pool water which could increase the rack k_{eff} to greater than 0.95 continues not to be a credible event. The proposed amendment regarding fuel storage requirements, nominal fuel enrichment, and amount of soluble boron in the spent fuel pool water specified by Specification 4.3 will have no effect on normal pool operations and maintenance. There are no changes in equipment design or in plant configuration. The Technical Specification changes will not result in the installation of any new equipment or modification of any existing equipment. Therefore, the proposed amendment will not result in the possibility of a new or different kind of accident.

3. Will the change involve a significant reduction in a margin of safety?

NO The proposed Technical Specification changes and the resulting spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Attachment 6). This methodology takes partial credit for soluble boron in the spent fuel pool 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% probability, 95% confidence (95/95) level; and

2) k_{eff} shall be less than or equal to 0.95 if flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The criticality analysis utilized credit for soluble boron to ensure k_{eff} will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 k_{eff} calculation to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including uncertainties, tolerances and accident conditions in the presence of spent fuel pool soluble boron. The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to exceeding a k_{eff} of 0.95 has been evaluated and shown to be not credible. Accordingly, the required margin to criticality is not reduced.

Therefore the proposed changes in this license amendment will not result in a significant reduction in the facility's margin of safety.

References

1. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, June 1987.

2. ANS, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations, ANSI/ANS-57.2-1983.

3. Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, Memorandum from L. Kopp (NRC) to T. Collins (NRC), U.S. Nuclear Regulatory Commission, August 19, 1998.

4. Attached McGuire Criticality Analysis and other attached documentation (including references therein) forming the basis for this license amendment request.

ATTACHMENT 5

ENVIRONMENTAL IMPACT ASSESSMENT

Environmental Impact Assessment:

The proposed Technical Specification amendment has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed amendment will allow credit to be taken for different fuel types, burnup, cooling time and soluble boron to maintain an acceptable margin of subcriticality in the spent fuel pool. Appropriate controls are in place to monitor the soluble boron concentration in the spent fuel pool water and to monitor the placement of different fuel types in the spent fuel storage cells. Consequently, the proposed amendment does not involve a significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase individual or cumulative occupational radiation exposures. Therefore, the proposed amendment meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Assessment.

ATTACHMENT 6

MCGUIRE FUEL STORAGE CRITICALITY ANALYSIS

1 Introduction

This analysis examines the criticality aspects of fuel storage in the McGuire new fuel storage vaults (NFVs) and spent fuel pools (SFPs), to ensure that all pertinent regulatory subcriticality criteria are satisfied for proposed configurations of fuel stored in these areas. The objective of this criticality evaluation is to demonstrate that:

- Fresh fuel up to 5.0 wt % U-235 may be stored in the NFV.
- Fresh or irradiated fuel up to 5.0 wt % U-235 may be stored in the SFP if specific requirements for minimum burnup, fuel assembly design, cooling time, and storage pattern are met.

The NFV criticality evaluation looks at the most reactive fresh fuel assembly designs used at McGuire, to determine whether these assemblies meet the requirements of 10 CFR 50.68 (b) (2,3) when stored in the normally-dry NFVs.

The SFP criticality analysis evaluates the Region 1 (flux trap) and Region 2 (egg-crate) storage racks in the McGuire SFPs. These high-density racks originally contained Boraflex poison panels for reactivity holddown. However, ongoing degradation of the Boraflex material in these racks (see, e.g., Reference 1) has limited the effectiveness of this poison material in Region 1 and Region 2. To address the continuing degradation of the Boraflex panels, the McGuire criticality analysis for the SFPs considers "permanent solutions" to these issues. The permanent solutions for Region 1 and Region 2 include the following:

Region 1 Re-rack with new storage racks containing Boral poison panels. The old Region 1 Boraflex racks were removed and replaced with new racks containing Boral, which were supplied and installed by Holtec International in mid-2003. The new Region 1 racks have the same dimensions as the old racks, and thus the same storage capacity (286 cells per SFP). As Section 8.1 demonstrates, the new Region 1 racks will allow unrestricted storage of fresh McGuire reactor fuel up to 5.0 wt % U-235.

Region 2 Retain the existing egg-crate storage racks, but eliminate credit for any remaining Boraflex poison. The revised evaluation of fuel storage in the Region 2 racks takes credit for cooling time reactivity reduction (due primarily to Pu-241 decay and Gd-155 buildup following the end of reactor irradiation). The analysis also segregates storage burnup requirements by fuel assembly type, to take advantage of lower reactivity associated with certain fuel designs. In addition, the revised Region 2 evaluation takes credit for burnup in storage arrays containing empty cells, thereby allowing increased fuel storage density – such as 3 assemblies with one empty cell – within these arrays. The SFP Region 2 criticality analysis, documented in Section 8.2, employs specific 3-D calculations for all of the fuel storage configurations considered, and meets a rigorous interpretation of 10 CFR 50.68 (b) (4).

The revised criticality analyses for both Region 1 and Region 2 continue to take partial credit for soluble boron in the SFPs under normal conditions, in accordance with the criteria of 10 CFR 50.68 (b) (4).

The general goal in developing the SFP Region 2 storage requirements in this analysis is to model the fuel isotopic inventory as accurately as possible. In order to do this, it is necessary to base the 3-D burned fuel models on actual core operation data, considering axial profiles for burnup, moderator temperature history, fuel temperature history, boron concentration history, and burnable poison exposure. The methods used in quantifying the reactivity effects of these variables, as well as their uncertainties, are discussed in Section 8.2. The results of the calculations performed to generate minimum burnup requirements for Region 2 storage are also documented in that section.

2 Fuel Storage Facilities at McGuire

Figure 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. 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 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. Fresh fuel and irradiated reload fuel assemblies are transported to the reactor via the water-filled Fuel Transfer Area. Discharged fuel assemblies from the reactor are also returned to the SFP through the Fuel Transfer Area. Qualified spent fuel assemblies 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 McGuire SFPs are designed to store fresh and irradiated fuel assemblies in a wet, borated environment. The SFPs are divided into two regions: Region 1 and Region 2. The Region 1 storage racks have a flux trap design, with stainless steel rack cells. Boral poison panels are attached to the outsides of each of the Region 1 rack cell walls (with the exception of the outer perimeter cells adjacent to the SFP walls). Figure 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 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. Boraflex poison panels – which are not credited in this criticality analysis – are attached to each of the cell walls in the Region 2 racks (again with the exception of the outer perimeter cells adjacent to the SFP walls).

Tables 1 and 2 provide the McGuire NFV and SFP rack data important to the criticality modeling of these storage areas.

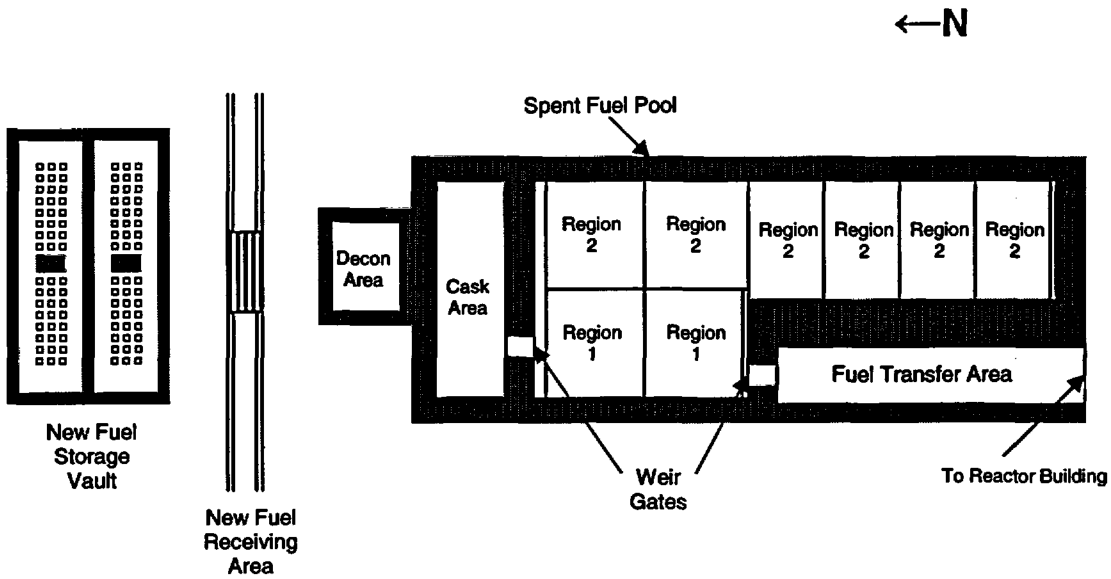


Figure 1. Overhead View of the McGuire Fuel Building (Typical of Each Unit)

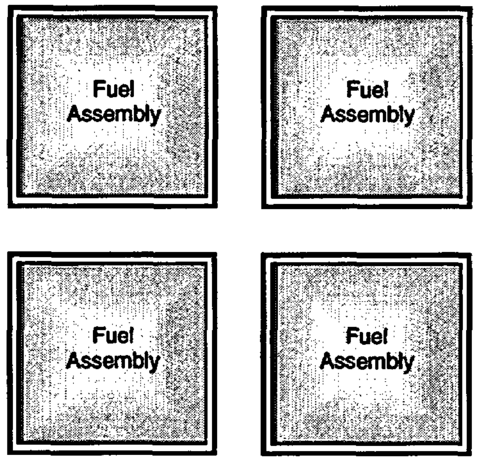


Figure 2. McGuire SFP Region 1 "Flux Trap" Storage Cell Arrangement

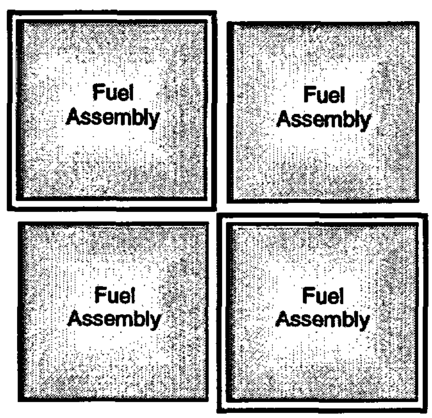


Figure 3. McGuire SFP Region 2 "Egg-Crate" Storage Cell Arrangement

Table 1. General Design Information for the McGuire NFV Storage Racks

Design Parameter	Value
# of storage locations in each NFV	96
Storage cell pitch (cm)	53.3
Storage cell ID (cm)	22.9
Concrete center dividing wall thickness (cm)	81.9

Table 2. General Design Information for the McGuire SFP Storage Racks

Design Parameter	McGuire Region 1 Value	McGuire Region 2 Value
# of storage locations in each SFP	286	1177
Storage cell pitch (cm)	26.4	23.2 (avg.)
Boral minimum B-10 Loading (g/cm^2)	0.020	--
Storage cell wall thickness (cm)	0.19	0.19
Normal SFP water temperature range (°F)	50 – 150	50 - 150
Minimum required SFP boron concentration (ppm)	2675	2675

3 Fuel Assembly Designs Considered

The following fuel types are considered for the McGuire criticality analyses:

- **MkBI** – this generic fuel type represents the old Oconee 15x15 MkB2, MkB3, and MkB4 fuel assembly designs, which used Inconel spacer grids in the active fuel area. 300 of these assemblies, which operated in the Oconee reactors up through September 1983, were transshipped to McGuire in the 1980s. Currently, 35 of the MkBI assemblies reside in Region 2 of the McGuire Unit 1 SFP, and 265 reside in Region 2 of the McGuire Unit 2 SFP.

- **W-STD** – this is the standard 17x17 Westinghouse fuel design which was used in the initial cycles (batches 1-3) of both the McGuire reactors. At that time the W-STD design had Inconel grids.

- **W-OFA** – this is the 17x17 Westinghouse "Optimized Fuel Assembly" design, which had thin rods, Zircaloy grids, and a low total uranium loading. This design was deployed for batches 4 through 9 in both McGuire units.

- **MkBW** – this is the standard 17x17 Framatome (B&W) fuel design which was modeled after the standard Westinghouse product. The MkBW design contains Zircaloy grids. This fuel type (without axial blankets) was used for batches 10 through 13 in both McGuire reactors.

- **MkBWb1** – this is the same design as the standard MkBW, but it employs **solid, 6-inch, 2.00 wt % U-235** axial blankets at the top and bottom of the active fuel zone. This fuel type was used in McGuire Unit 1, batches 14 to 16, and McGuire Unit 2, batch 14.

- **MkBWb2** – this is also the same design as the standard MkBW, but it employs **solid, 6-inch, 2.60 wt % U-235** axial blankets at the top and bottom of the active fuel zone. This fuel type was used in McGuire Unit 2, batch 15.

- **W-RFA** – this is the advanced 17x17 Westinghouse fuel design. It is similar to the MkBW assembly design, and contains Zircaloy grids, but uses **annular, 6-inch, 2.60 wt % U-235** axial blankets at the top and bottom of the active fuel zone. This fuel type has been chosen for McGuire Unit 1, batches 17 to present, and McGuire Unit 2, batches 16 to present.

The reason the MkBW fuel design has been split into non-blanketed, 2.00 wt % U-235 blanketed, and 2.60 wt % U-235 blanketed fuel types is that axial blankets have a profound effect on the axial burnup profiles of irradiated fuel assemblies. Note that it is not necessary to consider the blanketed MkBW fuel types for the SFP Region 1 criticality analysis because, as Section 8.1 shows, burnup credit will not be used for Region 1.

Likewise, the blanketed MkBW fuel types are not considered in the NFV criticality analysis, which assumes the entire fresh MkBW assembly is enriched to 5.0 wt % U-235.

Note also that since the 300 MkBI fuel assemblies that were transshipped from Oconee to McGuire are typically stored only in Region 2 of the McGuire SFPs, and because these old fuel assemblies are irradiated (with a maximum enrichment of just 3.20 wt % U-235), this fuel type is not explicitly evaluated in either the SFP Region 1 or NFV criticality analyses. However, the MkBI fuel assemblies are judged to be sufficiently low in reactivity that they may also be stored without restriction in Region 1 of the SFPs.

Pertinent design data for all of these fuel types, and the BPRAs they have contained, are provided in Tables 3 and 4. Note that the "WABA" and "Pyrex" BPRAs detailed in Table 4 have a standard ^{10}B content. The other BPRA designs that have been used – in the MkBI fuel and the MkBW fuel – can have variable ^{10}B content. For this criticality calculation, it is assumed that the MkBI BPRA contained 1.4 wt % B_4C, and the MkBW BPRA contained 4.0 wt % B_4C. These are at or very near to the highest boron concentrations that have been used in the BPRAs for these fuel types. Reference 2 shows that higher BPRA boron concentrations yield higher k_{eff} increases in the fuel assemblies that once contained those BPRAs during irradiation.

Table 4 also indicates the numbers of BPRA rodlets that have been employed in their corresponding fuel assembly types. Note that as the number of BPRA rodlets increases, so does the amount of fissile plutonium production in the irradiated fuel assembly, as the BPRA rodlets displace moderator from the fuel assembly lattice, resulting in local spectral hardening. For conservatism in the SFP Region 2 criticality analysis, it is assumed that the maximum number of BPRA rodlets (16 with the MkBI assembly, and 24 for all other fuel designs) were present for the fuel assemblies that underwent irradiation with BPRAs inserted.

Table 3. Design Data for Fuel Types Considered in the McGuire Criticality Analysis

	W-OFA	W-RFA	W-RFA blnkt	MkBI	W-STD	MkBW (b1,b2)
Avg fuel density (g/cc)	10.30	10.34	8.02	10.20	10.29	10.36
Fuel pellet OR (cm)	0.3922	0.4096	0.4096	0.4681	0.4096	0.4058
Cladding IR (cm)	0.4001	0.4178	0.4178	0.4790	0.4178	0.4140
Cladding OR (cm)	0.4572	0.4750	0.4750	0.5460	0.4750	0.4750
Pin pitch (cm)	1.26	1.26	1.26	1.442	1.26	1.26
Pin array size	17x17	17x17	17x17	15x15	17x17	17x17
Guide tube IR (cm)	0.561	0.561	0.561	0.632	0.572	0.572
Guide tube OR (cm)	0.602	0.612	0.612	0.673	0.612	0.612
Full op pressure (bars)	155	155	155	151.7	155	155
Avg power density (W/gU)	41.73	38.10	49.10	31.30	38.30	38.74
Spacer grid material	zirc	zirc	zirc	inconel	inconel	zirc
Grid linear density (g/cm)	19.2	18.1	18.1	11.3	14.8	18.7

Table 4. Design Data for Burnable Poison Rod Assemblies (BPRAs) Considered in the McGuire SFP Region 2 Criticality Analysis

	W-OFA	W-RFA	MkBI	W-STD	MkBW (b1,b2)
BPRA type	WABA	WABA	B_4C	Pyrex	B_4C
Poison pellet density (g/cc)	2.577	2.577	3.38	2.23	3.10
Poison pellet IR (cm)	0.3531	0.3531	0	0.2413	0
Poison pellet OR (cm)	0.4039	0.4039	0.432	0.4267	0.401
B_{10} conc (wt %)	1.9374	1.9374	0.2004	0.7118	0.5740
B_{11} conc (wt %)	8.6282	8.6282	0.8956	3.1702	2.5565
C conc (wt %)	2.9344	2.9344	0.304	-	0.8695
O conc (wt %)	40.720	40.720	46.416	55.218	45.192
Al conc (wt %)	45.780	45.780	52.184	-	50.808
Si conc (wt %)	-	-	-	40.900	-
# of rodlets (fingers) in BPRA	4 to 16	4 to 24	16	9 to 20	4 to 24

4 Criticality Computer Code Validation

The main neutronics codes employed in the criticality analysis are SCALE 4.4/KENO V.a and CASMO-3/SIMULATE-3. These codes are well-suited to SFP and NFV criticality applications, and have been extensively benchmarked to critical experiments and reactor operational data. KENO V.a is a 3-D Monte Carlo criticality module in the SCALE (Reference 3) package. CASMO-3 (Reference 4) is a 2-D transport code that performs fuel criticality and depletion calculations, using a 70-group cross-section library that is based on ENDF/B-IV. CASMO-3 also produces nodal macro-group cross-sections that can be used by SIMULATE-3 (Reference 5), its counterpart 3-D nodal diffusion code, for applications involving arrays of fuel assemblies with varying enrichments or burnups.

SCALE 4.4/KENO V.a is used for the evaluation of fresh fuel storage in the NFVs and in Region 1 of the McGuire SFPs, as well as verification of the Checkerboard/Empty configurations considered in the SFP Region 2 analyses. As discussed in Section 8.2, CASMO-3/SIMULATE-3 cannot properly model a true "empty cell" within a Checkerboard/Empty configuration. Instead, CASMO-3 requires some fissile material in order to generate nodal cross-section data for SIMULATE-3.

CASMO-3/SIMULATE-3 is used for all SFP Region 2 irradiated fuel cases because this is the only code system qualified by Duke to perform criticality analyses using burnup credit. Note that KENO V.a is capable of doing calculations for burned fuel, using isotopic data produced via the SAS2H module of SCALE 4.4. However, because SAS2H (which was not originally intended for fuel criticality applications) is a 1-D transport code, it is preferable to use a more explicit 2-D transport code such as CASMO-3 for irradiated fuel evaluations. 2-D calculations should more accurately model fuel assemblies that are not radially uniform, such as the fuel types described in Section 3 that contain BPRAs during initial reactor irradiation.

The following subsections discuss the benchmarking validation that has been performed for both SCALE 4.4/KENO V.a and CASMO-3/SIMULATE-3. Given the similar types of critical experiments with which these code systems have been validated, the use of these code packages is appropriate for the McGuire NFV and SFP criticality evaluations.

As an additional check on the accuracy of both code systems used in these analyses, comparisons were made between results from CASMO-3/SIMULATE-3 and SCALE 4.4/ KENO V.a for several of the same SFP Region 2 storage configurations. These comparisons are presented in Section 8.2.

4.1 Validation of Benchmark Critical Experiments for SCALE 4.4/KENO V.a

Duke Power performed a SCALE 4.4/KENO V.a benchmark analysis of critical experiments to determine calculational biases and uncertainties for both the 44-group and 238-group cross-section libraries included with the SCALE 4.4 package.

For McGuire SFP criticality applications, the SCALE 4.4/KENO V.a biases and uncertainties are based on analysis of 58 critical experiments performed by Pacific Northwest Laboratories (see References 6 to 8). The critical experiments evaluated cover a wide range of enrichment (2.35 and 4.31 wt % U-235), and include both over- and under-moderated lattices.

For the NFV criticality analyses, a subset of 41 of the 58 critical experiments described above was employed. Because the NFV analysis models fresh fuel at high (4.76 to 5.00 wt % U-235) enrichments, the 41 critical experiments were all at the highest enrichment (4.31 wt % U-235) used in the PNL experiments.

The results from the benchmark analyses indicate that the 238-group cross-section library yields the more consistent results (i.e., smaller variations in reactivity bias) across the ranges of moderation and enrichment considered. Therefore, the 238-group cross-section library is used for all the SCALE 4.4/KENO V.a computations performed in this criticality analysis.

The 41 critical experiments used for the NFV analysis yielded a benchmark calculational bias of **+0.0061** Δk (average under-prediction of k_{eff}) and an uncertainty of **±0.0071** Δk. The 58 experiments used in the benchmarking for the McGuire SFP criticality analyses resulted in a calculational bias of **+0.0064** Δk and an uncertainty of **±0.0066** Δk. These biases and uncertainties are used in determining the total bounding 95/95 system k_{eff}s for each NFV or SFP storage configuration analyzed with SCALE 4.4/KENO V.a.

4.2 Validation of Benchmark Critical Experiments for CASMO-3/SIMULATE-3

For all of the SFP Region 2 irradiated-fuel criticality evaluations, the CASMO-3/ SIMULATE-3 code set is used. All CASMO-3 calculations will be carried out with the fine-energy-group (70-group) neutron cross-section library available with that code. Duke Power has performed a benchmark analysis of 10 B&W critical experiments with CASMO-3 and SIMULATE-3. These B&W critical experiments (Reference 9) were specifically designed for reactivity benchmarking purposes. Results from these 10 B&W critical benchmark cases yielded a calculational bias of **-.0015** Δk (average over-prediction of k_{eff}) and an uncertainty of **±0.0121** Δk. Even though SIMULATE-3 tends to over-predict k_{eff}, its negative bias will be conservatively ignored. The uncertainty, however, will still be used in computing the overall 95/95 k_{eff}s for the McGuire SFP Region 2 irradiated-fuel storage configurations described in Section 5.

5 Proposed Storage Configurations for the McGuire NFVs and SFPs

Figure 4 shows the various NFV, SFP Region 1, and SFP Region 2 fuel storage configurations that are specifically evaluated in Sections 7 and 8. The minimum burnup limits for SFP Region 2 storage, in accordance with these configurations, are determined in Section 8.2. The symbols in the repeating patterns of Figure 4 correspond to the following storage types:

U – Fuel assembly qualified for **Unrestricted** storage in the NFV, SFP Region 1, or SFP Region 2

R – Fuel assembly qualified for **Restricted** storage in the SFP Region 2

F – Fuel assembly qualified for **Filler** storage in the SFP Region 2

C – Fuel assembly qualified for **Checkerboard** storage in the SFP Region 2

E – Empty storage location

Unrestricted Storage – NFV, SFP Region 1, or SFP Region 2

U	U	U	U
U	U	U	U
U	U	U	U
U	U	U	U

2/4 **Restricted/Filler** Storage – SFP Region 2

R	F	R	F
F	R	F	R
R	F	R	F
F	R	F	R

3/4 **Checkerboard/Empty** Storage – SFP Region 2

C	C	C	C
C	E	C	E
C	C	C	C
C	E	C	E

Figure 4. McGuire SFP Fuel Storage Configurations Considered in this Analysis

6 Computation of the Maximum 95/95 k_{eff}

For every fuel assembly design, fuel enrichment, cooling time, and storage region combination that is considered in the scope of the McGuire SFP and NFV criticality analyses, a nominal k_{eff} is calculated. This k_{eff} is only the base value, however. A total k_{eff} is determined by adding several pertinent reactivity biases and uncertainties, to provide an overall 95 percent probability, at a 95 percent confidence level (95/95), that the true system k_{eff} does not exceed the 95/95 k_{eff} for that particular storage condition.

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

$$k_{eff} = k_{nominal} + \sum B_x + \sqrt{\sum ks_x^2}$$

where:

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

B_x is a pertinent bias, as indicated in Table 5.

ks_x is the pertinent 95/95 independent uncertainty on $k_{nominal}$, as indicated in Table 5.

Table 5 lists the various biases and uncertainties that are considered in the McGuire NFV and SFP criticality analyses. Each of these biases and uncertainties is discussed in more detail below:

- **Benchmark Method Bias**

  This bias is determined from the benchmarking of the code system used (SCALE 4.4/KENO V.a or CASMO-3/SIMULATE-3), 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 critical experiment benchmarks for these codes are discussed in Sections 4.1 and 4.2. The bias for SCALE 4.4/KENO V.a with its 238-group cross-section library is **+0.0061** Δk for NFV applications, and **+0.0064** Δk for SFP applications. The bias for CASMO-3/ SIMULATE-3 with its 70-group cross-section library is **-0.0015** Δk. Note that negative biases are conservatively ignored in this calculation, per Reference 10.

- **Fixed Poison Self-Shielding Bias**

This reactivity penalty accounts for the slight self-shielding effects associated with the clustering of boron carbide particles in the SFP Region 1 Boral panels. The self-shielding bias was conservatively estimated for the Region 1 Boral replacement panels to be **+0.0010** Δk.

- **Cooling Time / Enrichment Interpolation Error**

Section 8.2 discusses this reactivity penalty, which accounts for the maximum difference in k_{eff} between a minimum burnup limit "estimate" using the interpolation technique specified in that section, and the "true" burnup limit that specific evaluation at that enrichment and cooling time would yield. That section determines a bounding error of **+0.00036** Δk for interpolation between the tabulated SFP Region 2 minimum burnup data points (see Tables 18 through 21).

- **Benchmark Method Uncertainty**

This uncertainty is determined from the benchmarking of the code system used (SCALE 4.4/KENO V.a or CASMO-3/SIMULATE-3), 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 are discussed in Sections 4.1 and 4.2. The method uncertainty for SCALE 4.4/KENO V.a with its 238-group cross-section library is **±0.0066** Δk for SFP applications and **±0.0071** Δk for NFV applications. The uncertainty for CASMO-3/SIMULATE-3, with its 70-group cross-section library, is **±0.01211** Δk.

- **Monte Carlo Computational Uncertainty**

For all the nominal SCALE 4.4/KENO V.a computations performed in this analysis to determine 95/95 k_{eff}s, the Monte Carlo computational uncertainty is equal to either $\mathbf{1.752*\sigma_{nominal}}$ (if 600 neutron generations are run), or $\mathbf{1.778*\sigma_{nominal}}$ (if 400 neutron generations are run). The $\sigma_{nominal}$ factor is the calculated standard deviation of $k_{nominal}$ (the nominal k_{eff} for that particular case). The 1.752 or 1.778 multiplier is the one-sided 95/95 tolerance factor for 600 or 400 neutron generations, respectively. Each of the SCALE 4.4/KENO V.a cases in the SFP Region 1 and NFV calculations counted 400 neutron generations, and the SFP Region 2 calculations used 600 neutron generations.

- **Mechanical Uncertainties**

The "mechanical uncertainty" represents the total reactivity uncertainty contributions of various independent fuel manufacturing-related and storage rack-related mechanical uncertainty factors. These factors include reactivity effects for variations in fuel enrichment, fuel pellet diameter, fuel density, cladding dimensions, storage rack dimensions and material tolerances, fixed poison panel width, fuel assembly positioning within the storage cell, etc. The following bounding total mechanical uncertainties have been determined:

NFV (no boron in full-density water): **±0.0073** Δk
NFV (no boron in optimum-density water): **±0.0079** Δk
SFP Region 1 (no boron in SFP water): **±0.00973** Δk
SFP Region 1 (310 ppm boron in SFP water): **±0.01324** Δk
SFP Region 2 (no boron in SFP water): **±0.01110** Δk
SFP Region 2 (800 ppm boron in SFP water): **±0.01247** Δk

- **Burnup Computational Uncertainty**

This burnup-related uncertainty quantifies, in a global sense, the ability of the CASMO-3/SIMULATE-3 codes to accurately determine the isotopic content, and hence k_{eff}, of a collection of irradiated assemblies in the McGuire reactors, assuming the actual average burnup of the fuel in the reactor core is the same as the average burnup of the SIMULATE model for that reactor core. Duke Power has determined a bounding McGuire CASMO-3/SIMULATE-3 burnup computational reactivity uncertainty of **±{0.00454 * BU / 50}**Δk, where BU is the average burnup of the system modeled, in GWD/MTU.

- **Burnup Measurement Uncertainty**

This uncertainty represents the reactivity penalty associated with difference between the measured burnup and the code-predicted burnup. Measured burnups, which are used for Technical Specification verification, have many sources of instrumentation error that can contribute to overall measurement inaccuracies. Section 8.2 discusses the method used to calculate a bounding measured burnup reactivity uncertainty for fuel storage in Region 2 of the McGuire SFP. The analysis of predicted and measured core follow data yields a burnup measurement uncertainty of **±0.00125** Δk.

- **Axial Profile Uncertainty**

This uncertainty represents the bounding reactivity penalty associated with differences between the k_{eff} calculated using the average "estimated" axial burnup and history profiles for a particular fuel assembly, and the k_{eff} calculated using the actual axial burnup and history profiles for that fuel assembly. Section 8.2 discusses the method used to determine average "estimated" profiles, and how to quantify the axial profile uncertainty for McGuire SFP Region 2 irradiated fuel applications. An analysis of the k_{eff} differences for a large database of "estimated" and actual axial profiles has determined a bounding axial profile uncertainty of **±0.00305** Δk.

Table 5. Pertinent 95/95 Biases and Uncertainties to be Considered in the McGuire NFV and SFP Criticality Analysis

Biases	**Include for NFV Analyses?**	**Include for SFP Region 1 Analyses?**	**Include for SFP Region 2 Analyses?**
Benchmark Method Bias	✓	✓	✓
Fixed Poison Self-Shielding Bias		✓	
Cooling Time / Enrichment Interpolation Error			✓
Uncertainties			
Benchmark Method Uncertainty	✓	✓	✓
Monte Carlo Computational Uncertainty	✓	✓	
Mechanical Uncertainties	✓	✓	✓
Burnup Computational Uncertainty			✓
Burnup Measurement Uncertainty			✓
Axial Profile Uncertainty			✓

7 McGuire New Fuel Storage Vault Criticality Analysis

To allow storage of fuel in the normally-dry environment of the NFVs, the following requirements of 10 CFR 50.68 (b) (2) and (3) must be satisfied:

> "The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. ...
>
> If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level."

The McGuire NFVs are described in Section 2. The following assumptions and simplifications are made in performing the criticality analysis of the NFVs:

1) All fuel designs that have been or are projected to be used in the McGuire reactors are evaluated. This includes the W-STD, W-OFA, MkBW, and W-RFA fuel assembly types described in Section 3.

2) A simplified 3-D axial model of the fuel assembly is employed. Only the active fuel region is modeled – the top and bottom nozzles are ignored.

3) All fuel is unirradiated. The W-OFA fuel assembly design is limited to 4.76 wt % U-235. All other fuel assemblies considered are allowed to be enriched up to 5.00 wt % U-235.

4) The fuel assemblies are stored without any location restrictions in the NFVs, in accordance with Figure 4 in Section 5.

5) No credit is taken for spacer grid material in the active fuel regions of the fuel assemblies.

6) No credit is taken for any burnable poison assemblies (BPRAs), control rods, or other neutron poisons that may be inserted in the fuel assemblies.

Using the pertinent reactivity biases and uncertainties described in Section 6, the SCALE 4.4/KENO V.a analyses for fuel storage in the NFVs yield the following maximum 95/95 k_{eff}s:

- NFV flooded with full-density unborated water: **0.9498**
- NFV flooded with optimum-moderation unborated "water": **0.9618**

Table 6 presents the various biases and uncertainties that comprise the NFV maximum 95/95 k_{eff}s.

Table 6. Maximum 95/95 k_{eff}s for Fuel Storage in the McGuire NFVs (No Boron in "Water" flooding NFV)

	NFV flooded with full-density moderator	NFV flooded with optimum-density moderator
Maximum Nominal k_{eff}	0.9329	0.9446
Biases		
Benchmark Method Bias	0.0061	0.0061
Fixed Poison Self-Shielding Bias	--	--
Cooling Time / Enrichment Interpolation Error	--	--
Uncertainties		
Benchmark Method Uncertainty	0.0071	0.0071
Monte Carlo Computational Uncertainty	0.0035	0.0032
Mechanical Uncertainties	0.0073	0.0079
Burnup Computational Uncertainty	--	--
Burnup Measurement Uncertainty	--	--
Axial Profile Uncertainty	--	--
Maximum 95/95 k_{eff}	**0.9498**	**0.9618**

8 McGuire Spent Fuel Pool Criticality Analysis

For storage of fuel in the McGuire SFPs, the following requirements of 10 CFR 50.68 (b) (4) must be satisfied:

> "... If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

In addition, for evaluations of burned fuel in SFP criticality analyses, Reference 10 provides the following general criteria:

> "A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties."

> "A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution."

The following assumptions and bases are employed for the McGuire SFP criticality evaluations:

1) Partial soluble boron credit is used in both the Region 1 and Region 2 criticality evaluations. These analyses adhere to the regulatory subcriticality criteria defined in 10 CFR 50.68 (b) (4), as well as the guidance provided in Reference 10.

2) All McGuire Region 2 criticality calculations are performed in three dimensions, with 24 axial fuel segments analyzed. The 3-D model includes top and bottom axial reflectors containing a mix of water, steel, and Zircaloy. Reference 11 supports the assumption that 24 axial fuel segments are more than sufficient to accurately capture the reactivity effects associated with axial variations in fuel burnup. Extensive historic and projected 3-D burnup, temperature, boron, and burnable poison data are employed to appropriately quantify the isotopic content of the fuel assembly designs considered.

3) McGuire Region 1 calculations are performed in 2-D, with perfect axial reflection. This is acceptable, because only fresh fuel is considered in the criticality evaluation for the Region 1 racks. It is also conservative, because it ignores axial leakage.

4) Credit is taken for the fixed Boral poison material within the new Region 1 SFP racks.

5) It is conservatively assumed that no Boraflex remains in the McGuire SFP Region 2 storage racks. In reality, some Boraflex remains in the Region 2 racks, which are not currently being replaced. This assumption is part of the "permanent solution" proposed in this amendment to the licensing basis for fuel storage in the McGuire SFPs.

6) For one of the storage configurations defined for the McGuire SFP Region 2, a water hole (empty cell) is used in one out of every four cells. This water hole cannot be modeled directly with CASMO-3, which requires at least a trace amount of fissile material in each unit cell. Thus, a low-enrichment, low-loading "water hole" is modeled with CASMO-3 to allow the overall storage configuration to be evaluated with SIMULATE-3. This is a conservative approach, as comparisons with KENO V.a in Section 8.2 show.

7) No credit is taken for any short-lived Xe-135 poisons in the fuel stored in the SFPs, consistent with Reference 10.

8) In the McGuire SFP Region 2 analysis, credit is taken for the spacer grids in each fuel assembly design considered. The standard CASMO-3 grid model, which homogenizes the grid material into the coolant surrounding the fuel assembly, is used to account for the effects of the grids. This is the same model as that used in the McGuire reactor core design and core follow calculations.

9) For accident conditions, the McGuire SFP is fully-flooded (full-density water) at the minimum McGuire SFP boron concentration as specified in the Core Operating Limits Report (2675 ppm). Per the double contingency principle (see Reference 10), it is allowable to assume that the minimum boron concentration is present in the event of an accident condition – such as a misloaded fuel assembly – in the McGuire SFP.

10) Credit for the reactivity reduction associated with fuel burnup and cooling time is employed for SFP Region 2 storage in this calculation. The reactivity reduction with cooling time is primarily due to Pu-241 decay (~14.3 yr half-life), and Gd-155 buildup (via Eu-155 decay with ~ 4.7 yr half-life).

8.1 SFP Region 1 Criticality Analysis

Section 6 documented the biases and uncertainties pertinent to the Region 1 Boral storage racks. Note that the biases and uncertainties related to fuel assembly burnup are not applicable for the Region 1 criticality analysis because the fuel storage requirements for Region 1 do not take credit for burnup.

The new Region 1 Boral storage rack design is almost identical to the previous Region 1 Boraflex rack design. The pertinent design information used as input to the Region 1 criticality analyses is provided in Tables 2 and 3.

Each of the McGuire SFP Region 1 criticality computations considers the SFP water temperature at both 32 °F and 212 °F. This ensures the maximum-reactivity condition is properly determined for every case. According to the McGuire UFSAR Section 9.1.3.1.1, SFP water temperatures will not exceed 150 °F under "normal" conditions, or 212 °F under "accident" conditions.

The normal-condition Region 1 criticality calculations are performed with no boron in the SFP water [to satisfy the 95/95 $k_{eff} < 1.0$ criterion of 10 CFR 50.68 (b) (4)], and with 310 ppm of soluble boron credit (to satisfy the 95/95 $k_{eff} < 0.95$ criterion of the same regulation).

Since the Region 1 normal-condition calculations are already performed at the conceivable extremes of SFP water temperature, the only Reference 10 accident conditions that need to be evaluated are the fuel assembly misload and fuel assembly drop events. In addition, per NUREG-0612, the criticality consequences of dropping a load heavier than a fuel assembly on the Region 1 racks are considered. All of these accident conditions are allowed to take full credit for the minimum required boron concentration in the McGuire SFPs. That minimum boron concentration, controlled though the COLR per McGuire TS 3.7.14, is currently 2675 ppm.

As discussed in Section 3, specific Region 1 criticality calculations are performed for the W-STD, W-OFA, W-RFA, and MkBW fuel types, using SCALE 4.4/KENO V.a. These cases consider fresh 5.0 wt % U-235 fuel, stored in the Unrestricted Region 1 configuration shown in Figure 4. The maximum nominal k_{eff} in unborated SFP water is computed to be 0.9631. The maximum Region 1 95/95 k_{eff} from this case, as shown in Table 7, is **0.9829**. This includes the pertinent biases and uncertainties identified in Section 6. In unborated SFP conditions, then, the maximum 95/95 k_{eff} for Region 1 storage remains below 1.0.

The SCALE 4.4/KENO V.a calculations also show that if credit is taken for 310 ppm soluble boron in the McGuire SFP, the maximum 95/95 k_{eff} for Region 1 fuel storage is reduced below 0.95 for all normal conditions.

These results demonstrate that, in the new McGuire Region 1 SFP racks, Unrestricted storage of any type of fresh McGuire reactor fuel up to 5.0 wt % U-235 meets the boron credit subcriticality criteria of 10 CFR 50.68 (b) (4) for normal storage conditions.

Three Region 1 accident conditions were identified earlier in this section – the fuel assembly misload, assembly drop, and heavy load drop events. Because any type of McGuire reactor fuel, with any enrichment and burnup, can be stored without restriction in the new Region 1 racks, there is no possibility of a misloaded assembly.

The fuel assembly drop accident, from a criticality perspective, may be considered in the same category as a single isolated fuel assembly stored in water. That is because a dropped fuel assembly dropped onto the McGuire storage racks will rest far enough above the active fuel zones of the normally stored fuel assemblies that it is effectively isolated. SCALE 4.4/KENO V.a was used to model a single, fresh, 5.0 wt % U-235 assembly of the most reactive type (W-OFA), surrounded by 30 cm of water in all directions. With only 170 ppm of boron credit taken for this "accident" condition, the largest 95/95 k_{eff} (at 32 °F) was only 0.916, well below the 0.95 subcriticality criterion.

As far as loads heavier than a fuel assembly are concerned, the largest loads that may be moved over the Region 1 area of the McGuire SFPs are the weir gates (see Figure 1). An analysis of the criticality consequences of a worst-case weir gate drop on the new Region 1 Boral racks demonstrates that even with up to 9 fuel assemblies crushed by the weir gate into an optimum-reactivity configuration, the maximum achievable 95/95 k_{eff} (0.874) is well below the 0.95 subcriticality criterion, when credit is taken for 2475 ppm boron in the SFP.

Table 7. Maximum 95/95 k_{eff} for Fuel Storage in Region 1 of the McGuire SFPs (No Boron in SFP Water)

	SFP Region 1 Storage
Maximum Nominal k_{eff}	0.9631
Biases	
Benchmark Method Bias	0.0064
Fixed Poison Self-Shielding Bias	0.0010
Cooling Time / Enrichment Interpolation Error	--
Uncertainties	
Benchmark Method Uncertainty	0.0066
Monte Carlo Computational Uncertainty	0.0038
Mechanical Uncertainties	0.0097
Burnup Computational Uncertainty	--
Burnup Measurement Uncertainty	--
Axial Profile Uncertainty	--
Maximum 95/95 k_{eff}	**0.9829**

8.2 SFP Region 2 Criticality Analysis

The first step in analyzing the McGuire SFP Region 2 racks is to assess the validity of the CASMO-3/SIMULATE-3 models that are employed to determine fuel burnup requirements for storage in Region 2. The CASMO-3/SIMULATE-3 models use the fuel, burnable poison, and SFP Region 2 rack data summarized in Tables 2, 3, and 4.

Figure 3 shows a heterogeneous (actual) representation of the Region 2 egg-crate racks. Note the heterogeneous Region 2 racks have their storage cell walls very close to the centerline between assemblies stored in neighboring "cells" (storage locations with cell walls) and "non-cells" (storage locations without cell walls). To simplify the analysis of the Region 2 racks with the nodal SIMULATE-3 code, it is desirable to use a homogeneous CASMO-3 model of the Region 2 racks. A homogeneous rack model allows all nodal interfaces between adjacent fuel assemblies to look the same. To accomplish this, the homogeneous Region 2 model for this analysis adjusts the Figure 3 cell wall location to be at the midpoint between stored assemblies, making neighboring cells identical to each other. An individual cell within the homogeneous Region 2 rack model would then have a stainless steel wall approximately half the actual cell wall thickness at its outer edge.

Table 8 shows the k_{eff} results from SCALE 4.4/KENO V.a calculations for heterogeneous and homogeneous Region 2 rack models storing different fuel types and enrichments, and also provides the k_{eff}s from the equivalent CASMO-3 homogeneous model. The results in Table 8 indicate that the homogeneous Region 2 rack model is valid, and yields essentially the same k_{eff}s as the heterogeneous model. Likewise, the CASMO-3 computations agree very well with the SCALE 4.4/KENO V.a results.

Table 8. McGuire SFP Region 2 – Fresh Fuel k_{eff} Comparisons between Homogeneous and Heterogeneous KENO V.a Models, and Homogeneous CASMO-3 Models {0 ppm boron in SFP water}

Region 2 Storage Model	SFP water temp (°F)	W-OFA 2.00 wt % U-235	W-OFA 5.00 wt % U-235	W-STD 2.00 wt % U-235	W-STD 5.00 wt % U-235
KENO V.a Heterogeneous model	150	1.1862	1.4423	1.1947	1.4411
KENO V.a Homogeneous model	150	1.1840	1.4441	1.1948	1.4405
CASMO-3 Homogeneous model	150	1.1877	1.4424	1.1960	1.4395

Section 1 mentioned that separate Region 2 fuel storage burnup limits would be determined for each fuel assembly type considered. These seven (7) fuel types, as well as the discrete BPRAs they have contained during irradiation, were described in Section 3. Note that the concept of separate Technical Specification storage limits for different fuel types does have a precedent. In Reference 12, the NRC approved separate sets of burnup requirements for storage of MkB10 and MkB11 fuel in the Oconee SFPs.

As first noted in Section 1, it is desired to determine, for each of the seven fuel types described in Section 3, the "average" axial distributions of the following five (5) reactor irradiation environment history variables that affect the isotopic composition and, hence, reactivity, of irradiated fuel:

- exposure (burnup)
- moderator temperature history
- fuel temperature history
- soluble boron history
- burnable poison (BPRA) exposure history

Fortunately, McGuire has an extensive repository of reactor core follow data available. These sources provide the complete SIMULATE-3 irradiation histories for McGuire Units 1 and 2, from their initial cycles to the present. The McGuire core follow information provides a comprehensive database of axial distributions for the W-STD, W-OFA, MkBW, MkBWb1, and MkBWb2 fuel types described in Section 3.

For the old Oconee "MkBI" fuel stored in the McGuire SFPs, representative single-cycle Oconee core data are used. Individual fuel assembly fuel temperature histories are not available in these data, so a core-average fuel temperature history profile is used as the "average" for all the MkBI fuel.

It is notable that most of the MkBI fuel assemblies in the McGuire SFP have cooling times well over 20 years (average ~ 25.6 years as of June 2003). However, the minimum burnup requirements documented for all fuel types (later in this section) are tabulated only to a maximum cooling time of 20 years, and no extrapolation is to be performed beyond 20 years. The additional uncaptured reactivity reduction for the actual MkBI fuel assemblies that have cooled significantly longer than 20 years may be considered further conservatism for the Oconee MkBI fuel model. As Reference 2 mentions, the post-irradiation reactivity of fuel assemblies continues to decrease for around 100 years, after which the reactivity begins to increase again very gradually, due to decay of longer-lived poisonous isotopes such as Am-241 and Pu-240. However, calculations show that for McGuire Region 2 storage of a spent fuel assembly, more than 500 years must elapse before the fuel assembly again achieves the reactivity it had after 20 years of cooling.

For the current W-RFA fuel that has been recently implemented at McGuire, actual high-burnup core follow data are not yet available. Therefore equilibrium-cycle projections for W-RFA fuel are used to provide the most realistic axial profile data for this fuel type. Use of the projected profiles for W-RFA fuel is judged to be conservative, due to the fact that current and projected W-RFA burnable poison exposure histories are almost all attributable to irradiation of fuel with integral burnable poisons (IFBAs). References 2 and 13 demonstrate that IFBAs have a much smaller effect on Pu isotopic production than discrete BPRAs, primarily because integral poisons do not displace moderator as the discrete BPRAs do. However, as noted in Section 3 and Table 4, the criticality analysis of the W-RFA fuel type considers all of its burnable poison exposure histories to be due to discrete, "WABA"-type BPRAs.

The final profile data histories for each of the seven fuel types considered in the SFP Region 2 evaluation are compiled and then further segregated into four different burnup "groups." Table 9 shows the burnup groups that are used for this analysis, and the ranges of average burnup data that are used to determine "average" 24-level profiles for those burnup groups. The following procedure for generating the average profiles is used:

- Collect all the 24-level axial profile data (normalized burnup, fuel temperature history, moderator temperature history, boron concentration history, and BPRA exposure history) for each of the seven fuel types. Note that a normalized burnup profile is determined by taking the actual burnup profile of an assembly and dividing each axial level by the average (2-D) burnup of that fuel assembly.

- Sort these profiles by fuel type, and then by average (2-D) burnup.

- Determine an average value, at each axial level, of each history parameter, for the fuel type being considered, using an average of the data that fall within the burnup ranges listed in the rightmost column of Table 9. Note the 2.5 GWD / MTU overlap beyond the boundaries of the final burnup groupings is used to enhance the "smoothness" of the transition between one final burnup group and the next.

To help ensure conservatism in the overall averaging of these axial profiles, with each fuel type the individual burnup "group" axial profile for BPRA exposure that yields the highest average (2-D) BPRA exposure is applied to all burnup groups for that fuel type.

The 24-level "average" axial profiles resulting from the above process are shown in Tables 10 through 16, for each of the fuel assembly designs described in Section 3. Note that the grouping of axial profiles into applications within burnup ranges helps to simplify the overall Region 2 criticality analysis, and is similar to the axial profile burnup-grouping concept documented in Reference 11.

Table 9. Grouping of 24-Level Axial Profile Data by Average Burnup Range

"Group" Average Burnup Range	Average Burnup Range of Axial Profile Data Used to Determine the "Average" History Profiles within this "Group"
< 20 GWD / MTU	0 to 22.5 GWD / MTU
20 – 30 GWD / MTU	17.5 to 32.5 GWD / MTU
30 – 40 GWD / MTU	27.5 to 42.5 GWD / MTU
> 40 GWD / MTU	37.5 to max GWD / MTU

Table 10. Average 24-Level Axial Profiles for MkBW Fuel

	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History $(K)^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/MTU)
axial level	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU > 0**
1 (top)	0.588	0.592	0.585	0.590	0.641	0.651	0.661	0.660	28.904	28.410	27.765	27.728	358.206	409.136	499.441	517.457	0.000
2	0.774	0.780	0.788	0.793	0.644	0.654	0.664	0.663	29.769	29.168	28.438	28.372	362.422	416.774	515.777	533.276	11.655
3	0.900	0.905	0.915	0.919	0.648	0.657	0.667	0.666	30.278	29.607	28.796	28.711	368.792	424.298	526.971	544.520	15.995
4	0.977	0.982	0.994	0.997	0.651	0.661	0.670	0.670	30.521	29.823	28.981	28.893	375.551	431.433	535.814	553.255	16.780
5	1.021	1.025	1.036	1.038	0.656	0.664	0.674	0.673	30.619	29.912	29.054	28.963	381.087	437.255	543.131	560.783	17.589
6	1.028	1.031	1.042	1.043	0.660	0.668	0.678	0.677	30.569	29.871	29.024	28.937	385.243	441.680	548.840	566.692	17.751
7	1.039	1.042	1.050	1.051	0.665	0.672	0.681	0.681	30.551	29.857	29.007	28.916	388.505	445.086	553.228	571.420	17.920
8	1.063	1.065	1.072	1.072	0.670	0.677	0.685	0.685	30.594	29.895	29.037	28.945	391.043	447.711	556.619	575.107	18.344
9	1.065	1.067	1.073	1.073	0.674	0.681	0.689	0.689	30.559	29.864	29.009	28.917	392.683	449.414	558.894	577.637	18.394
10	1.059	1.060	1.063	1.063	0.679	0.685	0.693	0.693	30.497	29.806	28.945	28.849	393.656	450.358	560.202	579.298	18.298
11	1.070	1.070	1.075	1.074	0.684	0.690	0.697	0.697	30.504	29.814	28.960	28.866	394.308	451.019	561.189	580.381	18.483
12	1.081	1.080	1.083	1.082	0.689	0.694	0.700	0.700	30.519	29.825	28.958	28.860	394.578	451.187	561.460	580.909	18.684
13	1.070	1.069	1.070	1.069	0.694	0.698	0.704	0.704	30.450	29.760	28.893	28.793	394.305	450.817	561.076	580.703	18.497
14	1.082	1.081	1.080	1.078	0.698	0.702	0.707	0.707	30.476	29.781	28.903	28.799	393.956	450.312	560.434	580.227	18.716
15	1.099	1.096	1.095	1.092	0.702	0.706	0.711	0.711	30.523	29.819	28.923	28.816	393.283	449.430	559.288	579.247	19.007
16	1.095	1.092	1.091	1.088	0.706	0.710	0.714	0.714	30.495	29.791	28.894	28.785	391.967	447.937	557.494	577.533	18.948
17	1.085	1.081	1.077	1.075	0.710	0.713	0.717	0.717	30.446	29.742	28.836	28.722	390.072	445.796	554.882	575.100	18.775
18	1.099	1.096	1.093	1.089	0.715	0.717	0.721	0.721	30.501	29.787	28.873	28.758	387.850	443.316	551.867	572.145	19.017
19	1.099	1.096	1.091	1.087	0.719	0.721	0.724	0.724	30.506	29.787	28.860	28.741	385.030	440.130	547.836	568.294	19.064
20	1.081	1.076	1.069	1.066	0.723	0.725	0.727	0.727	30.447	29.726	28.785	28.658	380.922	435.653	542.535	563.379	18.697
21	1.072	1.067	1.059	1.056	0.727	0.728	0.731	0.731	30.432	29.709	28.760	28.628	376.427	430.734	536.622	557.974	18.519
22	1.025	1.021	1.010	1.009	0.731	0.732	0.734	0.734	30.262	29.561	28.621	28.487	375.385	428.459	530.754	552.775	18.358
23	0.888	0.887	0.873	0.876	0.734	0.735	0.736	0.736	29.655	29.040	28.188	28.069	373.772	425.303	522.781	545.466	13.653
24	0.639	0.638	0.616	0.622	0.737	0.738	0.739	0.739	28.456	27.988	27.293	27.225	365.316	415.490	508.486	531.469	0.000

Table 11. Average 24-Level Axial Profiles for MkBWb1 Fuel

	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History (K)$^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/MTU)
axial level	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU > 0**
1 (top)	0.368	0.375	0.383	0.393	0.645	0.643	0.662	0.655	28.084	28.165	27.211	27.451	474.149	472.070	525.571	515.425	0.000
2	0.718	0.718	0.747	0.750	0.648	0.647	0.669	0.661	30.401	30.336	28.385	28.689	353.641	356.121	490.872	474.225	14.313
3	0.866	0.864	0.894	0.894	0.651	0.650	0.672	0.664	30.961	30.863	28.764	29.091	367.703	369.921	506.756	487.383	17.219
4	0.988	0.986	1.007	1.004	0.655	0.654	0.675	0.667	31.272	31.156	29.017	29.359	380.263	382.442	520.009	499.131	19.677
5	1.045	1.043	1.057	1.053	0.660	0.659	0.679	0.671	31.317	31.200	29.088	29.437	390.582	392.857	530.737	508.720	20.835
6	1.065	1.063	1.073	1.069	0.664	0.663	0.682	0.675	31.256	31.142	29.072	29.424	398.789	401.242	539.427	516.620	21.252
7	1.073	1.070	1.079	1.076	0.669	0.668	0.686	0.680	31.177	31.065	29.033	29.392	405.134	407.779	546.394	523.149	21.404
8	1.100	1.098	1.102	1.098	0.674	0.673	0.690	0.684	31.181	31.068	29.055	29.409	410.206	413.032	552.111	528.655	21.969
9	1.106	1.104	1.107	1.103	0.679	0.678	0.694	0.689	31.132	31.019	29.028	29.383	413.764	416.787	556.333	532.829	22.097
10	1.083	1.082	1.085	1.084	0.684	0.683	0.698	0.693	31.005	30.897	28.931	29.284	415.951	419.182	559.052	535.478	21.648
11	1.116	1.114	1.113	1.110	0.689	0.688	0.701	0.697	31.062	30.949	28.980	29.330	418.025	421.381	561.709	538.355	22.313
12	1.121	1.119	1.116	1.113	0.694	0.693	0.705	0.701	31.039	30.926	28.961	29.307	418.994	422.489	563.115	539.897	22.418
13	1.106	1.104	1.101	1.100	0.698	0.698	0.708	0.704	30.962	30.851	28.896	29.238	419.020	422.666	563.513	540.384	22.121
14	1.118	1.116	1.111	1.110	0.702	0.702	0.711	0.708	30.977	30.861	28.902	29.247	418.773	422.525	563.690	540.826	22.357
15	1.131	1.129	1.122	1.120	0.707	0.706	0.714	0.712	31.003	30.885	28.916	29.256	417.910	421.748	563.087	540.420	22.633
16	1.133	1.131	1.122	1.120	0.711	0.710	0.718	0.715	30.999	30.878	28.900	29.236	416.236	420.172	561.583	539.050	22.666
17	1.110	1.108	1.102	1.102	0.715	0.715	0.721	0.719	30.929	30.807	28.827	29.160	413.418	417.472	558.834	536.352	22.213
18	1.140	1.138	1.127	1.125	0.719	0.719	0.725	0.723	31.036	30.907	28.892	29.221	410.363	414.529	555.889	533.704	22.811
19	1.137	1.135	1.123	1.121	0.723	0.723	0.728	0.726	31.053	30.919	28.879	29.201	405.702	410.094	551.330	529.420	22.756
20	1.097	1.096	1.088	1.087	0.728	0.727	0.731	0.730	30.966	30.827	28.779	29.089	399.244	404.040	544.994	523.443	21.973
21	1.084	1.083	1.075	1.074	0.732	0.732	0.734	0.733	30.979	30.831	28.753	29.043	392.089	397.498	538.204	517.698	21.721
22	1.013	1.016	1.006	1.009	0.736	0.735	0.737	0.737	30.796	30.651	28.570	28.821	384.223	390.821	530.216	511.644	20.398
23	0.877	0.894	0.855	0.868	0.739	0.739	0.740	0.740	30.286	30.192	28.130	28.303	390.176	398.546	521.765	507.049	18.073
24	0.403	0.413	0.406	0.416	0.742	0.742	0.742	0.742	27.514	27.575	26.667	26.821	500.777	505.802	558.191	547.530	0.000

Table 12. Average 24-Level Axial Profiles for MkBWb2 Fuel

	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History $(K)^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/MTU)
axial level	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU > 0
1 (top)	0.399	0.429	0.433	0.442	0.659	0.647	0.663	0.650	27.871	28.493	27.384	27.772	301.837	305.805	481.979	517.706	0.000
2	0.729	0.725	0.757	0.764	0.663	0.650	0.670	0.653	29.723	30.286	28.403	29.032	229.666	229.243	480.300	507.591	16.587
3	0.875	0.863	0.899	0.902	0.666	0.654	0.672	0.657	30.185	30.728	28.782	29.467	245.157	242.180	488.905	516.379	19.753
4	0.991	0.980	1.002	1.002	0.670	0.658	0.675	0.661	30.432	30.985	29.011	29.716	258.948	254.750	495.593	525.098	22.416
5	1.043	1.033	1.046	1.045	0.674	0.662	0.679	0.665	30.443	31.007	29.066	29.774	270.478	265.633	502.154	532.708	23.647
6	1.060	1.052	1.062	1.061	0.678	0.667	0.682	0.670	30.357	30.925	29.048	29.759	280.177	274.942	508.585	539.434	24.085
7	1.066	1.059	1.072	1.071	0.682	0.672	0.686	0.675	30.253	30.824	29.024	29.754	288.466	282.871	514.903	545.629	24.244
8	1.095	1.088	1.089	1.087	0.687	0.677	0.690	0.680	30.237	30.810	29.014	29.718	295.913	289.918	518.793	550.245	24.905
9	1.101	1.095	1.096	1.094	0.691	0.682	0.694	0.685	30.170	30.744	28.989	29.696	301.772	295.479	522.697	554.150	25.066
10	1.078	1.073	1.079	1.079	0.695	0.686	0.697	0.689	30.029	30.603	28.911	29.620	306.042	299.559	525.523	556.411	24.563
11	1.110	1.106	1.102	1.099	0.699	0.691	0.701	0.694	30.069	30.643	28.932	29.623	310.127	303.407	527.370	558.905	25.306
12	1.115	1.110	1.105	1.102	0.702	0.696	0.705	0.698	30.034	30.607	28.910	29.592	312.902	306.033	528.642	560.165	25.420
13	1.100	1.096	1.093	1.092	0.706	0.700	0.708	0.702	29.950	30.522	28.854	29.531	314.551	307.580	529.311	560.413	25.092
14	1.111	1.107	1.105	1.103	0.709	0.704	0.711	0.706	29.953	30.524	28.864	29.544	315.867	308.758	530.021	561.140	25.349
15	1.125	1.121	1.114	1.111	0.713	0.708	0.715	0.710	29.971	30.543	28.864	29.528	316.495	309.240	529.235	560.579	25.664
16	1.126	1.122	1.114	1.111	0.716	0.712	0.718	0.714	29.959	30.530	28.846	29.500	316.195	308.854	527.777	559.048	25.695
17	1.104	1.100	1.098	1.097	0.720	0.716	0.721	0.717	29.884	30.449	28.795	29.455	314.692	307.325	525.750	556.372	25.172
18	1.134	1.129	1.118	1.115	0.723	0.720	0.725	0.721	29.974	30.542	28.837	29.474	313.142	305.789	522.575	553.831	25.843
19	1.132	1.127	1.115	1.113	0.727	0.724	0.728	0.725	29.980	30.543	28.821	29.445	310.015	302.926	518.450	549.809	25.796
20	1.095	1.090	1.084	1.084	0.730	0.728	0.731	0.729	29.893	30.440	28.734	29.347	305.098	298.723	513.278	544.366	24.962
21	1.086	1.082	1.071	1.071	0.734	0.732	0.734	0.733	29.899	30.431	28.689	29.267	299.488	294.560	508.137	540.291	24.762
22	1.018	1.023	1.009	1.010	0.737	0.736	0.737	0.736	29.726	30.240	28.503	29.010	292.385	290.916	503.115	537.027	23.422
23	0.873	0.915	0.879	0.881	0.740	0.739	0.740	0.740	29.223	29.795	28.125	28.458	289.684	303.042	498.840	535.987	21.011
24	0.435	0.475	0.461	0.465	0.742	0.742	0.742	0.742	27.138	27.704	26.854	27.012	361.672	371.658	510.922	546.591	0.000

Table 13. Average 24-Level Axial Profiles for W-STD Fuel

	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History $(K)^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/ MTU)
axial level	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU > 0**
1 (top)	0.459	0.536	0.568	0.568	0.659	0.656	0.655	0.655	28.344	28.737	28.622	28.622	386.217	418.381	423.989	423.989	2.445
2	0.665	0.735	0.759	0.759	0.662	0.659	0.658	0.658	29.238	29.441	29.172	29.172	400.502	429.149	434.440	434.440	3.806
3	0.841	0.899	0.914	0.914	0.666	0.663	0.662	0.662	29.933	29.966	29.579	29.579	414.495	439.269	444.133	444.133	5.033
4	0.956	0.993	1.000	1.000	0.669	0.666	0.666	0.666	30.240	30.151	29.719	29.719	427.350	447.697	451.957	451.957	5.951
5	1.024	1.041	1.041	1.041	0.673	0.671	0.670	0.670	30.304	30.137	29.706	29.706	438.712	454.474	458.061	458.061	6.613
6	1.064	1.063	1.059	1.059	0.678	0.675	0.674	0.674	30.266	30.061	29.650	29.650	448.240	459.642	462.598	462.598	7.072
7	1.087	1.074	1.068	1.068	0.682	0.679	0.678	0.678	30.206	29.991	29.603	29.603	455.845	463.334	465.766	465.766	7.378
8	1.100	1.080	1.072	1.072	0.686	0.683	0.682	0.682	30.141	29.928	29.563	29.563	461.698	465.775	467.811	467.811	7.583
9	1.108	1.082	1.075	1.075	0.690	0.688	0.687	0.687	30.080	29.874	29.526	29.526	466.036	467.226	468.974	468.974	7.721
10	1.113	1.084	1.076	1.076	0.694	0.691	0.691	0.691	30.030	29.831	29.496	29.496	469.143	467.978	469.487	469.487	7.814
11	1.116	1.086	1.078	1.078	0.697	0.695	0.695	0.695	29.992	29.799	29.470	29.470	471.239	468.220	469.500	469.500	7.878
12	1.120	1.088	1.080	1.080	0.701	0.699	0.698	0.698	29.965	29.774	29.448	29.448	472.473	468.035	469.082	469.082	7.922
13	1.123	1.090	1.082	1.082	0.704	0.702	0.702	0.702	29.950	29.755	29.428	29.428	472.927	467.492	468.287	468.287	7.952
14	1.128	1.093	1.084	1.084	0.707	0.706	0.705	0.705	29.946	29.743	29.410	29.410	472.612	466.639	467.157	467.157	7.970
15	1.132	1.096	1.087	1.087	0.710	0.709	0.708	0.708	29.956	29.738	29.396	29.396	471.436	465.433	465.664	465.664	7.974
16	1.136	1.100	1.089	1.089	0.714	0.712	0.712	0.712	29.982	29.744	29.388	29.388	469.208	463.739	463.712	463.712	7.958
17	1.139	1.103	1.092	1.092	0.717	0.716	0.715	0.715	30.022	29.762	29.386	29.386	465.671	461.359	461.154	461.154	7.909
18	1.139	1.105	1.093	1.093	0.720	0.719	0.719	0.719	30.071	29.790	29.388	29.388	460.508	458.038	457.792	457.792	7.807
19	1.131	1.103	1.092	1.092	0.723	0.722	0.722	0.722	30.128	29.833	29.402	29.402	453.363	453.482	453.384	453.384	7.614
20	1.110	1.095	1.087	1.087	0.726	0.726	0.725	0.725	30.192	29.904	29.443	29.443	443.864	447.372	447.648	447.648	7.275
21	1.058	1.063	1.061	1.061	0.730	0.729	0.729	0.729	30.175	29.933	29.458	29.458	432.110	439.614	440.454	440.454	6.709
22	0.952	0.979	0.985	0.985	0.733	0.732	0.732	0.732	29.899	29.766	29.327	29.327	418.665	430.339	431.825	431.825	5.804
23	0.763	0.811	0.828	0.828	0.735	0.735	0.735	0.735	29.185	29.246	28.926	28.926	404.116	419.688	421.792	421.792	4.451
24	0.534	0.601	0.629	0.629	0.738	0.738	0.738	0.738	28.238	28.535	28.378	28.378	389.374	408.574	411.243	411.243	2.887

Table 14. Average 24-Level Axial Profiles for MkBI Fuel

axial level	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History $(K)^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/MTU)
	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU > 0
1 (top)	0.580	0.582	0.613	0.622	0.664	0.667	0.681	0.687	27.539	27.539	27.539	27.539	424.585	465.007	548.130	551.778	9.947
2	0.768	0.770	0.792	0.801	0.667	0.670	0.683	0.689	28.376	28.376	28.376	28.376	440.484	480.869	562.852	565.534	13.073
3	0.920	0.922	0.936	0.943	0.670	0.672	0.686	0.691	28.803	28.803	28.803	28.803	454.796	495.159	576.115	577.656	15.600
4	1.001	1.002	1.011	1.014	0.674	0.676	0.688	0.694	28.954	28.954	28.954	28.954	465.511	505.896	586.122	586.355	16.950
5	1.037	1.037	1.042	1.042	0.678	0.680	0.691	0.697	28.966	28.966	28.966	28.966	473.267	513.716	593.510	592.567	17.536
6	1.051	1.051	1.053	1.051	0.682	0.683	0.694	0.699	28.949	28.949	28.949	28.949	478.696	519.253	598.905	597.210	17.768
7	1.057	1.057	1.058	1.055	0.685	0.687	0.697	0.702	28.925	28.925	28.925	28.925	482.357	523.047	602.747	600.732	17.881
8	1.060	1.060	1.061	1.058	0.689	0.691	0.700	0.704	28.903	28.903	28.903	28.903	484.730	525.543	605.285	603.127	17.940
9	1.062	1.062	1.062	1.059	0.693	0.694	0.703	0.706	28.867	28.867	28.867	28.867	486.200	527.111	606.817	604.557	17.980
10	1.064	1.064	1.063	1.060	0.696	0.697	0.705	0.709	28.845	28.845	28.845	28.845	487.060	528.046	607.690	605.367	18.010
11	1.066	1.065	1.064	1.060	0.699	0.700	0.708	0.711	28.827	28.827	28.827	28.827	487.580	528.615	608.166	605.794	18.039
12	1.067	1.067	1.064	1.061	0.702	0.704	0.710	0.713	28.827	28.827	28.827	28.827	488.004	529.057	608.411	605.959	18.070
13	1.070	1.069	1.066	1.062	0.706	0.707	0.713	0.716	28.816	28.816	28.816	28.816	488.495	529.534	608.519	606.024	18.109
14	1.073	1.072	1.067	1.064	0.709	0.710	0.715	0.718	28.791	28.791	28.791	28.791	489.124	530.122	608.506	606.176	18.162
15	1.076	1.075	1.069	1.067	0.712	0.713	0.718	0.720	28.780	28.780	28.780	28.780	489.712	530.641	608.240	606.291	18.229
16	1.081	1.080	1.072	1.071	0.715	0.716	0.721	0.723	28.786	28.786	28.786	28.786	489.830	530.666	607.447	605.921	18.306
17	1.085	1.084	1.074	1.074	0.719	0.719	0.723	0.725	28.796	28.796	28.796	28.796	489.036	529.755	605.799	604.625	18.379
18	1.087	1.086	1.075	1.075	0.722	0.722	0.726	0.727	28.791	28.791	28.791	28.791	486.885	527.457	602.923	601.982	18.423
19	1.087	1.086	1.075	1.074	0.725	0.725	0.728	0.730	28.799	28.799	28.799	28.799	482.773	523.190	598.430	597.651	18.425
20	1.085	1.083	1.072	1.071	0.728	0.729	0.731	0.732	28.807	28.807	28.807	28.807	475.987	516.284	591.934	591.385	18.382
21	1.066	1.064	1.055	1.054	0.732	0.732	0.734	0.734	28.790	28.790	28.790	28.790	466.874	507.062	583.409	583.346	18.062
22	1.002	1.002	0.995	0.995	0.735	0.735	0.736	0.737	28.657	28.657	28.657	28.657	456.846	496.849	573.186	574.103	17.011
23	0.865	0.868	0.864	0.866	0.737	0.737	0.738	0.738	28.287	28.287	28.287	28.287	447.364	487.013	561.611	564.245	14.763
24	0.689	0.696	0.696	0.700	0.740	0.740	0.740	0.740	27.485	27.485	27.485	27.485	438.633	477.832	549.660	554.375	11.892

Table 15. Average 24-Level Axial Profiles for W-OFA Fuel

	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History $(K)^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/MTU)
axial level	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU < 20**	**BU 20-30**	**BU 30-40**	**BU > 40**	**BU > 0**
1 (top)	0.562	0.587	0.603	0.615	0.651	0.657	0.655	0.657	28.161	27.869	27.866	27.723	360.929	441.101	490.089	515.734	7.702
2	0.753	0.774	0.787	0.800	0.653	0.660	0.658	0.660	28.906	28.475	28.408	28.232	368.982	450.302	500.358	526.950	10.215
3	0.908	0.923	0.931	0.937	0.656	0.662	0.661	0.664	29.483	28.936	28.805	28.576	376.549	458.770	509.448	536.382	12.160
4	0.993	1.002	1.006	1.009	0.660	0.666	0.665	0.667	29.738	29.135	28.975	28.724	382.862	465.710	516.575	543.569	13.248
5	1.031	1.036	1.039	1.040	0.664	0.669	0.669	0.671	29.800	29.178	29.010	28.761	387.930	471.248	522.115	549.070	13.737
6	1.048	1.051	1.051	1.050	0.668	0.673	0.673	0.675	29.790	29.162	28.991	28.737	391.764	475.493	526.375	553.334	13.960
7	1.058	1.059	1.059	1.057	0.672	0.677	0.677	0.679	29.774	29.143	28.971	28.718	394.474	478.551	529.469	556.449	14.091
8	1.065	1.065	1.064	1.064	0.676	0.681	0.681	0.683	29.755	29.123	28.953	28.710	396.258	480.565	531.496	558.517	14.176
9	1.069	1.068	1.066	1.066	0.680	0.685	0.685	0.687	29.734	29.100	28.927	28.682	397.302	481.721	532.679	559.819	14.236
10	1.072	1.070	1.067	1.065	0.685	0.689	0.690	0.691	29.718	29.078	28.899	28.647	397.773	482.210	533.196	560.474	14.287
11	1.076	1.072	1.069	1.068	0.689	0.693	0.694	0.695	29.708	29.063	28.880	28.630	397.796	482.168	533.152	560.502	14.337
12	1.080	1.075	1.072	1.070	0.693	0.697	0.697	0.699	29.702	29.050	28.862	28.615	397.444	481.680	532.647	560.061	14.386
13	1.083	1.077	1.072	1.068	0.698	0.701	0.701	0.702	29.698	29.038	28.840	28.582	396.765	480.808	531.776	559.330	14.432
14	1.087	1.079	1.074	1.071	0.702	0.704	0.704	0.706	29.698	29.030	28.826	28.571	395.785	479.568	530.490	558.095	14.473
15	1.090	1.082	1.077	1.075	0.705	0.708	0.708	0.709	29.701	29.025	28.816	28.566	394.468	477.933	528.797	556.428	14.514
16	1.094	1.084	1.078	1.075	0.709	0.711	0.711	0.712	29.711	29.023	28.803	28.545	392.722	475.818	526.663	554.377	14.560
17	1.098	1.086	1.079	1.074	0.713	0.715	0.715	0.716	29.727	29.024	28.791	28.521	390.461	473.122	523.961	551.773	14.604
18	1.100	1.087	1.080	1.076	0.717	0.718	0.718	0.719	29.741	29.027	28.785	28.518	387.601	469.739	520.527	548.370	14.624
19	1.099	1.087	1.079	1.074	0.720	0.722	0.722	0.722	29.760	29.032	28.778	28.506	383.849	465.415	516.238	544.210	14.621
20	1.096	1.084	1.076	1.069	0.724	0.725	0.725	0.726	29.793	29.050	28.776	28.487	378.741	459.764	510.889	539.173	14.591
21	1.074	1.065	1.059	1.054	0.728	0.729	0.729	0.729	29.768	29.026	28.747	28.457	372.486	452.932	504.393	533.062	14.325
22	0.999	0.996	0.995	0.995	0.732	0.732	0.732	0.732	29.522	28.834	28.579	28.312	365.969	445.613	496.903	526.017	13.368
23	0.837	0.845	0.853	0.860	0.735	0.735	0.735	0.735	28.892	28.341	28.158	27.934	360.101	438.516	488.554	518.022	11.352
24	0.629	0.648	0.661	0.667	0.738	0.738	0.738	0.738	28.055	27.676	27.570	27.355	354.810	431.716	479.834	509.316	8.615

Table 16. Average 24-Level Axial Profiles for W-RFA Fuel

	Normalized Axial Burnup				Moderator Temperature History (g/cc)				Fuel Temperature History (K)$^{0.5}$				Soluble Boron Concentration History (ppm)				BPRA (GWD/ MTU)
axial level	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU < 20	BU 20-30	BU 30-40	BU > 40	BU > 0
1 (top)	0.424	0.464	0.479	0.476	0.667	0.646	0.670	0.664	27.016	27.762	26.829	26.927	526.165	549.175	630.332	657.616	0.000
2	0.731	0.756	0.771	0.771	0.669	0.649	0.677	0.669	29.244	30.116	28.243	28.453	460.213	491.773	620.620	649.071	0.000
3	0.878	0.879	0.905	0.906	0.672	0.653	0.683	0.674	29.854	30.735	28.409	28.759	437.632	444.446	607.091	636.009	21.496
4	0.987	0.981	1.000	0.999	0.675	0.658	0.686	0.678	30.186	31.080	28.618	29.012	446.630	452.192	614.668	643.605	23.971
5	1.032	1.025	1.039	1.038	0.679	0.663	0.689	0.681	30.246	31.142	28.664	29.086	455.164	460.634	621.977	651.076	25.044
6	1.050	1.043	1.055	1.053	0.683	0.667	0.693	0.685	30.210	31.104	28.649	29.083	462.489	467.974	628.462	657.715	25.484
7	1.069	1.062	1.070	1.068	0.687	0.672	0.695	0.688	30.191	31.081	28.639	29.080	468.702	474.388	634.057	663.655	25.946
8	1.077	1.070	1.075	1.074	0.691	0.677	0.699	0.692	30.149	31.034	28.613	29.057	473.313	479.100	638.154	668.003	26.150
9	1.087	1.079	1.082	1.081	0.694	0.681	0.702	0.696	30.124	31.005	28.596	29.040	476.742	482.647	641.164	671.268	26.384
10	1.075	1.069	1.072	1.072	0.698	0.686	0.705	0.699	30.040	30.920	28.541	28.980	478.435	484.154	642.535	672.580	26.122
11	1.093	1.086	1.085	1.084	0.701	0.690	0.708	0.703	30.064	30.940	28.551	28.990	480.053	485.986	643.717	674.176	26.548
12	1.096	1.089	1.087	1.086	0.705	0.695	0.711	0.706	30.046	30.920	28.535	28.970	480.519	486.421	643.763	674.387	26.626
13	1.087	1.081	1.078	1.078	0.708	0.699	0.714	0.710	29.995	30.869	28.497	28.926	479.882	485.572	642.774	673.360	26.414
14	1.104	1.097	1.090	1.090	0.712	0.704	0.717	0.713	30.039	30.908	28.515	28.940	479.288	485.145	641.681	672.665	26.809
15	1.111	1.103	1.094	1.094	0.716	0.708	0.720	0.717	30.055	30.923	28.516	28.935	477.514	483.361	639.418	670.622	26.956
16	1.111	1.104	1.093	1.093	0.719	0.713	0.723	0.720	30.063	30.932	28.512	28.924	474.586	480.309	636.078	667.354	26.974
17	1.103	1.096	1.086	1.087	0.723	0.717	0.726	0.723	30.054	30.927	28.496	28.900	470.232	475.688	631.348	662.568	26.787
18	1.120	1.113	1.099	1.098	0.726	0.721	0.729	0.727	30.146	31.017	28.540	28.934	465.319	470.960	626.009	657.664	27.191
19	1.118	1.111	1.096	1.096	0.730	0.726	0.732	0.730	30.191	31.065	28.552	28.934	458.375	463.955	618.867	650.705	27.151
20	1.089	1.083	1.071	1.073	0.733	0.730	0.735	0.734	30.159	31.042	28.516	28.878	449.216	454.474	609.869	641.604	26.470
21	1.067	1.064	1.052	1.053	0.737	0.735	0.738	0.737	30.168	31.050	28.496	28.826	439.586	445.098	600.659	632.994	26.000
22	0.984	0.987	0.978	0.982	0.740	0.739	0.741	0.740	29.925	30.798	28.322	28.591	429.262	435.910	591.112	624.315	24.123
23	0.810	0.838	0.826	0.830	0.743	0.742	0.743	0.743	29.201	30.051	28.077	28.178	449.321	479.329	602.230	633.709	0.893
24	0.436	0.475	0.477	0.480	0.745	0.745	0.745	0.745	26.516	27.181	26.315	26.371	513.316	534.007	611.839	639.983	0.000

With the 24-level axial profiles now "averaged" into burnup groups, fuel to be stored in Region 2 is evaluated by the following procedure:

- Determine the fuel assembly type for the assembly to be stored in SFP.
- Determine the average (measured) 2-D burnup of the assembly being analyzed.
- With the fuel type and average burnup, obtain the "average" profiles for the five history variables discussed at the beginning of this section, by selecting the profiles from the appropriate burnup "group" for that fuel type.
- Convert the selected normalized burnup profile into an estimated "real" burnup profile by multiplying the normalized value at each axial level by the user-defined or measured 2-D assembly-average burnup.
- Build a SIMULATE-3 Region 2 model for this fuel assembly with the 24-level estimated burnup profile from the previous step, along with the profiles for the other four history variables obtained in the third step of this procedure.

Having established the average axial profiles to be used in calculating system k_{eff}s for fuel assemblies stored in McGuire Region 2, it is necessary to have all the SIMULATE-3 nodal cross-section data available to analyze all the pertinent fuel types in 3-D. Fuel irradiation cases in reactor operating conditions are first needed to determine accurate fuel isotopic content as a function of burnup.

In addition to the seven fuel types discussed in Section 3, it is necessary to generate nodal cross-sections for a pseudo-fuel type that approximates a water hole (empty storage cell). This will allow the nodal SIMULATE-3 code to model the 3/4 Checkerboard/Empty storage configuration shown in Figure 4. The CASMO-3/SIMULATE-3 codes require at least a small amount of fissile material to compute nodal cross-section data for any fuel or "water holes" used in the SFP Region 2 rack model. For this calculation the best convergence, for test cases of the 3/4 Checkerboard/Empty model, was observed with a pseudo-fuel type that used a fuel pellet diameter of just 0.20 cm, an enrichment of 0.30 wt % U-235, and a fuel density of 10.00 g/cc. Verification of the accuracy and conservatism of using this "water hole" fuel type for the 3/4 Checkerboard/Empty SIMULATE-3 model is documented in Table 17, which compares SCALE 4.4/KENO V.a and SIMULATE-3 cases that specify unirradiated 2.00 wt % U-235 Checkerboard fuel assemblies mixed with varying ratios of "water holes."

Table 17. Comparisons between SIMULATE-3 and KENO V.a for Various Checkerboard/"Empty" Storage Configurations {all cases at 150 °F, 0 ppm boron}

Region 2 Storage Configuration	SIMULATE-3 k_{eff} using model with fissile material in "water hole"	KENO V.a k_{eff} using model with fissile material in "water hole"	KENO V.a k_{eff} using model with NO fissile material in water hole
All Checkerboard Assemblies (4/4)	1.1929	1.1901	1.1905
3 Checkerboard / 1 "Empty" (3/4)	1.0569	1.0565	1.0482
2 Checkerboard / 2 "Empty" (2/4)	0.8556	0.8464	0.8185
1 Checkerboard / 3 "Empty" (1/4)	0.7507	0.7526	0.7330
All "Empty" Cells (0/4)	0.1961	0.1970	--

Once all the nodal cross-section data for the necessary fuel types have been compiled into a master fuel library, actual 3-D models of the Region 2 racks can be constructed, using an automated form of the procedure outlined above. In this manner, minimum burnup requirements are determined for each of the SFP Region 2 storage configurations shown in Figure 4, as a function of fuel type, initial enrichment, and post-irradiation cooling time. These are the 2-D fuel burnups needed to satisfy the pertinent regulatory subcriticality criteria from 10 CFR 50.68 (b) (4). Tables 18 through 21 document, for the different Region 2 storage configurations shown in Figure 4, the minimum burnup requirements calculated by this process. Note that each of the "normal-condition" McGuire SFP Region 2 criticality computations considers the SFP water temperature at both 32 °F and 150 °F, to ensure the maximum-reactivity SFP temperature condition is determined for every case.

Since the master fuel library only has specific nodal cross-section data for enrichment increments of 0.50 wt % U-235 and cooling times at 5-year intervals, these are the only data points provided in Tables 18 through 21. From an implementation standpoint, it is important to define how the end user should determine the burnup requirements for a fuel assembly that has an enrichment and/or cooling time that is outside of or in between the specifically tabulated data points.

In evaluating a fuel assembly to determine whether it meets the minimum burnup requirements for the desired storage configuration, no extrapolations are performed. That is, if a fuel assembly type has a lower maximum enrichment than the lowest tabulated enrichment for that fuel type, the lowest tabulated value is used instead of performing an extrapolation to the actual assembly enrichment. Likewise, if a fuel assembly has cooled longer than 20 years, the minimum burnup requirement for a 20-year cooling time is used, rather than an extrapolation of the burnup data beyond 20 years.

Table 18. SFP Region 2 Unrestricted Storage -- Minimum Burnup Requirements as a Function of Initial Enrichment, Cooling Time, and Fuel Assembly Type.

	Cooling Time (yrs)	Enrichment (wt % U-235) 2.00	2.50	3.00	3.50	4.00	4.50	5.00
MkBW	0	22.20	30.01	36.67	43.61	50.47	57.18	63.72
	5	19.42	26.06	32.23	38.64	44.70	50.80	56.77
	10	17.76	24.07	30.01	36.02	41.76	47.56	53.24
	15	16.74	22.90	28.95	34.45	40.01	45.64	51.15
	20	16.07	22.13	28.05	33.44	39.08	44.38	49.78
MkBWb1	0		30.01	36.05	42.52	48.57	54.24	59.74
	5		27.27	31.69	37.20	42.92	48.03	53.01
	10		25.15	30.01	34.63	40.01	44.85	49.58
	15		23.89	29.37	33.09	38.21	43.00	47.57
	20		23.09	28.43	32.09	37.13	41.78	46.26
MkBWb2	0			37.51	44.11	49.95	55.50	60.90
	5			33.02	38.34	43.97	48.98	53.87
	10			30.72	35.73	40.95	45.67	50.30
	15			30.01	34.15	38.99	43.72	48.22
	20			29.62	33.12	37.88	42.47	46.85
W-STD	0	20.02	28.59	35.83	43.37	50.67	57.75	64.63
	5	18.50	25.10	31.56	38.35	44.97	51.39	57.66
	10	17.14	23.29	30.01	35.78	42.03	48.12	54.08
	15	16.32	22.21	28.83	34.24	40.29	46.19	51.97
	20	15.79	21.51	27.96	33.24	39.16	44.94	50.61
MkBI	0	20.21	28.01	34.47	40.82			
	5	17.71	24.76	30.66	36.92			
	10	16.35	23.04	29.42	34.60			
	15	15.53	22.01	28.16	33.19			
	20	15.00	21.33	27.34	32.27			
W-OFA	0	18.55	26.08	33.28	40.01	46.83	53.25	59.71
	5	16.53	23.30	30.01	36.27	42.01	48.05	53.98
	10	15.43	21.83	28.25	34.10	40.01	45.34	50.99
	15	14.75	20.92	27.12	32.78	38.60	43.68	49.19
	20	14.32	20.33	26.40	31.91	37.62	42.62	48.02
W-RFA	0			35.46	42.04	47.88	53.50	58.94
	5			31.19	36.62	42.23	47.30	52.23
	10			30.01	34.11	39.05	44.15	48.85
	15			28.85	32.63	37.41	42.31	46.87
	20			27.93	31.67	36.35	41.12	45.57

Table 19. SFP Region 2 Restricted Storage -- Minimum Burnup Requirements as a Function of Initial Enrichment, Cooling Time, and Fuel Assembly Type.

	Cooling Time (yrs)	Enrichment (wt % U-235)						
		2.00	2.50	3.00	3.50	4.00	4.50	5.00
MkBW	0	18.26	25.32	31.73	38.39	44.73	51.04	57.20
	5	15.69	22.29	28.56	34.16	40.01	45.66	51.27
	10	14.36	20.68	26.60	31.95	37.54	42.83	48.19
	15	13.59	20.01	25.42	30.62	36.04	41.16	46.36
	20	13.10	19.29	24.66	30.01	35.05	40.07	45.16
MkBWb1	0		26.29	31.14	37.02	43.12	48.55	53.83
	5		23.07	28.91	32.89	38.20	43.29	48.07
	10		21.37	26.88	30.73	35.78	40.54	45.07
	15		20.35	25.66	30.01	34.30	38.82	43.31
	20		19.66	24.86	29.76	33.36	37.77	42.16
MkBWb2	0			32.33	38.06	44.25	49.61	54.82
	5			29.93	33.86	38.94	44.09	48.80
	10			27.89	31.65	36.48	41.24	45.69
	15			26.66	30.32	34.99	39.40	43.85
	20			25.85	30.01	34.01	38.34	42.67
W-STD	0	16.34	23.70	30.62	37.69	44.55	51.21	57.70
	5	14.55	21.04	27.88	33.62	39.84	45.90	51.80
	10	13.58	20.42	26.08	31.49	37.37	43.11	48.73
	15	12.99	20.01	24.99	30.21	35.89	41.45	46.90
	20	12.63	19.56	24.28	30.01	34.93	40.37	45.70
MkBI	0	16.13	23.62	30.01	36.37			
	5	14.26	21.08	27.43	32.71			
	10	13.27	19.71	25.72	30.74			
	15	12.67	18.87	24.67	30.01			
	20	12.30	18.33	23.99	29.52			
W-OFA	0	14.85	22.04	29.10	35.62	41.63	47.88	54.01
	5	13.38	20.01	26.26	32.15	38.13	43.42	49.04
	10	12.53	19.03	24.74	30.33	36.01	41.04	46.41
	15	12.00	18.29	23.81	29.54	34.72	40.01	44.82
	20	11.67	17.82	23.20	28.80	33.87	39.17	43.77
W-RFA	0			30.73	36.55	42.59	47.99	53.22
	5			28.49	32.49	37.54	42.73	47.47
	10			26.49	30.38	35.15	40.01	44.50
	15			25.30	30.01	33.71	38.18	42.75
	20			24.53	29.33	32.78	37.16	41.61

Table 20. SFP Region 2 Filler Storage -- Minimum Burnup Requirements as a Function of Initial Enrichment, Cooling Time, and Fuel Assembly Type.

	Cooling Time (yrs)	Enrichment (wt % U-235)						
		2.00	2.50	3.00	3.50	4.00	4.50	5.00
MkBW	0	27.34	34.90	42.58	50.08	57.40	64.52	71.46
	5	23.28	30.12	37.14	43.78	50.40	56.89	63.22
	10	21.24	28.12	34.35	40.65	46.94	53.10	59.12
	15	20.02	26.67	32.70	38.98	44.88	50.86	56.72
	20	19.50	25.73	31.65	37.77	43.55	49.42	55.18
MkBWb1	0		35.45	42.48	48.89	55.13	61.02	66.75
	5		30.69	36.65	42.58	48.29	53.61	58.82
	10		29.76	33.85	39.24	44.86	49.90	54.84
	15		28.18	32.24	37.45	42.88	47.75	52.53
	20		27.20	31.19	36.27	41.58	46.35	51.02
MkBWb2	0			44.74	50.89	56.84	62.57	68.14
	5			38.31	44.17	49.58	54.76	59.81
	10			35.48	40.89	45.99	50.89	55.68
	15			33.78	38.71	43.90	48.63	53.27
	20			32.70	37.52	42.56	47.17	51.72
W-STD	0	25.55	33.83	42.22	50.27	58.03	65.54	72.89
	5	21.90	30.01	36.78	44.01	51.04	57.87	64.53
	10	20.06	27.68	34.03	40.87	47.52	54.01	60.34
	15	20.01	26.32	32.41	39.02	45.46	51.75	57.91
	20	19.68	25.44	31.39	37.84	44.13	50.29	56.33
MkBI	0	25.14	32.48	40.01	46.65			
	5	21.76	29.20	35.28	41.37			
	10	19.98	27.03	32.82	39.13			
	15	18.94	25.73	31.34	37.45			
	20	18.27	24.89	30.40	36.39			
W-OFA	0	22.71	30.79	38.56	45.46	52.60	59.57	66.38
	5	20.01	27.42	34.25	40.55	47.08	53.46	59.71
	10	18.87	25.56	32.01	38.51	44.22	50.31	56.27
	15	18.00	24.44	30.67	36.96	42.51	48.42	54.24
	20	17.43	23.71	30.01	35.96	41.41	47.20	52.92
W-RFA	0			41.90	48.19	54.22	60.04	65.72
	5			35.92	41.96	47.39	52.66	57.81
	10			33.22	38.50	44.00	49.01	53.90
	15			31.66	36.77	42.03	46.89	51.62
	20			30.66	35.65	40.76	45.49	50.14

Table 21. SFP Region 2 Checkerboard Storage (for 3/4 Checkerboard/Empty Configuration) – Minimum Burnup Requirements as a Function of Initial Enrichment, Cooling Time, and Fuel Assembly Type.

	Cooling Time (yrs)	Enrichment (wt % U-235)						
		2.00	2.50	3.00	3.50	4.00	4.50	5.00
MkBW	0	8.12	16.50	22.94	29.15	34.67	40.43	46.20
	5	7.49	14.77	20.81	26.50	31.60	37.03	42.18
	10	7.07	13.77	19.79	24.96	30.01	34.99	40.01
	15	6.81	13.16	18.98	24.00	29.10	33.73	38.67
	20	6.64	12.78	18.45	23.37	28.35	32.90	37.75
MkBWb1	0		16.23	23.10	28.95	33.14	38.33	43.68
	5		14.53	20.89	26.24	30.19	34.95	39.57
	10		13.55	19.52	24.65	29.63	32.99	37.40
	15		12.95	18.65	23.66	28.48	31.78	36.06
	20		12.58	18.08	23.01	27.73	30.99	35.17
MkBWb2	0			23.77	29.59	33.83	38.97	44.48
	5			21.44	26.88	30.81	35.52	40.41
	10			20.03	25.29	30.01	33.52	37.89
	15			18.88	24.29	29.01	32.29	36.53
	20			18.35	23.64	28.25	31.48	35.62
W-STD	0	7.24	14.97	21.42	28.18	34.01	40.25	46.34
	5	6.79	13.83	20.01	25.78	31.08	36.80	42.40
	10	6.46	13.11	19.69	24.37	30.01	34.81	40.14
	15	6.24	12.65	18.99	23.48	29.01	33.57	38.74
	20	6.11	12.37	18.54	22.91	28.31	32.76	37.83
MkBI	0	7.67	15.06	21.81	28.16			
	5	7.20	13.82	20.01	25.83			
	10	6.86	13.05	18.92	24.44			
	15	6.66	12.57	18.23	23.56			
	20	6.53	12.26	17.78	22.99			
W-OFA	0	6.69	14.08	20.44	26.78	32.70	38.68	44.03
	5	6.32	13.06	19.21	24.72	30.15	35.68	40.65
	10	6.07	12.40	18.26	23.50	28.94	33.93	39.12
	15	5.91	11.98	17.66	22.73	28.00	32.83	37.87
	20	5.82	11.71	17.27	22.22	27.38	32.10	37.05
W-RFA	0			22.87	28.80	32.88	38.06	43.49
	5			20.56	26.06	30.01	34.66	39.27
	10			18.99	24.48	29.27	32.73	37.11
	15			18.23	23.47	28.12	31.52	35.77
	20			17.74	22.81	27.37	30.73	34.89

On the other hand, when the user wishes to determine a burnup requirement for an enrichment and/or cooling time *in between* the specific data points listed in Tables 18 through 21, it is acceptable to perform some kind of interpolation procedure. With the assertion that changes in system k_{eff} for stored fuel are proportional to changes in concentrations of fissile and poisonous isotopes, one would expect a relatively linear increase in k_{eff} with increased initial U-235 enrichment, while k_{eff} should decrease in a logarithmic fashion with cooling time, since reactivity changes following reactor irradiation are primarily attributable to the decay of fissile Pu-241 (~ 14.3-yr half life) and the buildup of poisonous Gd-155 via Eu-155 decay (~ 4.7-yr half life).

Given this expected k_{eff} behavior, the following interpolation procedure is used:

- Determine the fuel type, maximum design enrichment, and cooling time for the fuel assembly being evaluated. Locate, for this fuel type and the desired SFP Region 2 storage configuration, the minimum burnup requirements tabulated as a function of cooling time and initial enrichment.

- Make fourth-order polynomial fits, as a function of cooling time, to the five cooling time data points at each enrichment. Use these equations to find the minimum burnup requirements for the actual cooling time of the fuel assembly being evaluated, at the two enrichment data points bounding the actual enrichment of the fuel assembly.

- Perform a linear interpolation between the bounding enrichment data points (as determined in the above step) to find the minimum burnup requirement for the actual enrichment of the fuel assembly being evaluated.

Note that there is some error associated with using the interpolation process described above. That is, at interpolated values of cooling time and enrichment between those points specifically calculated with CASMO-3/SIMULATE-3, one would expect the "true" burnup requirement to be slightly greater or less than the estimated value obtained via interpolation. To quantify this error, specific cases at various "in-between" enrichments and/or cooling times are analyzed and compared with the interpolation estimates. These cases show a maximum interpolation error of **+0.00036** Δk. This interpolation error is applied as a bias in the total 95/95 k_{eff} calculations for McGuire SFP Region 2 storage, as noted in Section 6.

To determine the maximum 95/95 Region 2 k_{eff} corresponding to the minimum burnup requirements listed in Tables 18 through 21, it is necessary to evaluate the potential reactivity increases associated with variations among fuel assemblies stored within a particular configuration, as well as increases due to boundary effects between adjacent Region 2 storage configurations.

Within a particular Region 2 storage configuration, reactivity increases are examined by "mixing up" the stored fuel; that is, by randomly matching assemblies of one fuel type / enrichment / cooling time combination with another. This is important to check because many of these combinations use different axial history profiles, and so it is possible for a non-uniform radial assortment of fuel assemblies in a storage configuration to have a slightly higher system k_{eff} than a uniform array of such fuel assemblies.

Because Region 2 has three defined fuel storage configurations (see Figure 4), it is also important to examine the reactivity effects of storing one storage configuration next to another. To limit the potential reactivity increases associated with storing one type of SFP configuration next to, within, or around another, the following Region 2 storage configuration boundary restrictions are proposed:

- **Unrestricted storage** – No boundary restrictions.
- **2/4 Restricted/Filler storage** – No boundary restrictions.
- **3/4 Checkerboard/Empty storage** – Any row or column of fuel in a 3/4 Checkerboard/Empty storage configuration that borders any other storage configuration must have alternating Checkerboard fuel and empty cells. That is, it cannot be a row or column of solid Checkerboard fuel.

Using the boundary restrictions defined above, several scenarios are considered in which one of these storage configurations is adjacent to or surrounded by another. These cases are evaluated with random variations of fuel type / enrichment / cooling time within the Unrestricted, 2/4 Restricted/Filler, and 3/4 Checkerboard/Empty storage arrays.

The results of all these analyzed storage configuration scenarios indicate that, with no boron in the SFP water, the maximum Region 2 95/95 k_{eff} associated with the minimum burnup requirements listed in Tables 18 through 21 is **0.99888**. As the discussion above demonstrates, this bounding 95/95 k_{eff} accounts for the variations of fuel assembly parameters such as fuel type, enrichment, and cooling time within a particular defined configuration, and it meets the proposed boundary restrictions between different SFP Region 2 storage configurations.

Prior to confirming this maximum 95/95 k_{eff} for all of the proposed Region 2 storage configurations, it is still necessary to quantify three burnup-related uncertainties discussed in Section 6 and listed in Table 5. These are the burnup computational uncertainty, the burnup measurement uncertainty, and the axial profile uncertainty. Each of these uncertainties can be determined by examining a "global" collection of fuel assemblies, either in the McGuire operating reactor or the SFP racks, and evaluating the maximum system reactivity increases associated with variations of the pertinent parameters for these assemblies from their nominal, or assumed, values.

As noted in Section 6, the burnup computational uncertainty quantifies the accuracy of the CASMO-3 / SIMULATE-3 codes in determining the isotopic content, and hence k_{eff}, of a collection of irradiated assemblies in the McGuire reactors, assuming the actual average burnup of the fuel in the reactor core is the same as the average burnup of the

SIMULATE model for that reactor core. Determining the burnup computational uncertainty in this manner is an alternative to performing benchmarking to actual isotopic measurements of irradiated fuel.

Several cycles of McGuire reactor operational data were examined to evaluate the differences between measured and SIMULATE-3-predicted core reactivity at various times during the operating cycle. The analysis of these data yielded a burnup-dependent reactivity uncertainty. No definitive bias was observed. The burnup computational reactivity uncertainty is **$\pm\{0.00454 * BU / 50\}\Delta k$**, where BU is the average burnup of the system modeled, in GWD/MTU.

Section 6 stated that the burnup measurement uncertainty represents the reactivity penalty associated with difference between the measured burnup and the code-predicted burnup. Measured burnups are used for Technical Specification verification of, for instance, the minimum burnup requirements listed in Tables 18 through 21. However, these 2-D measured burnups have many sources of instrumentation error that will result in the measurement value being different from the "true" burnup of a specific fuel assembly.

For the purposes of this analysis, the measured burnup error for an individual fuel assembly is defined as the difference between the measured burnup and the core follow predicted burnup. In this way, differences between measured and predicted burnups can be evaluated to produce the distribution of burnup measurement errors for a database of McGuire discharge fuel assemblies, and quantify an appropriate measurement uncertainty. This is similar to the approach used in Reference 14.

Measured burnups are available from the master special nuclear material (SNM) database used for Duke Power's reactors. These burnups are obtained from in-core detector measurements taken regularly during power operation. The code-predicted burnup for each of these fuel assemblies is taken from reactor core-follow computations using the SIMULATE-3 code. As expected, the differences between predicted and measured burnup data for the database of all McGuire reactor discharge fuel (from Cycle 1 through the present) form a distribution comparable to a normal distribution. The maximum individual assembly error observed is about 4.0 %.

When an array of fuel assemblies large enough to affect system reactivity is evaluated for the McGuire SFP Region 2, and the distribution of predicted-to-measured burnup differences is accounted for, the maximum system reactivity increase observed is ~ 0.00125 Δk. The burnup measurement uncertainty to be used in the maximum 95/95 k_{eff} calculation for Region 2 is thus specified as **$\pm 0.00125\ \Delta k$**.

The axial profile uncertainty, as Section 6 mentioned, represents the bounding reactivity penalty associated with differences between the k_{eff} calculated using the average "estimated" axial burnup and history profiles for a particular fuel assembly (see Tables 10 through 16), and the k_{eff} calculated using the actual axial burnup and history profiles for that fuel assembly (from core follow computations). Earlier in this section, the discussion of the average axial history profiles noted the large database of McGuire core follow profiles available. The axial profile k_{eff} error for an individual fuel assembly in this database is defined as the difference between the k_{eff} calculated with the actual core follow axial profiles for that fuel assembly and the k_{eff} calculated with the average axial profiles (based on fuel type and burnup) for the assembly.

As with the measured burnup errors, the distribution of the axial profile k_{eff} errors in SFP Region 2 storage compares rather well with a normal distribution. The slightly negative bias observed is conservatively ignored. The largest individual assembly axial profile error calculated is +0.030 Δk. However, the bounding axial profile uncertainty is quantified in the same "global" manner as the burnup measurement uncertainty, considering a group of fuel assemblies large enough to affect system reactivity in Region 2 of the McGuire SFPs, and taking into account the distributions of axial profile k_{eff} errors within this group of assemblies. In addition, the determination of the bounding uncertainty allows for the fact that groups of four or eight fuel assemblies are often symmetrically designed for reactor operation, and these fuel assembly groups will have the same axial profile characteristics when those assemblies are ultimately discharged together from the reactor. When all of these factors are analyzed, the resulting bounding axial profile uncertainty is **±0.00305** Δk.

Finally, now that all the pertinent reactivity biases and uncertainties have been determined, the maximum calculated 95/95 k_{eff} for McGuire Region 2 storage can be confirmed for normal conditions in unborated water, in accordance with the equation presented in Section 6. Table 22 includes all of the biases and uncertainties for Region 2 storage, and shows that the maximum 95/95 k_{eff} in unborated water remains less than 1.0, meeting the requirements of 10 CFR 50.68 (b) (4).

If credit is taken for 800 ppm soluble boron in the McGuire SFPs, SIMULATE-3 calculations considering all of the SFP Region 2 normal-condition storage requirements (viz., the minimum burnup limits specified in Tables 18 through 21, and the allowable storage configurations in Figure 4) show that the maximum 95/95 k_{eff} for Region 2 fuel storage is reduced well below 0.95. It is worth mentioning that because the cases that analyze 800 ppm of soluble boron credit in Region 2 are actually performed in 3-D with irradiated fuel, the potential non-conservatisms associated with applying fresh fuel reactivity-equivalencing to burned fuel in a borated environment (see Reference 15) are not applicable here.

The only remaining task for the Region 2 criticality analysis is to evaluate potential accident conditions. Of the Reference 10 accident scenarios, only the fuel assembly misload and high abnormal water temperature (212 °F) events need to be considered. The fuel assembly drop accident was discussed in Section 8.1 for the SFP Region 1 criticality analysis. The analysis for this accident is valid for Region 2 as well, since it is not rack-dependent. In addition, the heavy load drop accident mentioned in Section 8.1 does not need to be considered for the Region 2 criticality evaluation, because the weir gate is not carried directly over Region 2, and thus an end-drop of the gate onto Region 2 – the only type of weir gate drop capable of deforming the storage racks – is not possible.

Of the two Region 2 accident conditions that need to be analyzed, the misload accident clearly is much more severe, from a criticality perspective, than an increase in SFP water temperature from 150 °F to 212 °F. The fuel assembly misload event is thus considered the bounding SFP Region 2 accident condition. Reference 10 states that for a fuel assembly misload event, it is acceptable to consider a single misload error to be the worst case, "unless there are circumstances that make multiple loading errors credible."

Reference 10 also notes that it is permissible, for accident scenarios, to take credit for the full boron concentration (2675 ppm) required as a minimum in the McGuire SFPs. The worst-case misload event in Region 2 involves placing a fresh 5.0 wt % fuel assembly in an empty cell, within the 3/4 Checkerboard/Empty configuration storage configuration shown in Figure 4. The analysis of this misload event demonstrates that 1600 ppm is sufficient to bring the SFP Region 2 system k_{eff} below the 0.95 subcriticality criterion.

Table 22. Maximum 95/95 k_{eff} for Fuel Storage in Region 2 of the McGuire SFPs (No Boron in SFP Water)

	SFP Region 2 Storage
Nominal k_{eff}	0.98126
Biases	
Benchmark Method Bias	--
Fixed Poison Self-Shielding Bias	--
Cooling Time / Enrichment Interpolation Error	0.00036
Uncertainties	
Benchmark Method Uncertainty	0.01211
Monte Carlo Computational Uncertainty	--
Mechanical Uncertainties	0.01110
Burnup Computational Uncertainty	0.00413
Burnup Measurement Uncertainty	0.00125
Axial Profile Uncertainty	0.00305
Maximum 95/95 k_{eff}	**0.99888**

9 Conclusions

The criticality analysis for the McGuire NFVs and SFPs has been performed in accordance with the requirements of 10 CFR 50.68 (b). This evaluation takes credit for Boral poison material in the new SFP Region 1 storage racks, but no longer takes credit for any remaining Boraflex in the SFP Region 2 racks. Credit has been taken for burnup and cooling time reactivity reduction in Region 2. In addition, partial credit for soluble boron is employed in the SFPs.

This analysis determined that the McGuire NFVs can store unirradiated MkBW (with or without axial blankets), W-RFA, and W-STD fuel up to 5 wt % U-235, with no location restrictions. Fresh W-OFA fuel up to 4.76 wt % U-235 may be stored in the NFVs with no location restrictions.

The SFP criticality evaluation demonstrated that the Region 1 Boral racks can store fresh McGuire reactor fuel of any type, up to 5 wt % U-235, with no restrictions. The existing irradiated Oconee "MkBI" assemblies in the McGuire SFPs may also be stored in the Region 1 racks without restriction.

Minimum burnup requirements for SFP Region 2 storage were developed for seven different fuel types, as a function of initial enrichment and post-irradiation cooling time. These burnup requirements were specified for three defined Region 2 storage configurations: Unrestricted, 2/4 Restricted/Filler, and 3/4 Checkerboard/Empty.

The following restrictions for adjacent storage of different fuel configurations in Region 2 of the SFPs were determined in this analysis:

- **Unrestricted storage** – No boundary restrictions.
- **2/4 Restricted/Filler storage** – No boundary restrictions.
- **3/4 Checkerboard/Empty storage** – Any row or column of fuel in a 3/4 Checkerboard/Empty storage configuration that borders any other storage configuration must have alternating Checkerboard fuel and empty cells. That is, it cannot be a row or column of solid Checkerboard fuel.

The maximum 95/95 k_{eff}s for the NFV analysis were calculated to be **0.9498** (NFV flooded with full-density unborated water) and **0.9618** (NFV flooded with optimum-moderation unborated "water"). These results meet the requirements of 10 CFR 50.68 (b) (2,3).

For the SFP criticality analyses, the maximum 95/95 k_{eff}s with no boron in the SFP were calculated to be **0.9829** (SFP Region 1 storage) and **0.9989** (SFP Region 2 storage). These results meet the no-boron 95/95 k_{eff} criterion in 10 CFR 50.68 (b) (4).

The SFP criticality analysis confirmed that 800 ppm of partial soluble boron credit is sufficient to maintain the maximum 95/95 k_{eff} less than 0.95 for all normal conditions. The current minimum boron concentration required in the McGuire SFPs (2675 ppm) is adequate to maintain the maximum 95/95 k_{eff} below 0.95 for all credible accident conditions in the McGuire SFPs.

10 References

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4. **CASMO-3 – A Fuel Assembly Burnup Program**, STUDSVIK/NFA-89/3, June 1993.

5. **SIMULATE-3 – Advanced Three-Dimensional Two-Group Reactor Analysis Code**, STUDSVIK/SOA-95/15, October 1995.

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10. **"Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants"**, Memorandum from L. Kopp (NRC) to T. Collins (NRC), U.S. Nuclear Regulatory Commission, August 19, 1998.

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15. **NUREG/CR-6683, A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage**, J. Wagner and C. Parks, Oak Ridge National Laboratory, prepared for the U.S. Nuclear Regulatory Commission, September 2000.

ATTACHMENT 7

MARKUPS TO THE MCGUIRE TECHNICAL SPECIFICATION BASES

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.

DELETE

~~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 $k_{eff} \leq$

5.00

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 ~~850~~ 400 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~~ post-irradiation cooling time 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, and 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 ~~850~~ 400 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. 1600
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 800
or equal to 0.95 when flooded with water borated to 850 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 850 ppm is not a credible event.

900

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.

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

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.

INSERT A

~~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.~~

~~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.~~

To address this degradation, ~~The~~ the McGuire spent fuel storage racks (both Regions) 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~~ 5.00 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~~ at least 400 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 For Region 2 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 as ~~s~~pecified by LCO 3.7.15. INSERT B

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~~ decay time is not ~~satisfied.~~ met.

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
800 the 850 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 1600 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 800 or equal to 0.95 when flooded with water borated to 850 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
800 concentration to 850 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

INSERT C

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 850 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 850 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 850 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 850 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.

MB 5014
MB 5015

2. Issuance of Amendments, McGuire Nuclear Station, Units 1 and 2 (TAC NOS. MA9730 and MA9731), November 27, 2000.

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

February 4, 2003

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).

10 CFR 50.68, "Criticality Accident Requirements".

INSERT A

The McGuire Region 1 spent fuel storage racks are composed of individual cells made of stainless steel. These racks utilize Boral, a boron carbide aluminum cermet, as the neutron absorber material. The cells within a module are interconnected at six locations along the length of the cell using spacer plates to form an integral structure. Depending on the criticality requirements, some cells have a Boral wrapper on all four sides, some on three sides and some on two sides. The Region 1 racks will store the most reactive fuel (up to 5.00 weight percent Uranium-235 enrichment) without any burnup limitations.

Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physical stable and chemically inert form. The 1100 alloy aluminium is a lightweight metal with high tensile strength, which is protected from corrosion by a highly resistant oxide film. Boron carbide and aluminum are chemically compatible and ideally suited for long-term use in a spent fuel pool environment.

The McGuire Region 2 spent fuel storage racks contain Boraflex neutron-absorbing panels that surround each storage cell on all four sides (except for peripheral sides). 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".

INSERT B

No credit is taken for the Boraflex neutron absorber panels. The criticality analysis performed for Region 1 shows that the acceptance criteria for criticality is met for unrestricted storage without credit for burnup or plutonium decay of fuel assemblies with enrichments up to a maximum nominal value of 5.00 weight percent Uranium-235.

The storage criteria for fuel stored in Region 2 of the spent fuel pool is based upon criticality analysis that was performed in accordance with the criteria of 10 CFR 50.68(b). The fuel storage requirements are defined as a function of enrichment, burnup, cooling time and fuel type. The following are the fuel types considered in the criticality analyses:

MkBI – This generic fuel type represents the old Oconee 15x15 MkB2, MkB3, and MkB4 fuel assembly designs, which used Inconel spacer grids in the active fuel area. 300 of these assemblies, which operated in the Oconee reactors, were transshipped to McGuire.

W-STD – This is the standard 17x17 Westinghouse fuel design which was used in the initial cycles (batches 1-3) of both the McGuire reactors. At that time the W-STD design had Inconel grids.

W-OFA – This is the 17x17 Westinghouse "Optimized Fuel Assembly" design, which had thin rods, Zircaloy grids, and a low total uranium loading. This design was deployed for batches 4 through 9 in both McGuire units.

MkBW – This is the standard 17x17 Framatome (B&W) fuel design which was modeled after the standard Westinghouse product. The MkBW design contains Zircaloy grids. This fuel type (without axial blankets) was used for batches 10 through 13 in both McGuire reactors.

MkBWb1 – This is the same design as the standard MkBW, but it employs solid, 6-inch, 2.00 wt % U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type was used in McGuire Unit 1, batches 14 to 16, and McGuire Unit 2, batch 14.

MkBWb2 – – This is also the same design as the standard MkBW, but it employs solid, 6-inch, 2.60 wt % U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type was used in McGuire Unit 2, batch 15.

W-RFA – This is the advanced 17x17 Westinghouse fuel design. It is similar to the MkBW assembly design, and contains Zircaloy grids, but uses annular, 6-inch, 2.60 wt % U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type has been chosen for McGuire Unit 1, batches 17 to present, and McGuire Unit 2, batches 16 to present.

INSERT C

a

Unrestricted storage of fuel assemblies within Region 1 of the spent fuel pool is allowed provided that the maximum nominal Uranium-235 enrichment is equal to or less than 5.00 weight percent. This ensures the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95, assuming the pool is flooded with water borated to 800 ppm.

b

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool, which have accumulated burnup greater than or equal to the minimum qualified burnups and which have decayed greater than or equal to the minimum qualified cooling time in Table 3.7.15-1 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 800 ppm. Fuel assemblies not meeting the criteria of Table 3.7.15-1 shall be stored in accordance with Figure 3.7.15-1 per the initial enrichment, burnup and decay time criteria specified by Tables 3.7.15-2 and 3.7.15-3 for restricted/filler storage configuration. Another acceptable storage configuration is described by Figure 3.7.15-2 for fuel assemblies that satisfy the initial enrichment, burnup and decay time criteria specified in Table 3.7.15-4 for Checkerboard storage.

ATTACHMENT 8

REVISED TECHNICAL SPECIFICATION BASES

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

In the two region poison fuel storage rack (References. 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 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 (Reference 4) and NRC guidance (Reference. 5). The spent fuel storage racks are analyzed to allow storage of fuel assemblies with enrichments up to a maximum nominal value of 5.00 weight percent Uranium-235 while maintaining $k_{eff} \leq$ 0.95 including uncertainties, tolerances, biases, and credit for soluble boron. Soluble boron credit is used to offset 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 800 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 regulatory subcriticality requirements are met for fuel assembly storage within an allowable storage configuration, when the criteria for fuel assembly type, initial enrichment, burnup, and post-irradiation cooling time, as specified in LCO 3.7.15, are satisfied.

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

APPLICABLE SAFETY ANALYSES (continued)

the third is the misloading of a fuel assembly into a location which the restrictions on location, enrichment, and burnup are 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 (Reference. 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 800 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 determined that a spent fuel pool boron concentration of 1600 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 800 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 800 ppm is not a credible event.

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36 (Reference. 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. 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. MB5014 and MB5015), February 4, 2003.

3. 10 CFR 50.68, "Criticality Accident Requirements"

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.

REFERENCES (continued)

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.

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 Region 1 spent fuel storage racks are composed of individual cells made of stainless steel. These racks utilize Boral, a boron carbide aluminum cermet, as the neutron absorber material. The cells within a module are interconnected at six locations along the length of the cell using spacer plates to form an integral structure. Depending on the criticality requirements, some cells have a Boral wrapper on all four sides, some on three sides and some on two sides. The Region 1 racks will store the most reactive fuel (up to 5.00 weight percent Uranium-235 enrichment) without any burnup limitations.

Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physical stable and chemically inert form. The 1100 alloy aluminum is a lightweight metal with high tensile strength, which is protected from corrosion by a highly resistant oxide film. Boron carbide and aluminum are chemically compatible and ideally suited for long-term use in a spent fuel pool environment.

The McGuire Region 2 spent fuel storage racks contain Boraflex neutron-absorbing panels that surround each storage cell on all four sides (except for peripheral sides). 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, the McGuire spent fuel storage racks (both Regions) 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 5.00 weight percent Uranium-235 while maintaining $k_{eff} \leq 0.95$ including uncertainties, BACKGROUND (continued)

tolerances, biases, and credit for soluble boron. Soluble boron credit is used to offset 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 at least 800 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 for Region 2 shows that the regulatory subcriticality requirements are met for fuel assembly storage within an allowable storage configuration, when the criteria for fuel assembly type, initial enrichment, burnup, and post-irradiation cooling time, as specified in LCO 3.7.15, are satisfied. No credit is taken for the Boraflex neutron absorber panels in Region 2. The criticality analysis performed for Region 1 shows that the acceptance criteria for subcriticality are met for unrestricted storage of unirradiated fuel assemblies with enrichments up to a maximum nominal value of 5.00 weight percent Uranium-235.

The storage criteria for fuel stored in Region 2 of the spent fuel pool is based upon criticality analysis that was performed in accordance with the criteria of 10 CFR 50.68(b). The fuel storage requirements are defined as a function of enrichment, burnup, cooling time and fuel type. The following are the fuel types considered in the criticality analyses:

MkBI – This generic fuel type represents the old Oconee 15x15 MkB2, MkB3, and MkB4 fuel assembly designs, which used Inconel spacer grids in the active fuel area. 300 of these assemblies, which operated in the Oconee reactors, were transshipped to McGuire.

W-STD – This is the standard 17x17 Westinghouse fuel design which was used in the initial cycles (batches 1-3) of both the McGuire reactors. At that time the W-STD design had Inconel grids.

W-OFA – This is the 17x17 Westinghouse "Optimized Fuel Assembly" design, which had thin rods, Zircaloy grids, and a low total uranium loading. This design was deployed for batches 4 through 9 in both McGuire units.

MkBW – This is the standard 17x17 Framatome (B&W) fuel design which was modeled after the standard Westinghouse product. The MkBW design contains Zircaloy grids. This fuel type (without axial blankets) was used for batches 10 through 13 in both McGuire reactors.

MkBWb1 – This is the same design as the standard MkBW, but it employs solid, 6-inch, 2.00 wt % U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type was used in McGuire Unit 1, batches 14 to 16, and McGuire Unit 2, batch 14.

MkBWb2 – – This is also the same design as the standard MkBW, but it employs solid, 6-inch, 2.60 wt % U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type was used in McGuire Unit 2, batch 15.

W-RFA – This is the advanced 17x17 Westinghouse fuel design. It is similar to the MkBW assembly design, and contains Zircaloy grids, but uses annular, 6-inch, 2.60 wt % U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type has been chosen for McGuire Unit 1, batches 17 to present, and McGuire Unit 2, batches 16 to present."

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 decay time is not met.

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 800 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 1600 ppm was

APPLICABLE SAFETY ANALYSES (continued)

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 800 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 800 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

Unrestricted storage of fuel assemblies within Region 1 of the spent fuel pool is allowed provided that the maximum nominal Uranium-235 enrichment is equal to or less than 5.00 weight percent. This ensures the k_{eff} of the spent fuel pool will always remain $\leq$ 0.95, assuming the pool is flooded with water borated to 800 ppm.

b

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool, which have accumulated burnup greater than or equal to the minimum qualified burnups and which have decayed greater than or equal to the minimum qualified cooling time in Table 3.7.15-1 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 800 ppm. Fuel assemblies not meeting the criteria of Table 3.7.15-1 mayl be stored in accordance with Figure 3.7.15-1 per the initial enrichment, burnup and decay time criteria specified by Tables 3.7.15-2 and 3.7.15-3 for restricted/filler storage configuration. Another acceptable storage configuration is described by Figure 3.7.15-2 for fuel assemblies that satisfy the initial enrichment, burnup and decay time criteria specified in Table 3.7.15-4 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 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. MB5014 and MB5015), February 4, 2003.

3. 10 CFR 50.68, "Criticality Accident Requirements".

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).