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

November 6, 1995

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: ISSUANCE OF AMENDMENTS - McGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M89744 AND M89745)

Dear Mr. McMeekin:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 159 to Facility Operating License NPF-9 and Amendment No. 141 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated June 13, 1994, as supplemented by letters dated August 15, 1994, March 23, April 18, July 21, and September 22, 1995.

The amendments revise the TS to increase the initial fuel enrichment limit and establish new loading patterns for new and irradiated fuel in the spent fuel pool to accommodate this increase.

The March 23, 1995, supplement, which provided additional information that modified the June 13, 1994, application's no significant hazards consideration determination, also revises the TS to (1) change the surveillance requirement for boron concentration in the spent fuel pool (SFP), (2) remove the option to use alternate storage configurations in the SFP and replace it with footnotes, (3) add information contained in the Bases to the footnotes, and (4) change the Bases to discuss the option to use specific analyses on alternate fuel.

The April 18, July 21, and September 22, 1995, letters provided additional information and clarification that did not modify the scope of the June 13, 1994 application and the initial proposed no significant hazards consideration determination.

It should be noted that an issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," October 7, 1993, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will provide those requirements to you under separate cover.

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

Sincerely,

Original signed by:

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 159 to NPF-9
2. Amendment No. 141 to NPF-17
3. Safety Evaluation

cc w/encl: See next page

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

Sincerely,

Victor Nerses

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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1. Amendment No. 159 to NPF-9
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cc w/encl: See next page

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 159
License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated June 13, 1994, as supplemented August 15, 1994, March 23, April 18, July 21, and September 22, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

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

    Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 159 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
  Changes

Date of Issuance: November 6, 1995

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Facility Operating License No. NPF-17 filed by the Duke Power Company (licensee) dated June 13, 1994, as supplemented August 15, 1994, March 23, April 18, July 21, and September 22, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 141 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 6, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 159

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
XV	XV
XIX	XIX
XX	XX
3/4 9-16	3/4 9-16
3/4 9-17	3/4 9-17
-	3/4 9-18
-	3/4 9-19
-	3/4 9-20
-	3/4 9-21
-	3/4 9-22
-	3/4 9-23
-	3/4 9-24
-	3/4 9-25
B 3/4 9-3	B 3/4 9-3
-	B 3/4 9-4
-	B 3/4 9-5
5-7	5-7
5-7a	5-7a

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

SECTION PAGE

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DESIGN FEATURES

3/4.9.12 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.12 The boron concentration in the spent fuel pool shall be within the limit specified in the COLR.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within its limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 Verify at least once per 7 days that the spent fuel pool boron concentration is within its limit.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 Storage of new or irradiated fuel is limited to the configurations described in this specification.

a. New or irradiated fuel may be stored in Region 1 of the Spent Fuel Pool in accordance with these limits:

   1) Unrestricted storage of fuel meeting the criteria of Table 3.9-1; or

   2) Restricted storage in accordance with Figure 3.9-1, of fuel which does not meet the criteria of Table 3.9-1.

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 3.9-3; or

   2) Restricted storage in accordance with Figure 3.9-2, of fuel which meets the criteria of Table 3.9-4; or

   3) Checkerboard storage in accordance with Figure 3.9-3 of fuel which does not meet the criteria of Table 3.9-4.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the noncomplying fuel assembly to the correct location.

b. The provisions of Specification 3.0.3 are not applicable.

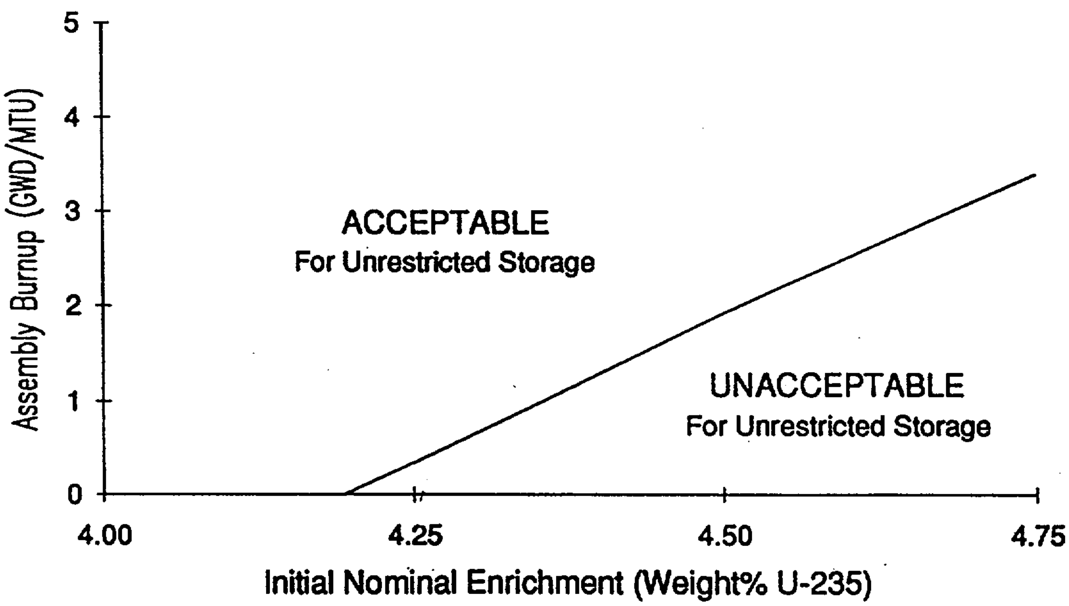
SURVEILLANCE REQUIREMENTS

4.9.13 Prior to storing a fuel assembly in the spent fuel storage pool, verify by administrative means the initial enrichment and burnup of the fuel assembly are in accordance with Specification 3.9.13.

Table 3.9-1

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Region 1 Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
4.19(or less)	0
4.20	0.04
4.50	1.92
4.75	3.40



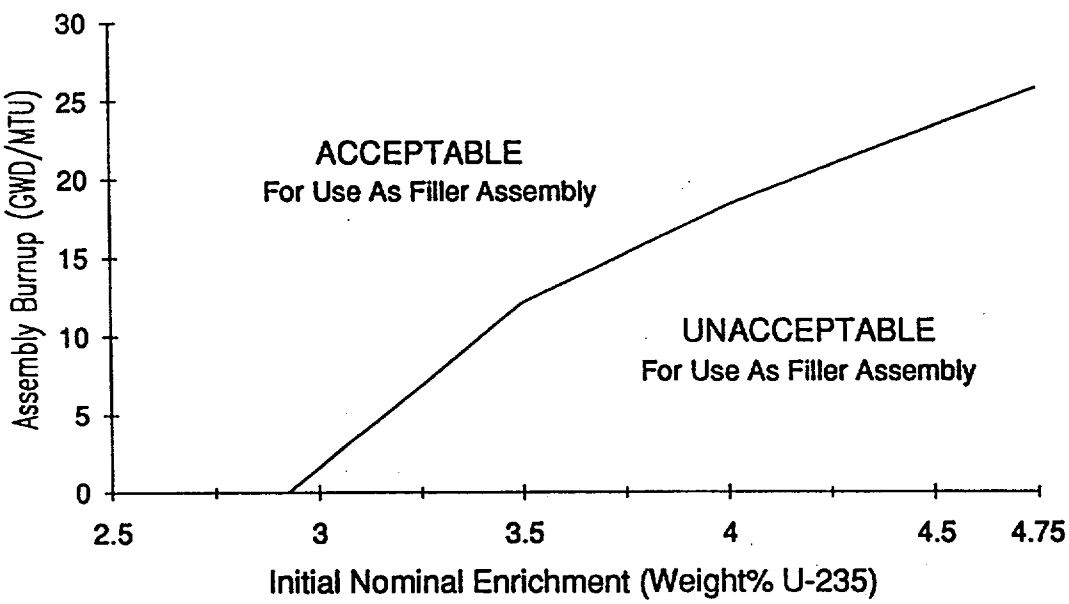
Fuel which differs from those designs used to determine the requirements of Table 3.9-1 may be qualified for Unrestricted Region 1 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Likewise, previously unanalyzed fuel up to 4.75 weight% U-235 may be qualified for Restricted Region 1 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-2

Minimum Qualifying Burnup Versus Initial Enrichment for Region 1 Filler Assemblies

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.92(or less)	0
3.00	1.57
3.50	13.30
4.00	18.32
4.50	23.36
4.75	25.84

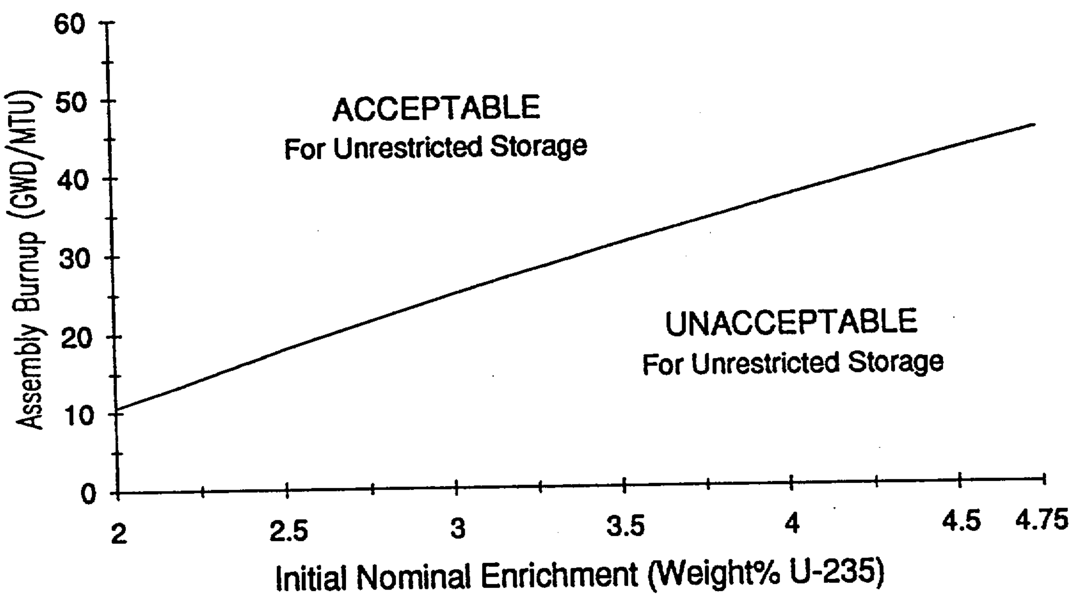


Fuel which differs from those designs used to determine the requirements of Table 3.9-2 may be qualified for use as a Region 1 Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-3

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Region 2 Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00(or less)	10.54
2.50	17.96
3.00	24.64
3.50	30.86
4.00	36.75
4.50	42.38
4.75	45.10

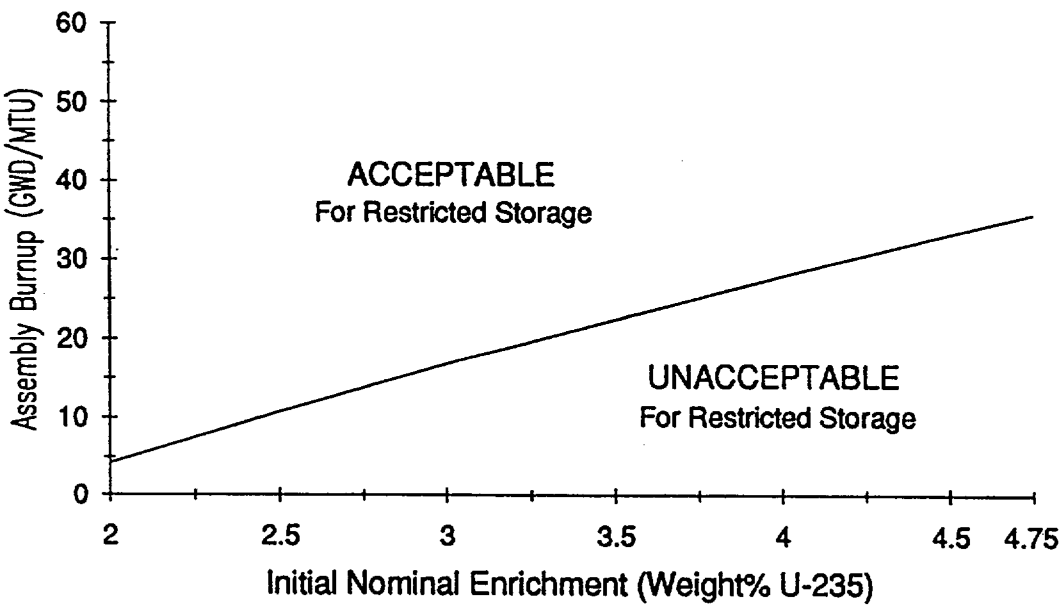


Fuel which differs from those designs used to determine the requirements of Table 3.9-3 may be qualified for Unrestricted Region 2 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-4

Minimum Qualifying Burnup Versus Initial Enrichment for Restricted Region 2 Storage with Fillers

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00(or less)	4.22
2.50	10.75
3.00	16.80
3.50	22.41
4.00	27.92
4.50	33.14
4.75	35.65

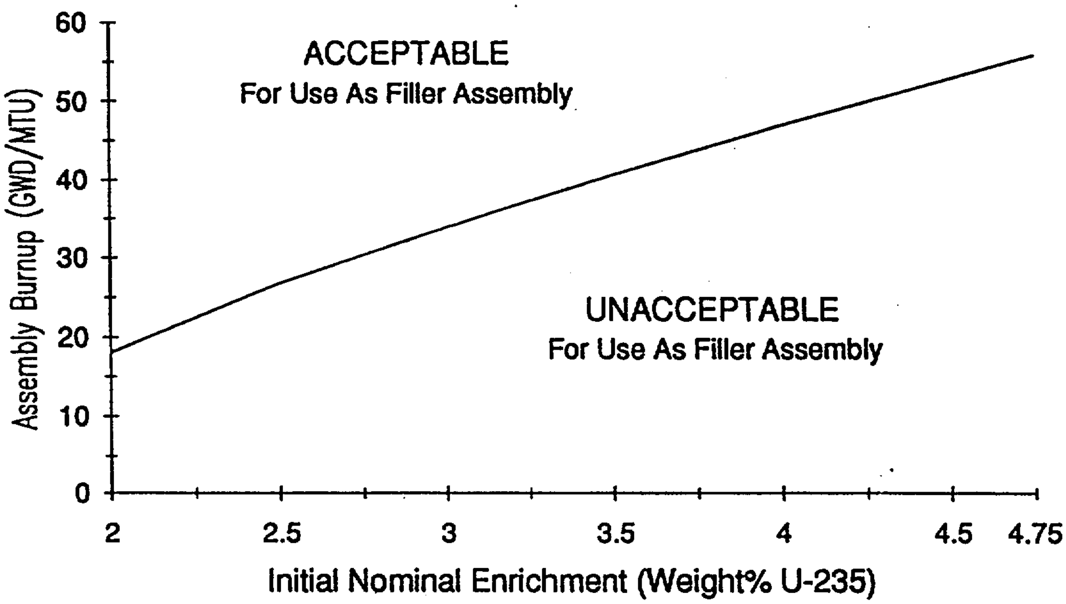


Fuel which differs from those designs used to determine the requirements of Table 3.9-4 may be qualified for Restricted Region 2 Storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-5

Minimum Qualifying Burnup Versus Initial Enrichment for Region 2 Filler Assemblies

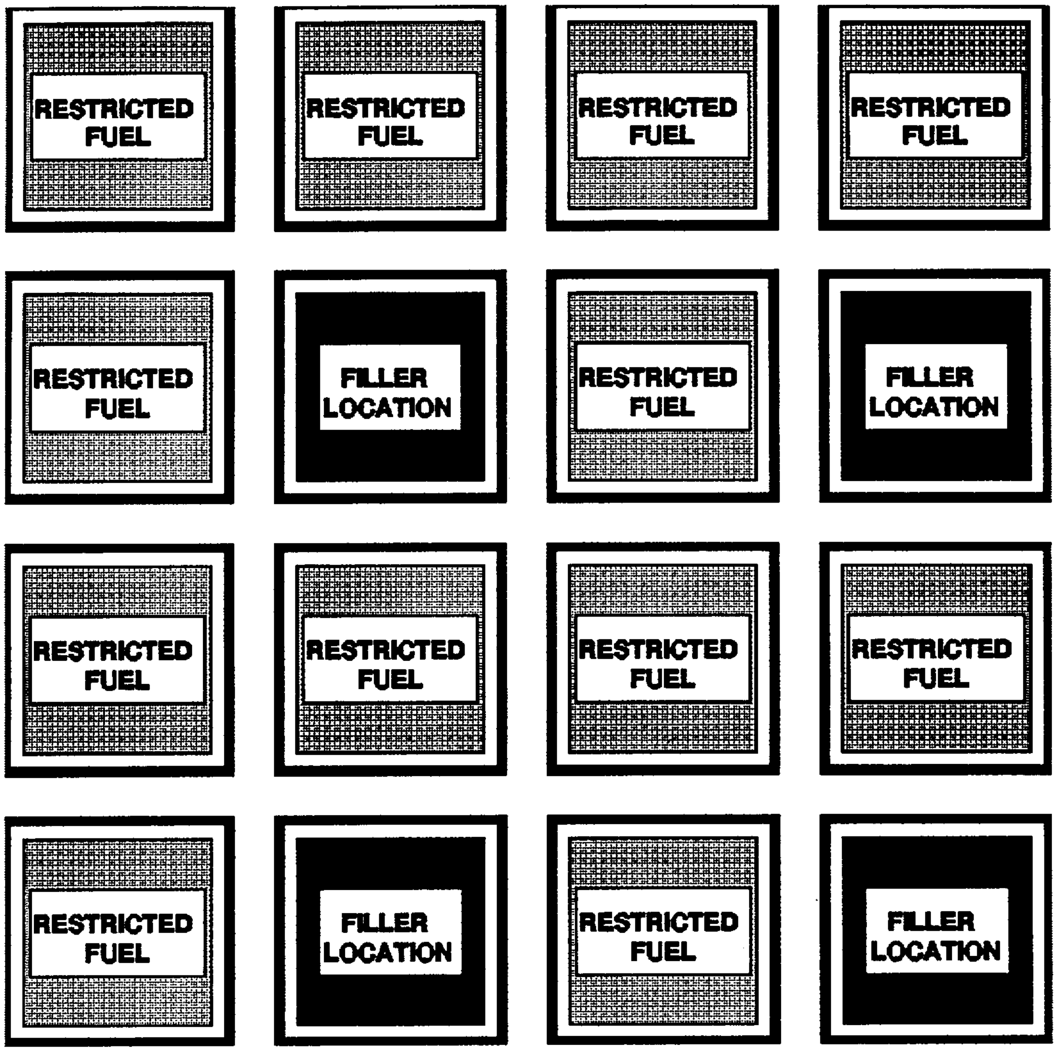
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00(or less)	18.03
2.50	26.71
3.00	33.79
3.50	40.56
4.00	46.83
4.50	52.86
4.75	55.78



Fuel which differs from those designs used to determine the requirements of Table 3.9-5 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 or equal to 0.95.

Figure 3.9-1

Required 3 out of 4 Loading Pattern
for Restricted Region 1 Storage



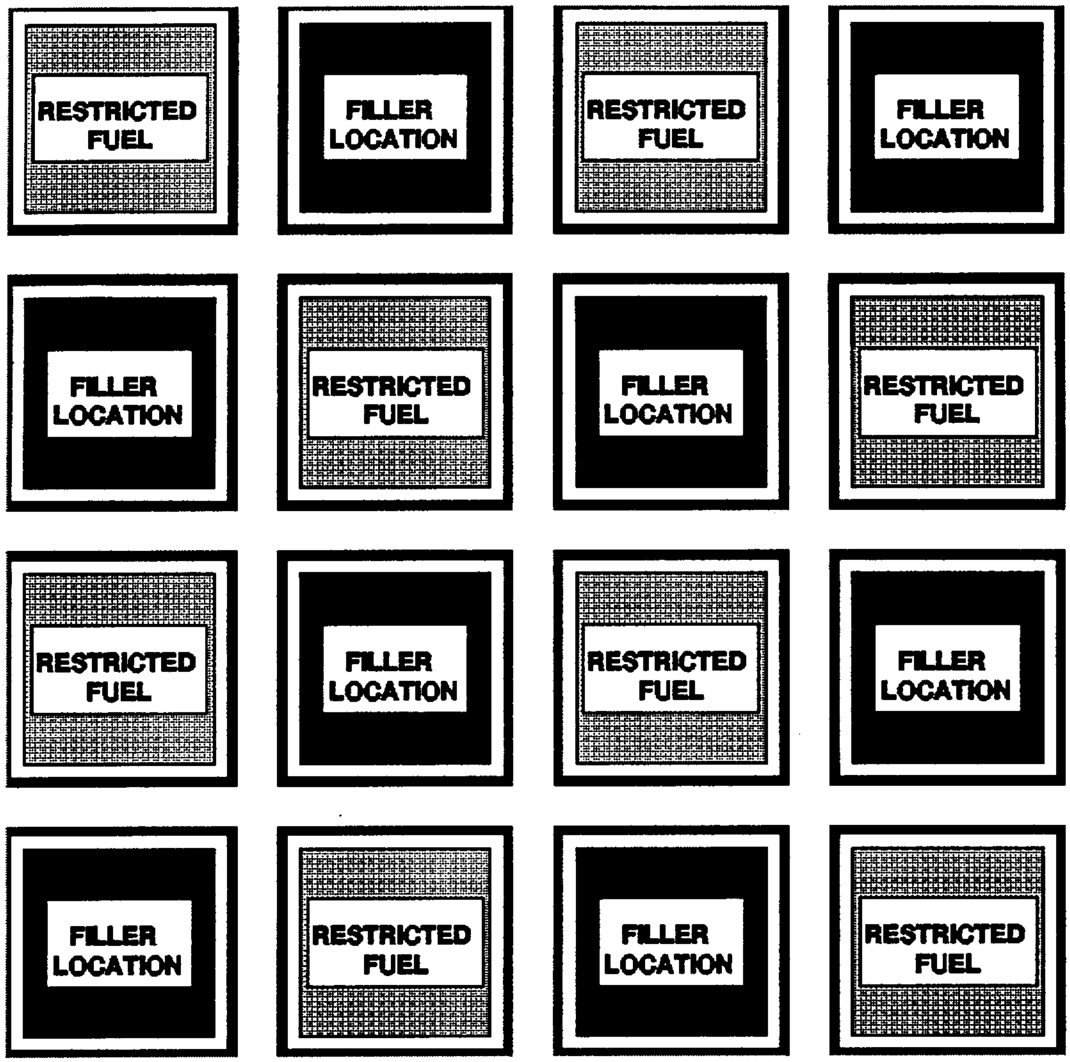
Restricted Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.9-1. (Fuel which does meet the requirements of Table 3.9-1, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.9-2, or an empty cell.

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

Figure 3.9-2

Required 2 out of 4 Loading Pattern
for Restricted Region 2 Storage



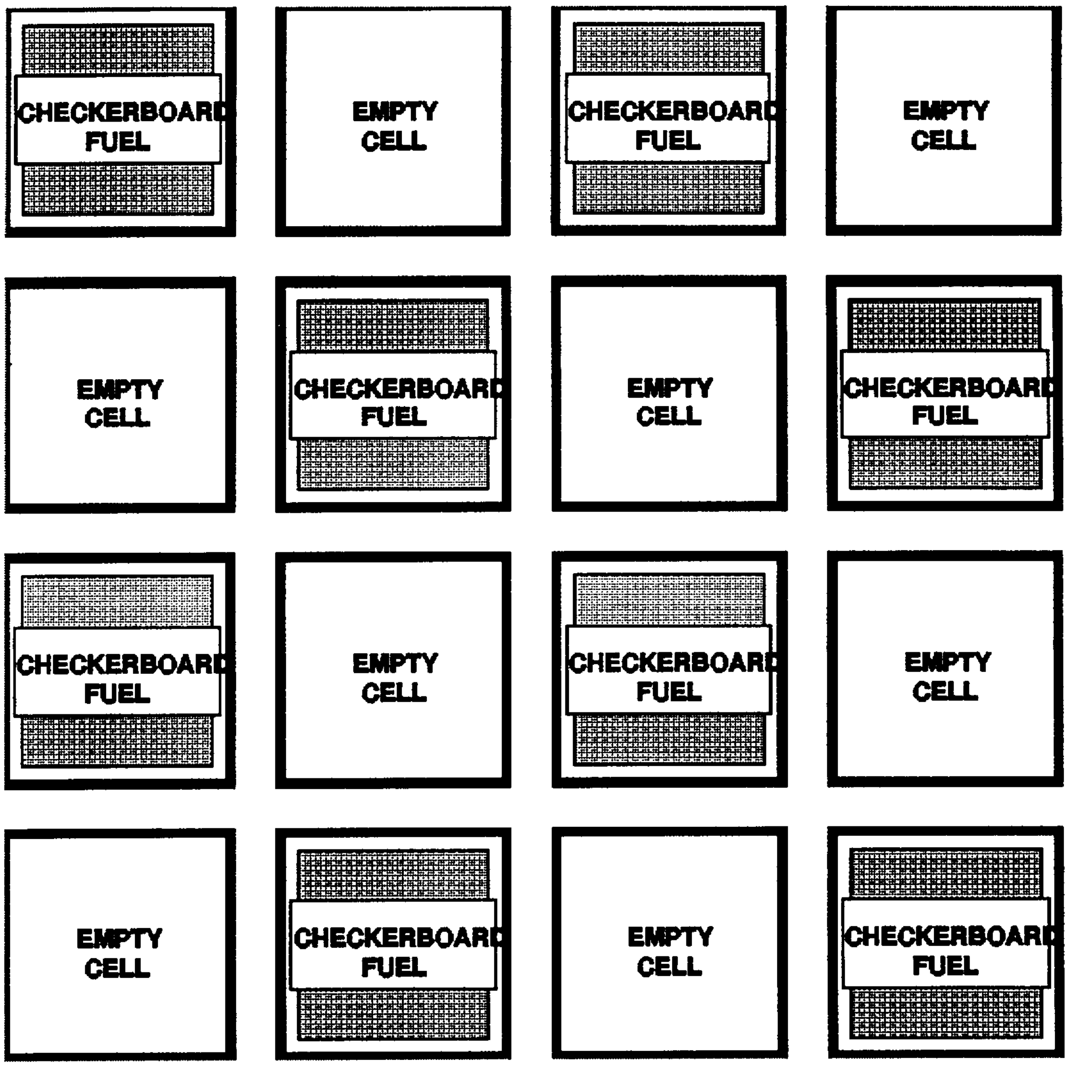
Restricted Fuel: Fuel which meets the minimum burnup requirements of Table 3.9-4, or non-fuel components, or an empty location.

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.9-5, or an empty cell.

Boundary Condition: No restrictions on boundary assemblies.

Figure 3.9-3

Required 2 out of 4 Loading Pattern
for Checkerboard Region 2 Storage



Checkerboard Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.9-4. (Fuel which does meet the requirements of Table 3.9-4, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Boundary Condition: At least two opposite sides shall be bounded by either an empty row of cells, or a spent fuel pool wall.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

The limitations on the Fuel Handling Ventilation Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing. The methyl iodide penetration test criteria for the carbon samples have been made more restrictive than required for the assumed iodine removal in the accident analysis because the humidity to be seen by the charcoal adsorbers may be greater than 70% under normal operating conditions.

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE

The requirements for spent fuel pool boron concentration specified in Specification 3.9.12 ensure that a minimum boron concentration is maintained in the pool. The requirements for spent fuel assembly storage specified in Specification 3.9.13 ensure that the pool remains subcritical. The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence the design of the spent fuel storage racks is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the spent fuel pool fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 1 or Region 2. This could increase the reactivity of the spent fuel pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water.

Tables 3.9-1 through 3.9-5 allow for specific criticality analyses for fuel which does not meet the requirements for storage defined in these tables. These analyses would require using NRC approved methodology to ensure that $k_{eff} \leq 0.95$ with a 95 percent probability at a 95 percent confidence level as described in Section 9.1 of the FSAR. This option is intended to be used for fuel not included in previous criticality analyses. Fuel storage is still limited to the configurations defined in TS 3.9-13. The use of specific analyses for qualification of previously unanalyzed fuel includes, but is not

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE (Continued)

limited to, fuel assembly designs not previously analyzed which may be as a result of new fuel designs or fuel shipments from another facility. Another more likely, and expected use of this specific analysis provision would be to analyze movement and storage of individual fuel pins as a result of reconstitution activities.

In verifying the design criteria of $k_{eff} \leq 0.95$, the criticality analysis assumed the most conservative conditions, i.e. fuel of the maximum permissible reactivity for a given configuration. Since the data presented in Specification 3.9.13.a and 3.9.13.b represents the maximum reactivity requirements for acceptable storage, substitutions of less reactive components would also meet the $k_{eff} \leq 0.95$ criteria. Hence, any non-fuel component may be placed in a designated empty cell location. Likewise, an empty cell, or a non-fuel component may be substituted for any designated fuel assembly location. These, or other substitutions which will decrease the reactivity of a particular storage cell will only decrease the overall reactivity of the spent fuel storage pool.

If both restricted and unrestricted storage is used in Region 1, an additional criteria has been imposed to ensure that the boundary row between these two configurations would not locally increase the reactivity above the required limit. Likewise if checkerboard storage is used in Region 2, an additional restriction has been imposed on the boundaries of the checkerboard storage region to ensure that the reactivity would not increase above the required limit. No other restrictions on region interfaces are necessary.

For storage in Region 2 requiring loading pattern restrictions, (per Specifications 3.9.13.b.2 or 3.9.13.b.3) fuel may be stored in either the "cell" or "non-cell" locations. "Cell" locations are the areas inside the fabricated storage cells and "non-cell" locations are the storage locations created by arranging the fabricated storage cells in a checkerboard configuration. Hence the "non-cell" locations are the areas defined by the outside walls of the 4 adjacent "cell" locations.

The action statement applicable to fuel storage in the spent fuel pool requires that 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. Prior to the resumption of fuel movement, the requirements of the LCOs must be met. This requires restoring the soluble boron concentration and the correct fuel storage configuration to within the corresponding limits. This does not preclude movement of a fuel assembly to a safe position.

The surveillance requirements ensure that the requirements of the two LCOs are satisfied, namely boron concentration and fuel placement. The boron concentration in the spent fuel pool is verified to be greater than or equal to the minimum limit. The fuel assemblies are verified to meet the subcriticality requirement by meeting either the initial enrichment and burnup

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE (Continued)

requirements of Table 3.9-1 through 3.9-5, or by using NRC approved methodology to ensure that $k_{eff} \leq 0.95$. By meeting either of these requirements, the analyzed accidents are fully addressed.

The fuel storage requirements and restrictions discussed here and applied in section 3.9.13 are based on a maximum allowable fuel enrichment of 4.75 weight% U-235. The enrichments listed in Tables 3.9-1 through 3.9-5 are nominal enrichments and include uncertainties to account for the tolerance on the as built enrichment. Hence the as built enrichments may exceed the enrichments listed in the tables by up to 0.05 weight% U-235. Qualifying burnups for enrichments not listed in the tables may be linearly interpolated between the enrichments provided. This is because the reactivity of an assembly varies linearly for small ranges of enrichment.

REFERENCES

1. "Regulatory Guide 1.13: Spent Fuel Storage Facility Design Basis", U.S. Nuclear Regulatory Commission, Office of Standards Development, Revision 1, December 1976.

2. "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations", American Nuclear Society, ANSI N210-1976/ANS-57.2, April 1976.

3. FSAR, Section 9.1.

4. 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).

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,040 $\pm$ 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

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

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) $k_{eff} \leq 0.98$ if moderated by aqueous foam as described in Section 9.1 of the FSAR; and

3) A nominal 21" center to center distance between fuel assemblies placed in the storage racks.

5.6 FUEL STORAGE (Continued)

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft. 7 in.

CAPACITY

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 159 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated June 13, 1994, as supplemented by letters dated August 15, 1994, and March 23, April 18, July 21, and September 22, 1995, Duke Power Company (the licensee) submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the TS to increase the initial fuel enrichment limit from a nominal value of 4.0 to 5.0 weight percent Uranium-235, and establish new loading patterns for new and irradiated fuel in the spent fuel pool to accommodate this increase.

The March 23, 1995, supplement provided additional information that modified the June 13, 1994, application's no significant hazards consideration determination, and revised the TS to (1) change the surveillance requirement for boron concentration in the spent fuel pool (SFP), (2) remove the option to use alternate storage configurations in the SFP and replace it with footnotes, (3) add information contained in the Bases to the footnotes, and (4) change the Bases to discuss the option to use specific analyses on alternate fuel. The April 18, July 21, and September 22, 1995, letters provided additional clarifying information that did not change the scope of the June 13, 1994, application and the initial proposed no significant hazards consideration determination.

In the two later submittals, dated July 21 and September 22, 1995, the licensee provided further detailed information relating to cooling and heat transfer in the spent fuel pool.

The staff's evaluation of the proposed changes follows.

2.0 EVALUATION

2.1 Criticality Aspects

The fresh fuel storage racks are used for temporary storage of unirradiated reload fuel and are built on 21-inch centers. The spent fuel pool consists of two regions. Region 1 is designed for storage of fresh or partially irradiated fuel. The stainless steel cells are spaced on a 10.4-inch center-to-center distance and utilize the neutron absorbing material Boraflex with a nominal 0.02 gm/cm^2 boron-10 loading attached to each exterior cell wall.

Region 1 has a storage capacity of 286 cells. The stainless steel cells in Region 2 are assembled in a checkerboard pattern, producing a honeycomb structure of cell and non-cell locations. The cell center-to-center pitch in Region 2 is 9.125 inches and these cells also utilize Boraflex having a lower boron-10 areal density (0.006 gm/cm^2) than that used in Region 1. Region 2 has a nominal capacity of 1177 cells.

The analysis of the reactivity effects of fuel storage in the new and spent fuel storage racks was performed with the SCALE system of computer codes with the three-dimensional multi-group Monte Carlo computer code, KENO Va. Neutron cross sections were generated by the NITAWL and BONAMI codes using the 123 Group GMTH library. Since the KENO Va code package does not have depletion capability, burnup analyses were performed with the CASMO-3 /SIMULATE-3 methodology. CASMO-3 is an integral transport theory code and SIMULATE-3 is a nodal diffusion theory code. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the McGuire fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber reactivity worth. The intercomparison between two independent methods of analysis (KENO Va and CASMO-3/SIMULATE-3) also provides an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO Va reactivity calculations, a minimum of 90,000 neutron histories were accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO Va reactivity calculations. Based on the above, the staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the McGuire storage racks with a high degree of confidence.

The fresh fuel storage racks are normally maintained in a dry condition, i.e., the new fuel is stored in air. However, the NRC criteria for new fuel storage require that the effective multiplication factor, k_{eff}, of the storage rack be no greater than 0.95 if accidentally flooded by pure water and no greater than 0.98 if accidentally moderated by low density hydrogenous material (optimum moderation). The new fuel storage racks were analyzed for 4.75 w/o U-235 enriched fuel for the full density flooding scenario and for the optimum moderation scenario. The calculated worst-case k_{eff} for a full rack of the Mark BW fuel design, which is the most reactive of the three fuel types which exist at McGuire, as a function of moderator density was 0.9499. Appropriate biases and uncertainties due to the calculational method and material tolerances were included at the 95/95 probability/confidence level. This meets the staff acceptance criteria of 0.95 for full density water flooding and 0.98 for optimum moderation conditions and is, therefore, acceptable.

For spent fuel storage, the staff's acceptance criterion is that k_{eff} of the storage racks be no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level, when fully flooded by unborated water. The licensee has used the acceptable methodology discussed above to demonstrate that fuel assemblies with nominal enrichments up to 4.19 w/o U-235 can be stored in every cell of the Region 1 spent fuel storage racks. To enable the storage of depleted fuel assemblies initially enriched to greater than 4.19 w/o U-235, the concept of burnup credit reactivity equivalencing was used. This is predicated upon the reactivity decrease associated with fuel depletion and has been previously accepted by the staff for spent fuel storage analysis. For burnup credit, a series of reactivity calculations are performed to generate a set of initial enrichment-fuel assembly discharge burnup ordered pairs which all yield an equivalent k_{eff} less than 0.95 when stored in the spent fuel storage racks. This is shown in Table 3.9-1 in which a fresh 4.19 w/o enriched fuel assembly yields the same rack reactivity as an initially enriched 4.75 w/o assembly depleted to 3.4 GWD/MTU. The curve shown in the Table includes biases due to methodology, Boraflex width shrinkage and B_4C self-shielding, as well as an uncertainty due to Boraflex axial shrinkage, a 95/95 methodology uncertainty, and a mechanical uncertainty due to manufacturing tolerances. In addition, a bias and uncertainty associated with fuel burnup was also included. The staff has reviewed the assumptions made in determining these biases and uncertainties, including the results obtained from blackness testing performed on representative Boraflex panels at McGuire in 1991, and concludes that they are appropriately conservative.

New or irradiated assemblies with initial enrichments up to 4.75 w/o U-235 which do not meet the requirements for unrestricted storage in Region 1, but which require temporary placement in Region 1 for operational requirements, must be placed in a restricted loading pattern. Reactivity analyses for these assemblies, arranged in a three-out-of-four storage configuration, were performed using the previously discussed methods. Acceptable fuel assemblies which qualify for storage in the fourth storage location of each three-out-of-four pattern are shown in Table 3.9-2 and are referred to as filler assemblies. These filler assemblies were also determined from minimum burnup versus initial enrichment calculations as described above.

Region 2 of the McGuire spent fuel pools has similarly been analyzed for storage of fuel initially enriched to a maximum 4.75 w/o U-235. Table 3.9-3 shows the minimum burnup required for unrestricted storage in this region, ranging up to 45.10 GWD/MTU for an assembly initially enriched to 4.75 w/o. Table 3.9-4 shows the minimum burnup requirements for restricted storage, i.e., a two-out-of-four configuration, with the remaining two locations either vacant or containing filler assemblies. The minimum qualifying burnup versus initial enrichment for Region 2 filler assemblies are given in Table 3.9-5. Fuel assemblies which do not meet any of these burnup requirements may be placed in a checkerboard configuration in Region 2, but each adjacent cell must remain empty.

These configurations have all been analyzed using the acceptable reactivity methods described previously and meet the NRC acceptance criterion of k_{eff} no greater than 0.95, including all appropriate uncertainties at the 95/95 probability/confidence level. The results are, therefore, acceptable.

Tables 3.9-1 through 3.9-5 contain a footnote which would allow for specific criticality analyses for fuel which differs from those designs used to determine the requirements for storage defined in these tables. This would allow storage of fuel from another facility or storage of individual fuel rods as a result of fuel assembly reconstitution. A similar specification was previously approved for the Oconee Nuclear Station. These analyses would require using the NRC approved methodology described above to ensure that k_{eff} does not exceed 0.95 at a 95/95 probability / confidence level and fuel storage would still be limited to the configurations defined in TS 3.9-13. At the staff's request, the Bases for TS 3.9-13 were revised to include additional discussion which reflects the intended use of this provision.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the spent fuel racks. However, it is possible to postulate events, such as the misloading of an assembly with a burnup and enrichment combination outside of the acceptable requirement, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of boron in the pool water required during storage of fuel by TS 3.9.12 since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. Therefore, the staff criterion of k_{eff} no greater than 0.95 for any postulated accident is met.

The following Technical Specification changes have been proposed as a result of the requested enrichment increase. The staff finds these changes, and the associated Bases changes, acceptable.

(1) TS 3/4.9.12 is being replaced by a new TS 3/4.9.12, which relocates the required minimum spent fuel pool boron concentration in the Core Operating Limits Report (COLR), and by TS 3/4.9.13, which specifies the new required fuel storage requirements given in Tables 3.9-1 through 3.9-5 and Figures 3.9-1 through 3.9-3 based on the reactivity analyses evaluated and approved above. The relocation of the minimum spent fuel pool boron concentration to the COLR has previously been approved by the NRC in a separate licensing action. Based on the NRC staff's recommendation, the licensee has also reduced the soluble boron surveillance interval from 31 days to 7 days, added a statement to Tables 3.9-1 through 3.9-5 indicating that specific analyses may be performed to qualify fuel assemblies for storage using NRC approved methodology, and added additional discussion in the Bases to allow for specific criticality analyses for special situations without requiring additional TS changes, as discussed above.

(2) TS 5.6.1 is being changed to reflect the NRC criticality acceptance criteria for both the new fuel storage racks and the spent fuel storage racks.

(3) TS 5.6.3 is being changed to eliminate reference to a maximum initial fuel enrichment limit since this limit is being relocated to the Tables associated with TS 3.9.13.

2.2 Spent Fuel Pool Cooling and Transfer Aspects

In addition to the initial submittal dated June 13, 1994, the licensee provided their responses to questions raised by the staff relating to cooling and heat transfer in the spent fuel pool.

The spent fuel pool cooling system (SFPCS) consists of two incompletely separated trains. Each train consists of a pump, a heat exchanger (HX) and associated piping and valves. The trains are separated from the pump suction line in the SFP to some distance downstream of the HX in each train, at which point they combine into a pipeline common to both trains, for discharge in to the SFP. Cooling water for each HX supplied by the Component Cooling System (CCS).

A prefilter, deionizer and post-filter serves as part of the SFPCS to remove corrosion and fission products from the water in the spent fuel pool. A portion of the SFP water being moved by a SFPCS pump may be diverted through the prefilter, deionizer and post-filter for the corrosion/fission products removal process. The volume of water corresponding, approximately, to one full SFP can be circulated thought the corrosion/fission product removal process. The volume of water corresponding, approximately, to one full SFP can be circulated through the corrosion/fission product removal process each day. A skimmer loop is also part of the SFPCS. This loop is used to remove debris on the surface of the SFP water.

The staff has reviewed both the licensee's initial submittal and responses to the questions raised and found the licensee's proposal to be acceptable as discussed below.

2.3 New Fuel Storage

The staff found no issues involved in storage of new fuel with increased initial enrichment. Therefore, storage of new fuel is found to be acceptable.

2.4 Spent Fuel Storage

2.4.1 Decay Heat Generation

The licensee calculated the decay heat load for two different cases:

1. The normal heat load, i.e., the heat load generated by a pool filled with 1463 spent fuel assemblies, assuming a normal offload of 76 assemblies is used as the final addition to the SFP, and

2. The maximum heat load, i.e., the pool filled (1463 fuel assemblies) with the final addition assumed to be a full core of 193 assemblies.

In each case, the licensee calculated the decay heat generation by using the methods specified in both ANSI 5.1 and BTP 9-2 of Standard Review Plan (NUREG-0800), Section 9.2.5, "Ultimate Heat Sink," with the following results:

Method	Normal (BTU/Hr.)	Maximum (BTU/Hr.)
ANSI	19.5E6	39.6E6
BTP 9-2	20.8E6	42.2E6

In order to be conservative, the licensee used the higher values found; those found when employing BTP 9-2. A check of some of these values was conducted by the staff. The staff concluded that the method employed by the licensee was conservative, met the criteria of the Standard Review Plan (SRP), and was found acceptable.

2.4.2 SFP Heat Exchanger (HX) Heat Transfer Coefficient

The licensee reported that tests had been conducted with heat exchanger 2B to determine the experimental value of heat transfer coefficient (U) in the equation: $Q = UAF\,\Delta T$

- Q = heat transferred, BTU/Hr.
- U = heat transfer coefficient BTU/HrFt$^{2\circ}$ F
- A = heat transfer area
- F = correction factor for HX
- ΔT = temperature difference

The licensee noted the value of U found for the 2B heat exchanger was 460 BTU/HrFt$^{2\circ}$ F while the value assumed in the design analyses is 321. Use of the lower value of U in the calculation would result, conservatively, in a higher value of SFP coolant temperature than would occur in actuality. Therefore, the use of the lower heat transfer coefficient is acceptable.

2.5. SFP Coolant Temperatures

2.5.1 Normal Case

The licensee reported the results of the analysis for the normal case, using the calculated decay heat generation value of 20.8E6 BTU/Hr. The calculated SFP coolant temperature was reported to be 136° F when using one train. This result is acceptable since it is lower than the SRP guideline of 140° F (SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System").

2.5.2 Maximum Case

The licensee reported that the analysis of coolant temperature in this case was determined to be 137° F when using two SPFC's trains and 180° F when using one train. These results are acceptable since they are lower than the guideline value of less than 212° F as noted in SRP Section 9.1.3.

2.6 Protection of Demineralizer

The licensee noted that the greatest potential for damaging the demineralizer resins would occur on loss of component cooling water (CCW) to the operating heat exchanger when using the demineralizer. In that case, it would take about 2-3 hours to raise the coolant temperature from its normal operating temperature, 90 to 100° F, to 140° F, the temperature at which the demineralizer resins would start to degrade. Annunciator alarms at the chemical and volume control (CVCS) heat exchangers and at upper and lower RCS pump bearing coolers, as well as computer indications at these and other locations served by CCW would indicate low CCW flow. These would serve to indicate both to the senior reactor operator and other operators that failure of the demineralizer could follow; therefore, there is sufficient time for action to be taken to protect the demineralizer. The staff finds this to be acceptable.

3.0 STAFF CONCLUSION

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the McGuire new and spent fuel pool storage racks are acceptable. All normal and accident conditions have been shown to result in a subcritical configuration (k_{eff} less than unity) and thus meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the McGuire TS have been modified to specify the above-mentioned fuel as acceptable for storage in the fresh or spent fuel racks, evaluations of reload core designs (using any enrichment) will, of course, be performed on a cycle-by-cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TS to ensure that reactor operation is acceptable. The staff finds the higher enrichment aspect for the new and spent fuel storage acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on August 24, 1995 (60 FR 44087).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality or the human environment.

6.0 CONCLUSION

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

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Date: November 6, 1995