May 3, 1995

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, SC 29679

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SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M91043, M91044, AND M91045)

Dear Mr. Hampton:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 209, 209, and 206 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 22, 1994, as supplemented by letters dated January 30, March 2, March 13, and May 2, 1995.

The amendments revise TS 3.8 to establish restricted loading patterns and associated burnup criteria for placing fuel in the Oconee spent fuel pools. In addition, the Design Features sections associated with the reactor and fuel storage are also revised.

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:

Leonard A. Wiens, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270 and 50-287

Enclosures:
1. Amendment No. 209 to DPR-38
2. Amendment No. 209 to DPR-47
3. Amendment No. 206 to DPR-55
4. Safety Evaluation

cc w/encl: See next page

*see previous concurrences

CP-1

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

May 3, 1995

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
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Sincerely,

Leonard A. Wiens, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270 and 50-287

Enclosures:
1. Amendment No. 209 to DPR-38
2. Amendment No. 209 to DPR-47
3. Amendment No. 206 to DPR-55
4. Safety Evaluation

cc w/encl: See next page

Mr. J. W. Hampton
Duke Power Company

Oconee Nuclear Station

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated November 22, 1994, as supplemented by letters dated January 30, March 2, March 13, and May 2, 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 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.209 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days 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: May 3, 1995

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated November 22, 1994, as supplemented by letters dated January 30, March 2, March 13, and May 2, 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 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 209, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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: May 3, 1995

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated November 22, 1994, as supplemented by letters dated January 30, March 2, March 13, and May 2, 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 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days 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: May 3, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 209

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 209

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 206

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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
iii	iii
v	v
vi	vi
vii	vii
3.5-11	3.5-11
3.5-30	3.5-30
3.5-30a	3.5-30a
3.5-30b	3.5-30b
3.8-1	3.8-1
3.8-3	3.8-3
-----	3.8-4
-----	3.8-5
-----	3.8-6
-----	3.8-7
-----	3.8-8
-----	3.8-9
-----	3.8-10
-----	3.8-11
5.3-1	5.3-1
5.4-1	5.4-1
5.4-2	-----
6.9-1	6.9-1
-----	6.9-2 (new page/no change)

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

LIST OF TABLES

LIST OF FIGURES

Bases

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

The power-imbalance envelope obtained in accordance with the approved methodology is based on LOCA analyses which have defined the maximum linear heat rate (see Figures 3.5.2-16a, b, and c) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

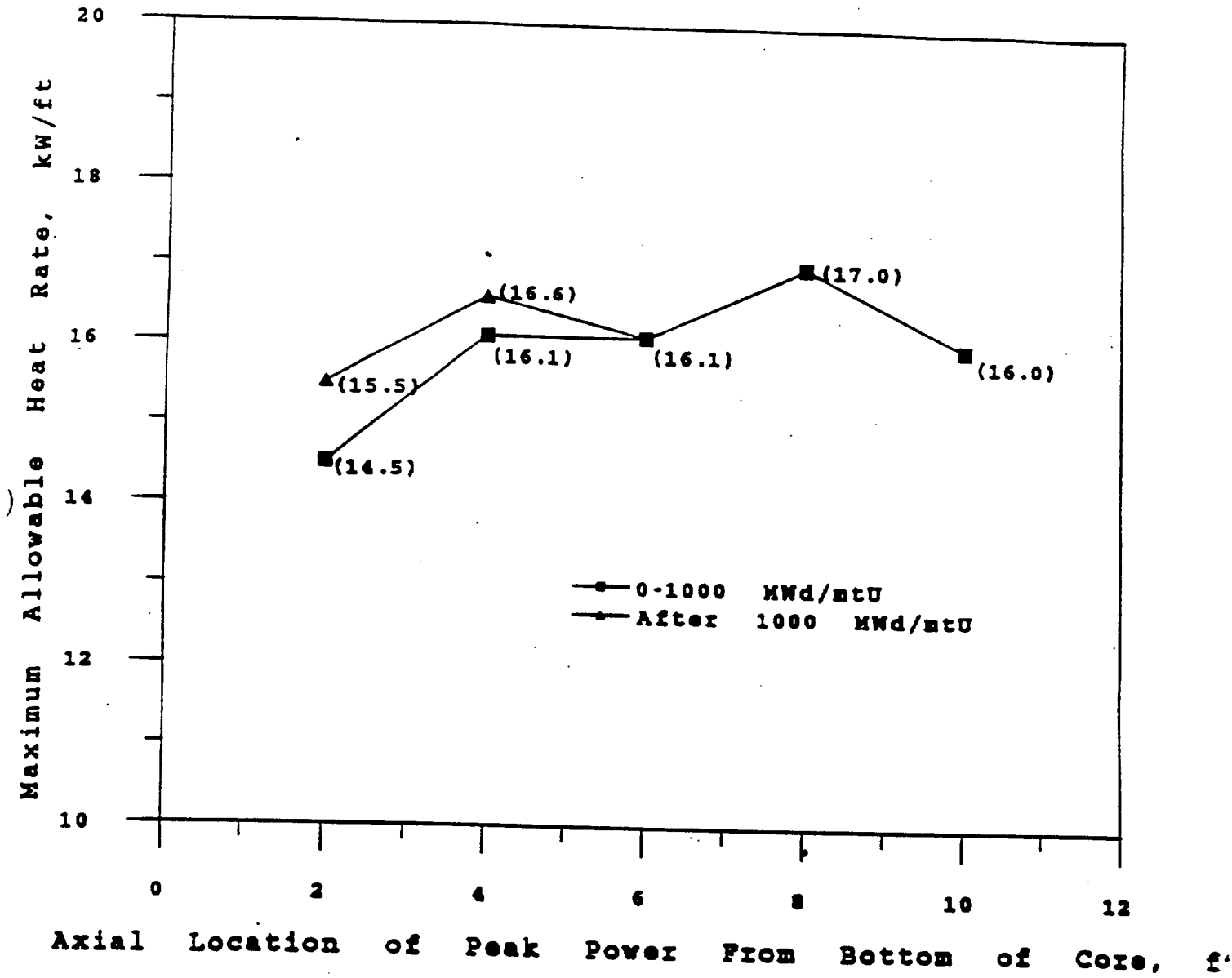
a. Nuclear uncertainty factors
b. Thermal calibration
c. Hot rod manufacturing tolerance factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Groups	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (Axial power shaping rods)

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

LOCA -Limited Maximum Allowable Linear Heat Rate For Mark B8* Fuel Rods

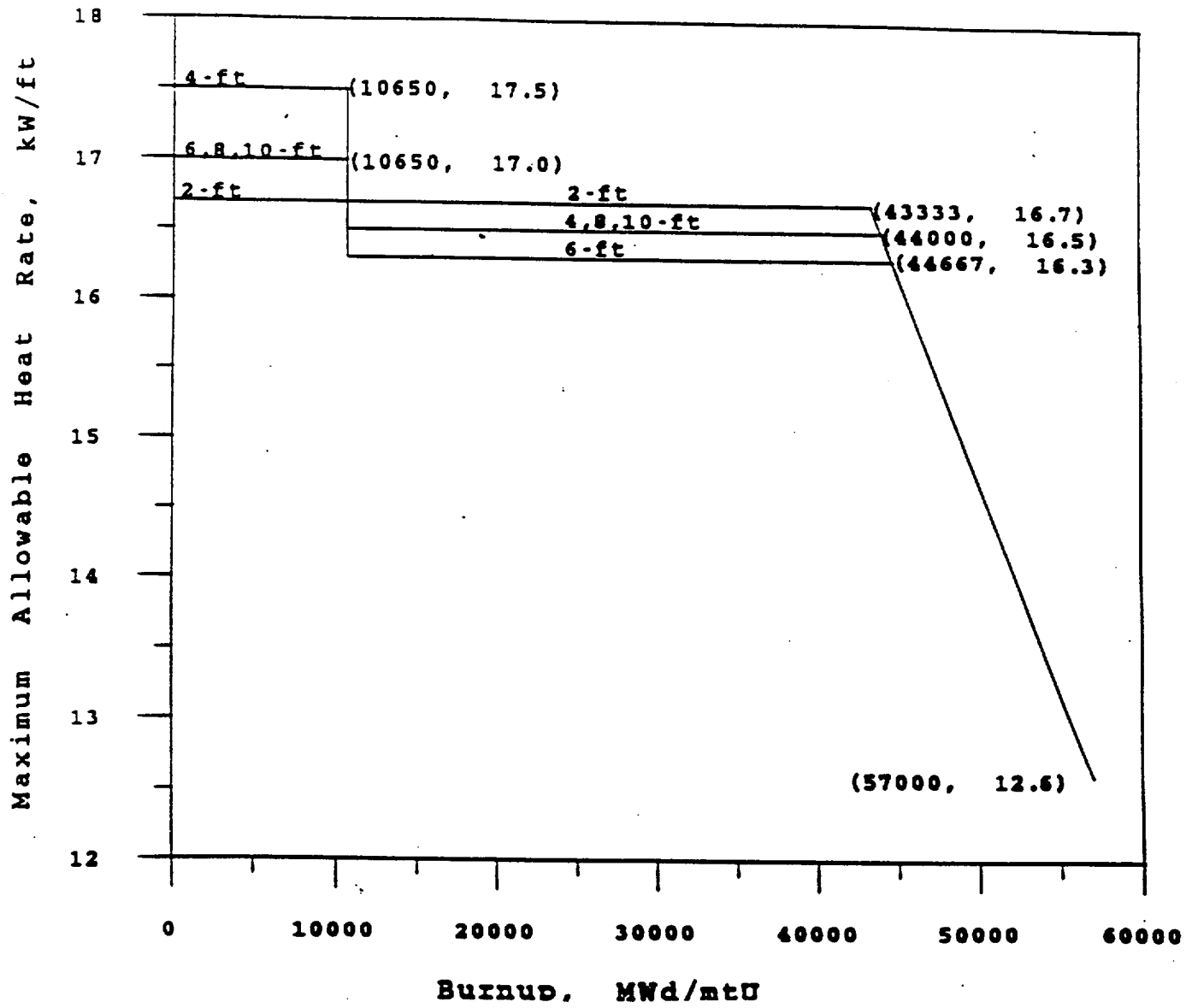


* Mark-B8 fuel rods with a fuel pellet diameter of 0.3686 inches are used in Mark-B8 and earlier fuel assemblies.

Oconee Nuclear Station
Figure 3.5.2-16a

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

LOCA- Limited Maximum Allowable Linear Heat Rate For Mark B9* Fuel Rods

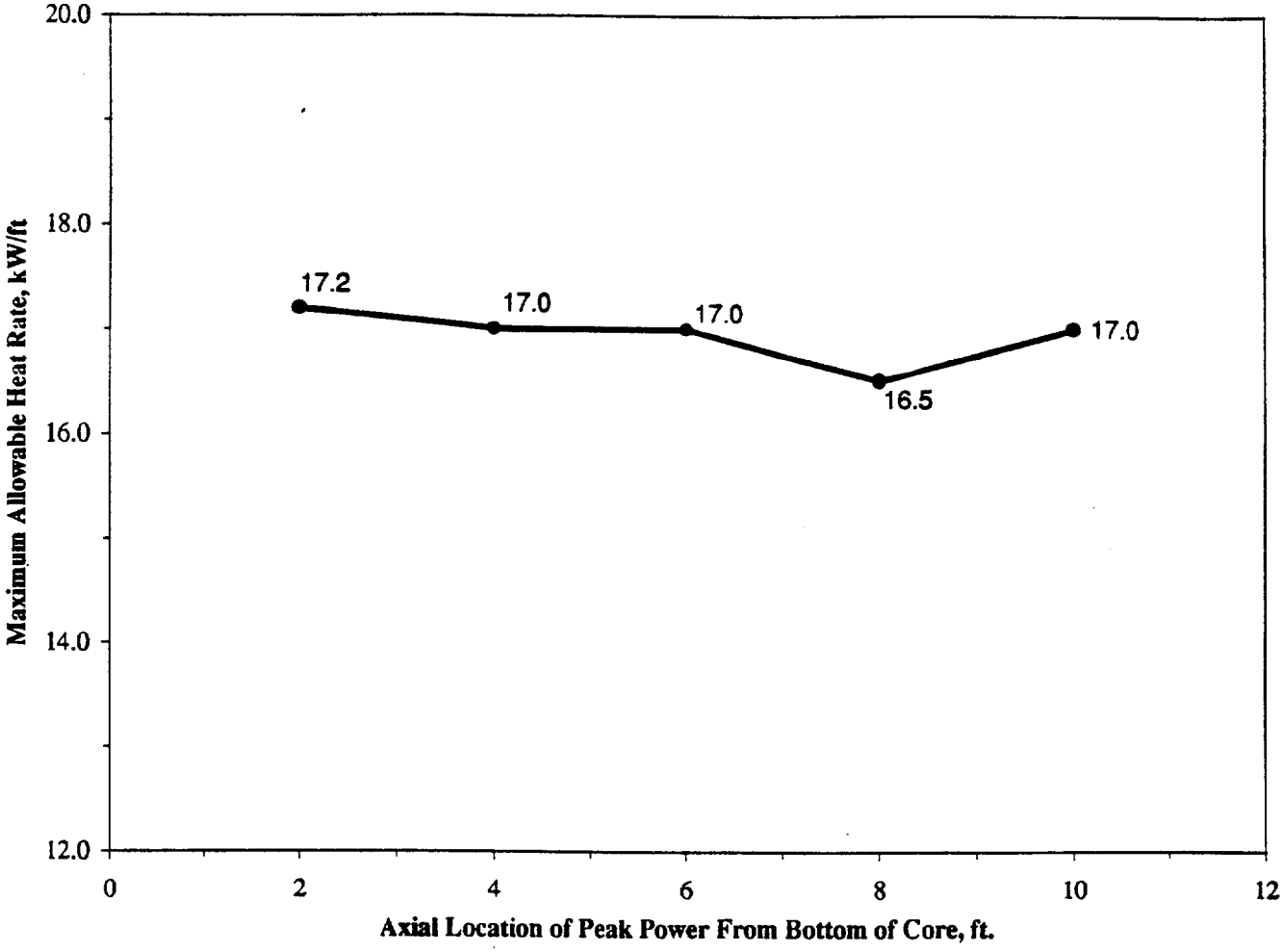


*Mark-B9 fuel rods with a fuel pellet diameter of 0.3700 inches are used in Mark-B9 and Mark-B10 fuel assemblies.

Oconee Nuclear Station
Figure 3.5.2-16b

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

LOCA-Limited Maximum Allowable Linear Heat Rate For Mark-B10* Fuel Rods, (BOL - 25,000 MWd/mtU)



* Mark-B10 fuel rods with a fuel pellet diameter of 0.3735 inches are used in the Mark B10T fuel assemblies.

Oconee Nuclear Station
Figure 3.5.2-16c

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by RIA-3 and by a portable bridge monitor for each bridge which is being used for fuel handling. Radiation levels in the spent fuel storage area shall be monitored by RIA-6 and a portable bridge monitor. If any of these required instruments becomes inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one low pressure injection pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required to shutdown the core to a $k_{eff} \leq 0.99$ if all control rods were removed.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the reactor building at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Both isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

Irradiated fuel assemblies may be handled with the Auxiliary Hoist provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

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

This specification applies when fuel is stored in the spent fuel pool.

3.8.16 a. New or irradiated fuel may be stored in the Spent Fuel Pool shared between Units 1 and 2 in accordance with these limits:

1). Unrestricted storage of fuel meeting the criteria of Table 3.8-1; or

2). Restricted storage in accordance with Figure 3.8-1, of fuel which does <u>not</u> meet the criteria of Table 3.8-1.

b. New or irradiated fuel may be stored in the Spent Fuel Pool for Unit 3 in accordance with these limits:

1). Unrestricted storage of fuel meeting the criteria of Table 3.8-3; or

2) Restricted storage in accordance with Figure 3.8-2, of fuel which does <u>not</u> meet the criteria of Table 3.8-3.

c. This specification applies when fuel is stored in the spent fuel pool.

3.8.17 If the limiting condition for spent fuel pool boron concentration specified in Specification 3.8.15 is not met, 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.

If the limiting conditions for fuel storage in the spent fuel pool specified in Specification 3.8.16 are not met, immediately initiate action to move the noncomplying fuel assembly to the correct location.

<u>Bases</u>

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.1.4 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation.

Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above the limit specified in the Core Operating Limits Report. Although this concentration is sufficient to

maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.90. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours. (3)

The off-site doses for the fuel handling accident are within the guidelines of 10 CFR 100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.8.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

The requirements for spent fuel pool boron concentration specified in Specification 3.8.15 ensure that a minimum boron concentration is maintained in the pool. The requirements for spent fuel assembly storage specified in Specification 3.8.16 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 the spent fuel pool in a subcritical condition during normal operation with the 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 abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of a fuel assembly. This could increase the reactivity of the spent fuel pool. To mitigate this postulated criticality related accident, boron is dissolved in the pool water.

Tables 3.8-1 through 3.8.4 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 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 3.8-16. The use of specific analyses for qualification of previously

unanalyzed fuel includes, but is not 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 Specifications 3.8.16 a and 3.8.16 b represent the maximum reactivity requirements for acceptable storage, substitutions of less reactive components would also meet the $k_{eff} \leq 0.95$ criteria. Hence 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, an additional criterion has been imposed to ensure that the boundary row between these two configurations would not locally increase the reactivity above the required limit.

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 Specifications 3.8.15 and 3.8.16 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 fuel storage requirements and restrictions discussed here and applied in specification 3.8.16 are based on a maximum allowable fuel enrichment of 5.0 weight% U235. The enrichments listed in Tables 3.8-1 through 3.8-4 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% U235. Qualifying burnups for enrichments not listed in the tables may be lineraly interpolated between the enrichments provided. This is because the reactivity of an assembly varies linearly for small ranges of enrichment.

REFERENCES

1. FSAR, Section 9.1.4

2. FSAR, Section 15.11.1

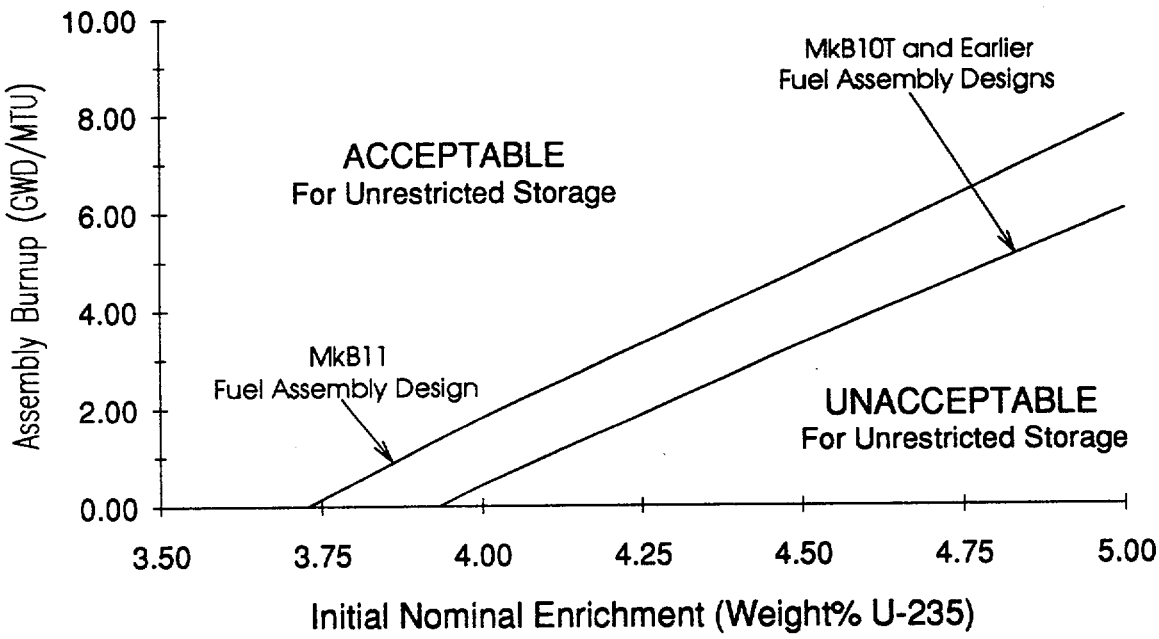
3. FSAR, Section 15.11.2.1

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)

Table 3.8-1

Minimum Qualifying Burnup Versus Initial Enrichment
for Unrestricted Storage in the Unit 1 and 2 Spent Fuel Pool

MkB10T and Earlier Fuel Assembly Designs		MkB11 Fuel Assembly Design	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)	Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
3.93 (or less)	0	3.73 (or less)	0
4.00	0.43	4.00	1.83
4.50	3.30	4.50	4.80
5.00	6.03	5.00	7.95



Fuel which differs from those designs used to determine the requirements of Table 3.8-1 may be qualified for Unrestricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

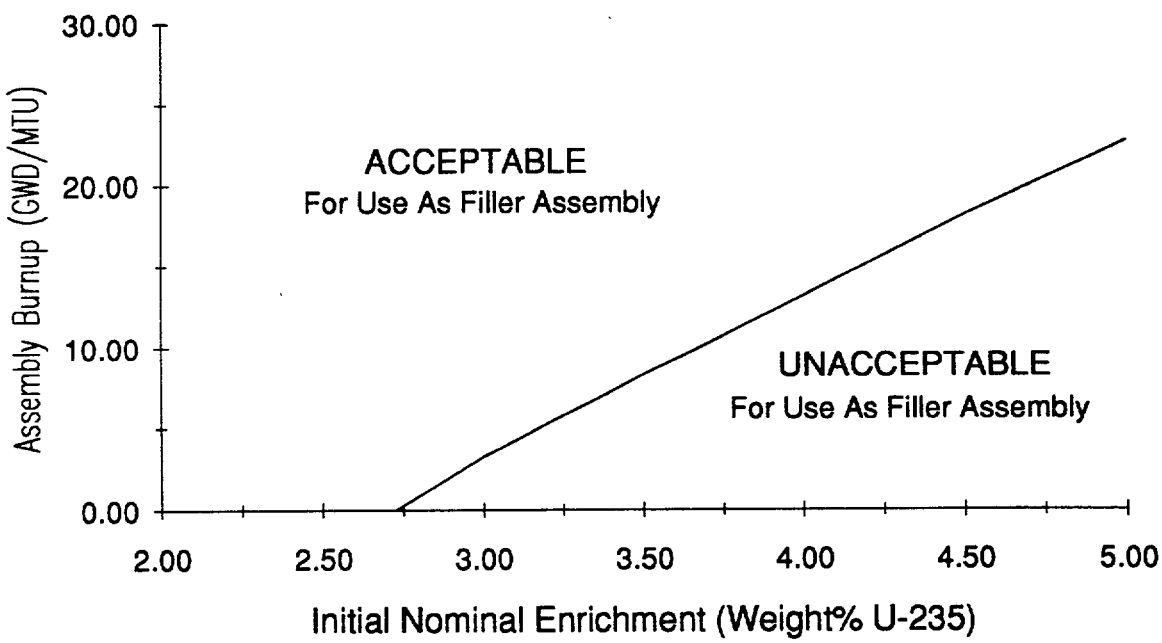
Likewise, previously unanalyzed fuel up to 5.0 weight% U-235 may be qualified for Restricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Table 3.8-2

Minimum Qualifying Burnup Versus Initial Enrichment for Filler Assemblies in the Unit 1 and 2 Spent Fuel Pool

All Fuel Assembly Designs	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.72 (or less)	0
3.00	3.25
3.50	8.22
4.00	13.13
4.50	18.10
5.00	22.69



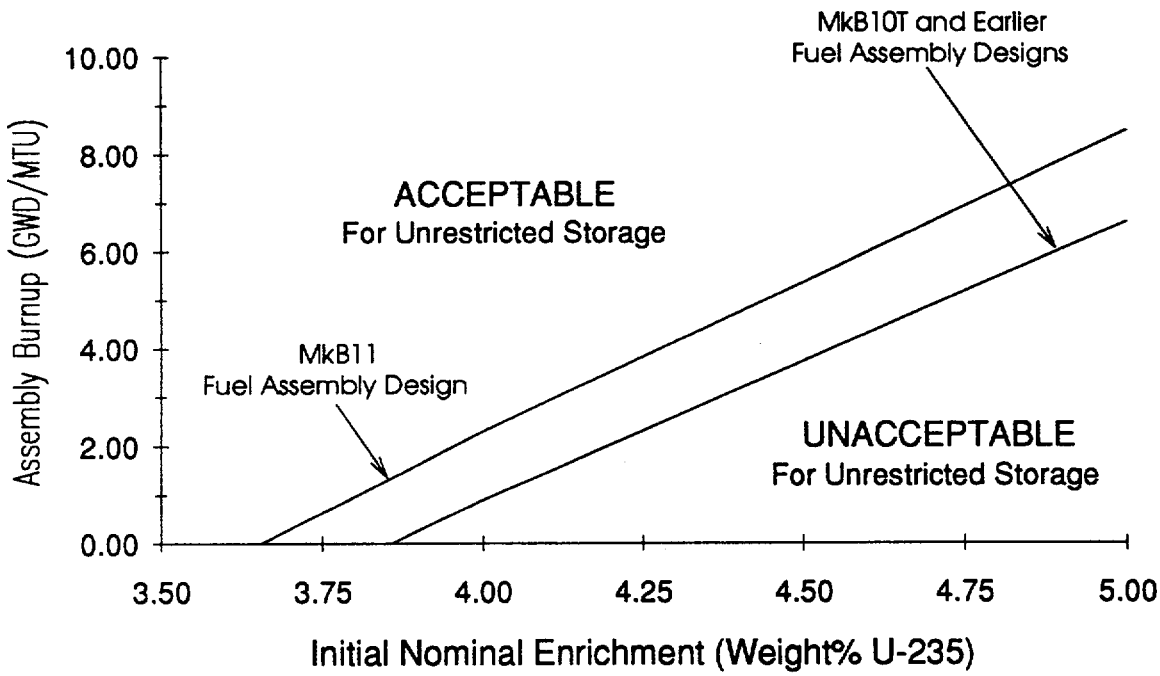
Fuel which differs from those designs used to determine the requirements of Table 3.8-2 may be qualified for use as a Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Table 3.8-3

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Storage in the Unit 3 Spent Fuel Pool

MkB10T and Earlier Fuel Assembly Designs		MkB11 Fuel Assembly Design	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)	Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
3.86 (or less)	0	3.66 (or less)	0
4.00	0.91	4.00	2.31
4.50	3.73	4.50	5.34
5.00	6.60	5.00	8.49



Fuel which differs from those designs used to determine the requirements of Table 3.8-3 may be qualified for Unrestricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

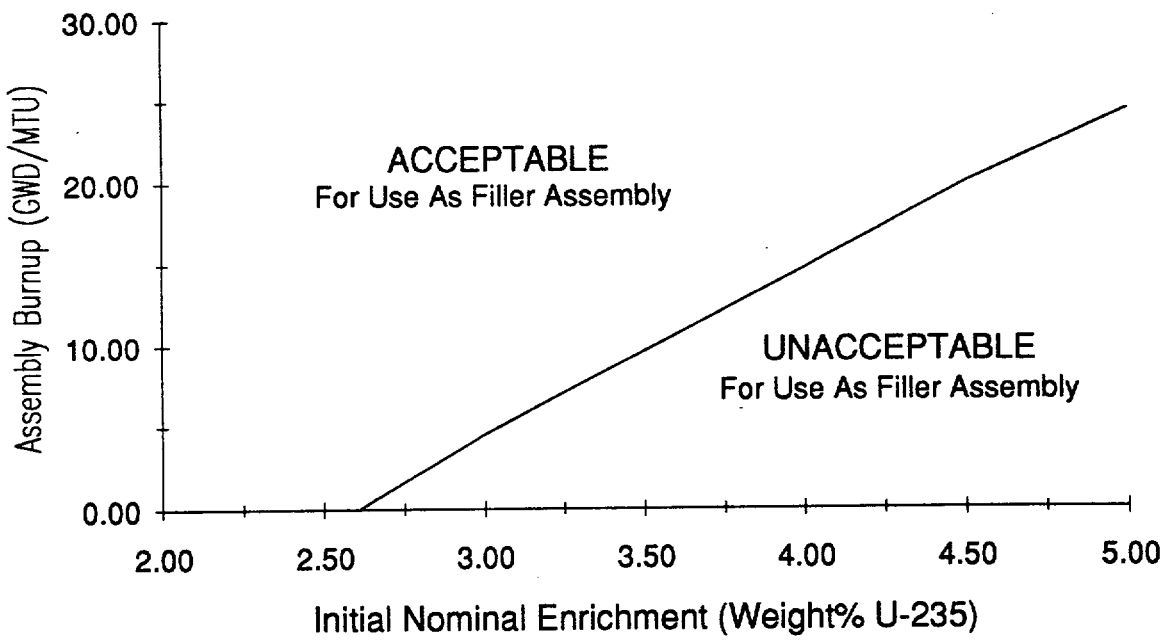
Likewise, previously unanalyzed fuel up to 5.0 weight% U-235 may be qualified for Restricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Table 3.8-4

Minimum Qualifying Burnup Versus Initial Enrichment for Filler Assemblies in the Unit 3 Spent Fuel Pool

All Fuel Assembly Designs	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.61 (or less)	0
3.00	4.49
3.50	9.62
4.00	14.68
4.50	19.96
5.00	24.37

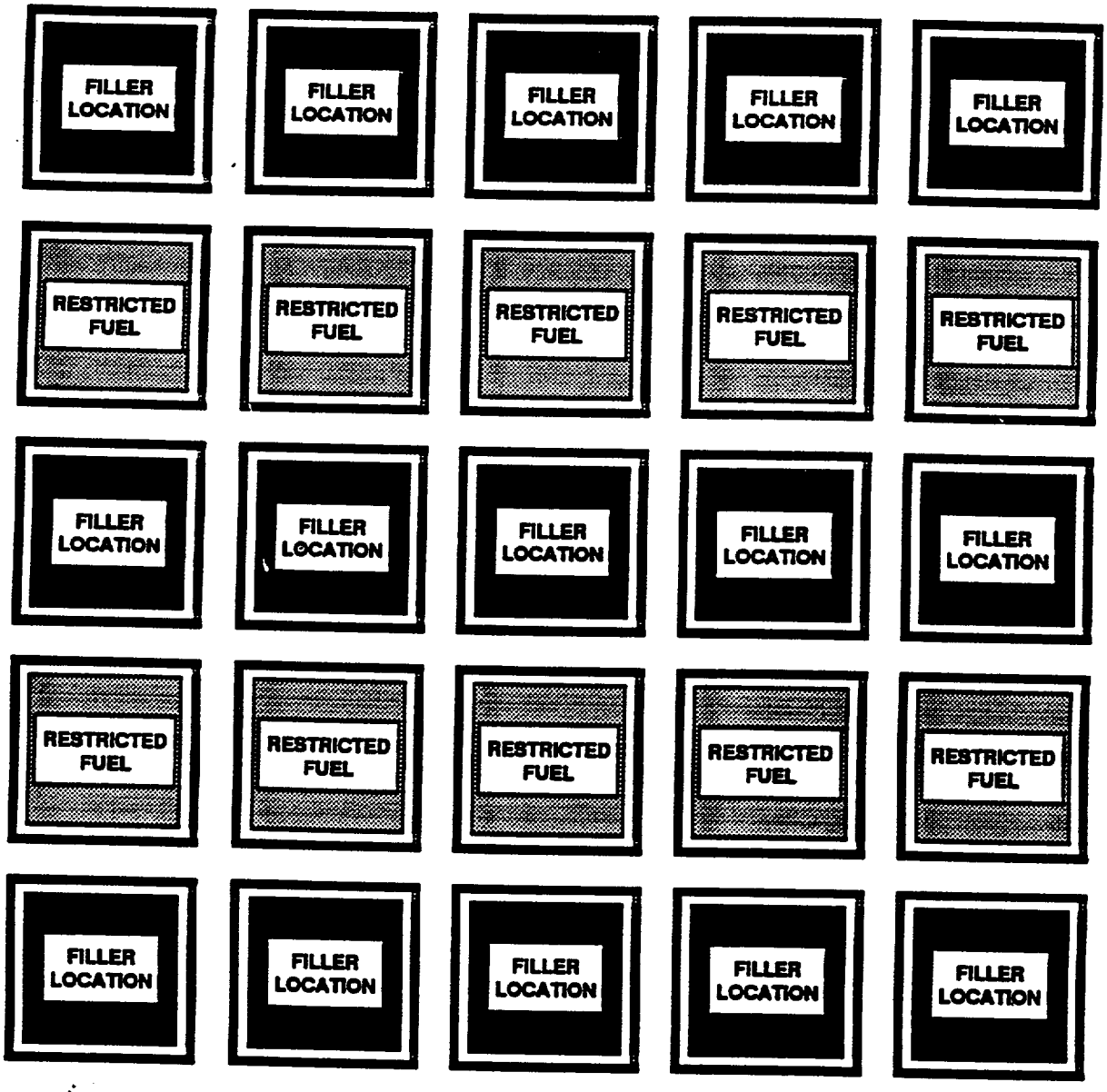


Fuel which differs from those designs used to determine the requirements of Table 3.8-4 may be qualified for use as a Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Figure 3.8-1

Required Loading Pattern for Restricted Storage in the Unit 1 and 2 Spent Fuel Pool



Restricted Fuel: Fuel which does <u>not</u> meet the minimum burnup requirements of Table 3.8-1. (Fuel which does meet the requirements of Table 3.8-1 may be placed in restricted fuel locations as needed)

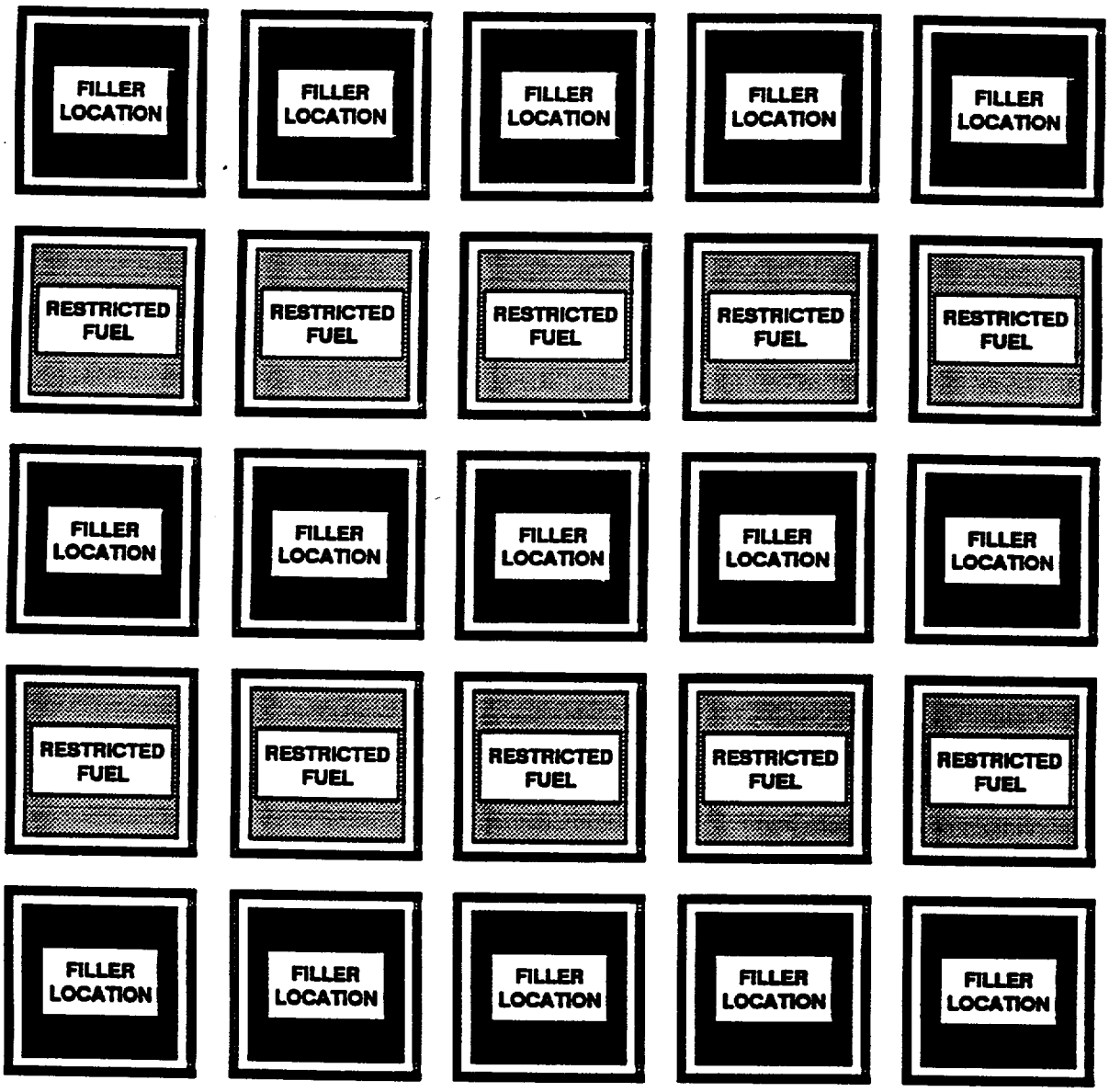
Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.8-2, or an empty cell.

Boundary Condition: Any row bounded by an Unrestricted Storage Area shall contain a row of filler locations (i.e. A row of Restricted fuel assemblies may not be adjacent to a row of Unrestricted fuel assemblies).

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

Figure 3.8-2

Required Loading Pattern for Restricted Storage in the Unit 3 Spent Fuel Pool



Restricted Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.8-3. (Fuel which does meet the requirements of Table 3.8-3 may be placed in restricted fuel locations as needed)

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.8-4, or an empty cell.

Boundary Condition: Any row bounded by an Unrestricted Storage Area shall contain a row of filler locations (i.e. A row of Restricted fuel assemblies may not be adjacent to a row of Unrestricted fuel assemblies).

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy 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. (1).

5.3.1.2 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 4.3-3. The full-length CRA and APSR shall conform to the design described in the FSAR or reload report.(1)

5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements. (2)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670° F. (3)

5.3.2.3 The maximum reactor coolant system volume shall be 12,200 ft^3.

REFERENCES

(1) FSAR Section 4.2.2

(2) FSAR Section 5.2.3.1

(3) FSAR Section 5.2.1

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

5.4 FUEL STORAGE

Specifications

5.4.1 Criticality

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 10.65" center to center distance between fuel assemblies placed in the spent fuel storage racks serving Units 1 and 2.

3) A nomimal 10.60" center to center distance between fuel assemblies placed in the spent fuel storage racks serving Unit 3.

4) A nominal 25.75" center distance between fuel assemblies placed in the fuel transfer canal.

5.4.2 CAPACITY

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1312 fuel assemblies in the spent fuel storage racks serving Units 1 and 2 and 825 fuel assemblies in the spent fuel storage racks serving Unit 3. In addition, up to 4 assemblies and/or 1 failed fuel container may be stored in each fuel transfer canal when the canal is at refueling level. Spent fuel may also be stored in the Oconee Nuclear Station Independent Spent Fuel Storage Installation.

REFERENCES

FSAR, Section 9.1

6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1 Core operating limits shall be established prior to each reload cycle or, prior to any remaining part of a reload cycle, for the following:

(1) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits for Specification 2.1.

(2) Reactor Protective System Trip Setting limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip functions in Specification 2.3.

(3) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1b, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(4) Concentrated Boric Acid Storage Tank volume and boron concentration for Specification 3.2.2.

(5) Core Flood Tank boron concentration for Specification 3.3.3.

(6) Borated Water Storage Tank boron concentration for Specification 3.3.4.

(7) Spent Fuel Pool boron concentration for Specification 3.8.15.

(8) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f.

(9) Power Imbalance Limits for Specification 3.5.2.6.

and shall be documented in the CORE OPERATING LIMITS REPORTS.

6.9.2 The approved methods used to determine the core operating limits given in the Technical Specification 6.9.1 are specified in the CORE OPERATING LIMITS REPORT. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October, 1985.

(2) NFS-1001A, Reload design Methodology, April, 1984

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992.

6.9.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

Amendment No. 209 (Unit 1)
Amendment No. 209 (Unit 2)
Amendment No. 206 (Unit 3)

6.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 209 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated November 22, 1994, as supplemented by letters dated January 30, March 2, March 13, and May 2, 1995, Duke Power Company, et al. (the licensee), submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS). The changes will allow an increased limit for fuel enrichment. The May 2, 1995, letter did not change the scope of the November 22, 1994, application and the initial proposed no significant hazards consideration determination.

The Oconee Nuclear Station has two separate spent fuel storage pools. One pool is shared by the Unit 1 and 2 reactors and has a current maximum nominal enrichment of 4.3 weight percent (w/o) U-235. The Unit 3 pool has a current maximum nominal enrichment of 4.0 w/o U-235. The proposed changes would allow for the storage of fuel with an enrichment not to exceed a nominal 5.00 w/o U-235 in the spent fuel storage racks. As-built manufacturing variations of up to 0.05 w/o U-235 are accounted for in the reactivity analyses.

The increased fuel enrichment limits for fuel storage in the Oconee spent fuel pools were evaluated against the requirements of General Design Criteria 62 of 10 CFR Part 50, Appendix A. The staff's evaluation of the criticality aspects of the proposed changes follows.

2.0 EVALUATION

Each of the two independent spent fuel pools is designed for storage of either fresh or irradiated fuel. The stainless steel cells for the Unit 1 and Unit 2 storage racks are spaced on a 10.65-inch center-to-center distance and have a storage capacity of 1312 fuel assemblies. The Unit 3 racks contain 825 available storage cells with a 10.60-inch center-to-center spacing.

The analysis of the reactivity effects of fuel storage in the spent fuel storage racks was performed with the SCALE system of computer codes using the three-dimensional multi-group Monte Carlo computer code, KENO Va. Neutron cross sections were generated by the NITAWL and BONAMI codes using the 27 Group NDF4 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 Oconee fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment and assembly spacing. 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 nominal 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. The staff has reviewed the licensee's analysis as described above and concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Oconee storage racks with a high degree of confidence.

Duke Power has indicated a desire to incorporate two new fuel assembly designs, designated MkB10T and MkB11, at Oconee in the near future. These new fuel assembly designs are more reactive than the designs previously analyzed and would violate the spent fuel storage design criteria under the current TS. Therefore, a reanalysis was performed by DPC to allow for an increase in the maximum allowable initial enrichment of the stored fuel. The results indicate that fuel with nominal enrichments up to 3.73 w/o U-235 for MkB11 fuel and 3.93 w/o U-235 for all other Oconee fuel can be stored in every cell of the Unit 1 and 2 spent fuel storage racks. For the Unit 3 storage racks, MkB11 fuel having an initial enrichment of up to 3.66 w/o U-235 or all other Oconee fuel having a maximum enrichment of 3.86 w/o U-235 can be stored in every cell. Since the reanalysis was performed using the methodology described above, the results stated are acceptable.

To enable the storage of depleted fuel assemblies initially enriched to greater than the 3.73 w/o and 3.93 w/o limits stated above, 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 3.8-1 in which a fresh 3.73 w/o MkB11 enriched fuel assembly yields the same rack reactivity as an initially enriched 5.00 w/o MkB11 assembly depleted to 7.95 GWD/MTU. A similar result is shown for other Oconee fuel assemblies where a fresh 3.93 w/o enriched assembly yields the same storage rack reactivity as an initially enriched 5.00 w/o assembly depleted to 6.03 GWD/MTU. The curves shown in the Table include biases due to methodology, Boraflex shrinkage, and boron self-shielding, 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 and concludes that they are appropriately conservative and the burnup limits are acceptable.

New or irradiated assemblies with initial enrichments up to 5.00 w/o U-235 which do not meet the requirements for unrestricted storage must be placed in a restricted loading pattern. Reactivity analyses for these assemblies, stored in every other row of the spent fuel pool, were performed using the previously discussed methods. Acceptable fuel assemblies which qualify for storage in the alternating rows between adequately depleted assemblies are shown in Table 3.8-2 and are referred to as filler assemblies. These filler assemblies were also determined from minimum burnup versus initial enrichment calculations as described above. These special configurations have 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.

Similar analyses were performed for the Unit 3 spent fuel pool and the resulting minimum qualifying burnups are shown in Tables 3.8-3 and 3.8-4.

Since fuel will be stored in the pools according to two different loading configurations to accommodate both unrestricted and restricted storage, the boundary conditions between these configurations were analyzed to determine the effects of neutronic coupling. The results show that, in order to satisfy the k_{eff} criterion, a row of restricted assemblies must not be directly adjacent to a row of unrestricted fuel. This additional restriction has been incorporated into the proposed Oconee fuel storage TS.

A statement is included in Tables 3.8-1 through 3.8-4 to 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 McGuire Nuclear Station and has been implemented to accommodate storage of Oconee spent fuel shipped to McGuire for storage. 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 3.8-16. At the staff's request, the Bases was revised to include additional discussion which reflects the intended use of this provision. The staff finds this proposed specification acceptable.

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 by TS 3.8.15 when fuel is stored in the spent fuel pool 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 TS 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 3.8.15 is being added to establish limits for the required spent fuel pool boron concentration in the Core Operating Limits Report (COLR). The relocation of the minimum spent fuel pool boron concentration to the COLR has previously been approved by the NRC in other licensing actions.

(2) TS 3.8.16 is being added to specify the new fuel storage requirements given in Tables 3.8-1 through 3.8-4 and Figures 3.8-1 and 3.8-2 based on the reactivity analyses evaluated above.

(3) TS 3.8.17 is being added to state the required actions if the limiting conditions stated in TS 3.8.15 or 3.8.16 are not met.

(4) The Bases is being modified to allow for specific criticality analyses for special situations without requiring additional TS changes, as described above. In addition, the Bases are being changed to address new Loss of Coolant Accident (LOCA) limits for the new MkB10T fuel assemblies.

(5) TS 5.3.1 is being revised to accommodate changes in the fuel assembly design evaluated above. The proposed changes are consistent with the standard TS.

(6) TS 5.4.1.1 is being revised to remove references to maximum fuel enrichments since this is now specified in TS Tables 3.8-1 through 3.8-4. In addition, TS 5.4.1.1 and TS 5.4.1.2 are being combined into TS 5.4.1.

(7) TS 5.4.2.1 is being modified to delete extraneous information.

(8) TS 5.4.3, which specifies the spent fuel pool boron concentration, is being relocated to TS 3.8.15.

(9) TS 6.9.1 is being changed to include the spent fuel pool boron concentration in the list of COLR parameters.

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the Oconee spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Although the Oconee 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 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.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings (60 FR 8746 dated February 15, 1995; 60 FR 16185 dated March 29, 1995). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

Attachments:
TS Tables

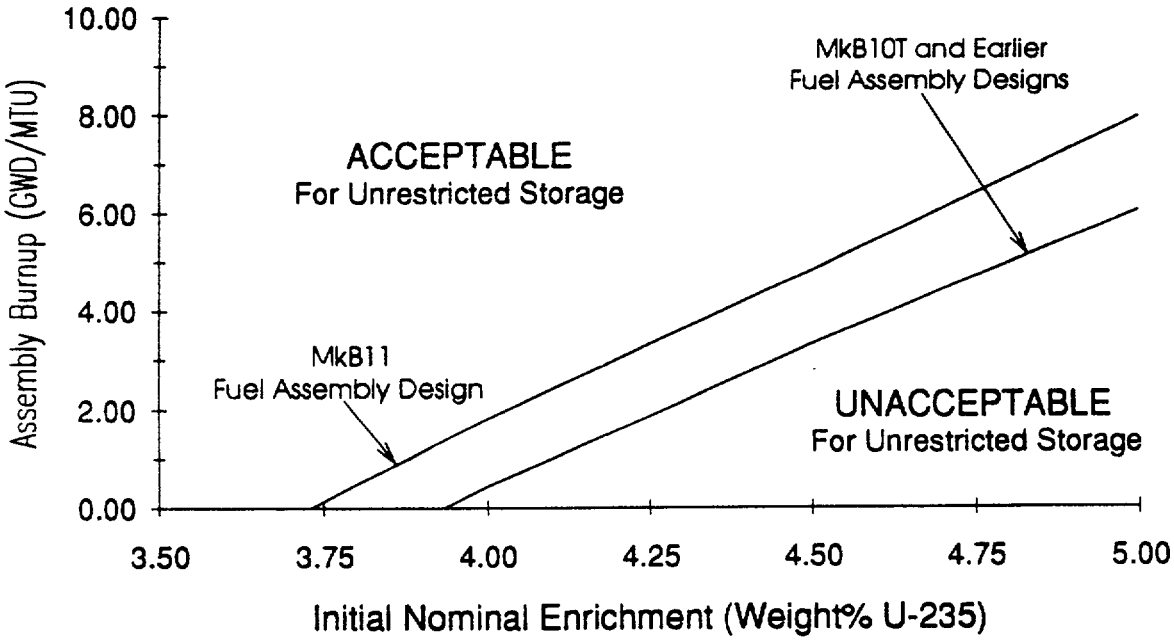
Principal Contributor: L. Kopp

Date: May 3, 1995

Table 3.8-1

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Storage in the Unit 1 and 2 Spent Fuel Pool

MkB10T and Earlier Fuel Assembly Designs		MkB11 Fuel Assembly Design	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)	Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
3.93 (or less)	0	3.73 (or less)	0
4.00	0.43	4.00	1.83
4.50	3.30	4.50	4.80
5.00	6.03	5.00	7.95



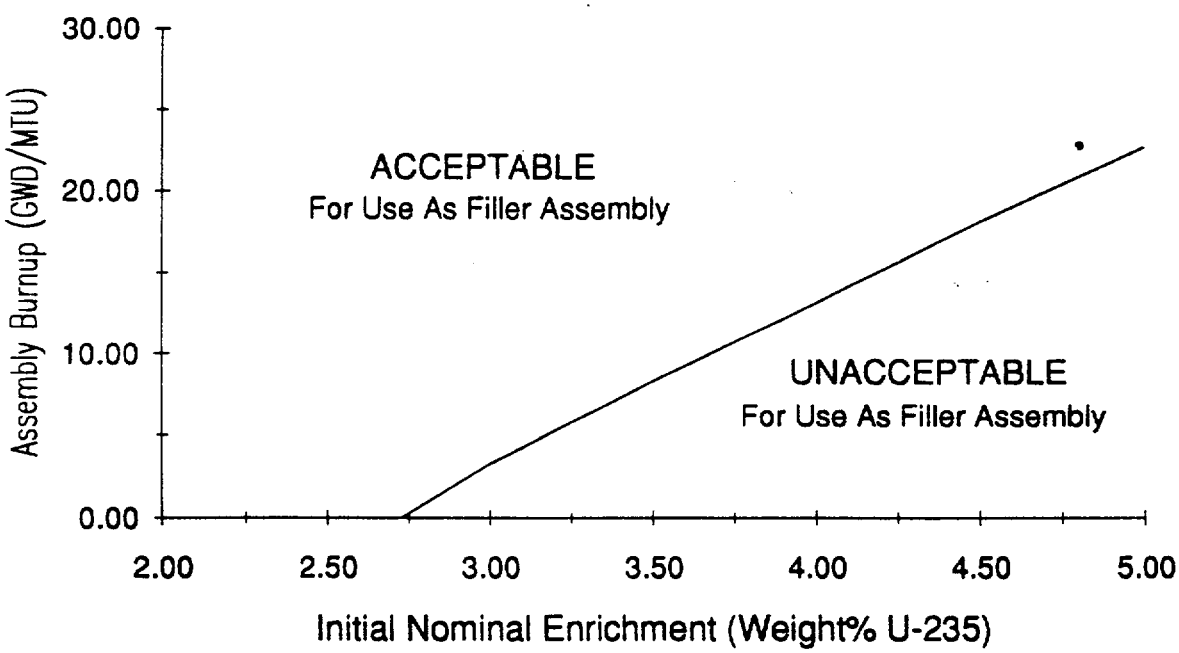
Fuel which differs from those designs used to determine the requirements of Table 3.8-1 may be qualified for Unrestricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Likewise, previously unanalyzed fuel up to 5.0 weight% U-235 may be qualified for Restricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.8-2

Minimum Qualifying Burnup Versus Initial Enrichment for Filler Assemblies in the Unit 1 and 2 Spent Fuel Pool

All Fuel Assembly Designs	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.72 (or less)	0
3.00	3.25
3.50	8.22
4.00	13.13
4.50	18.10
5.00	22.69

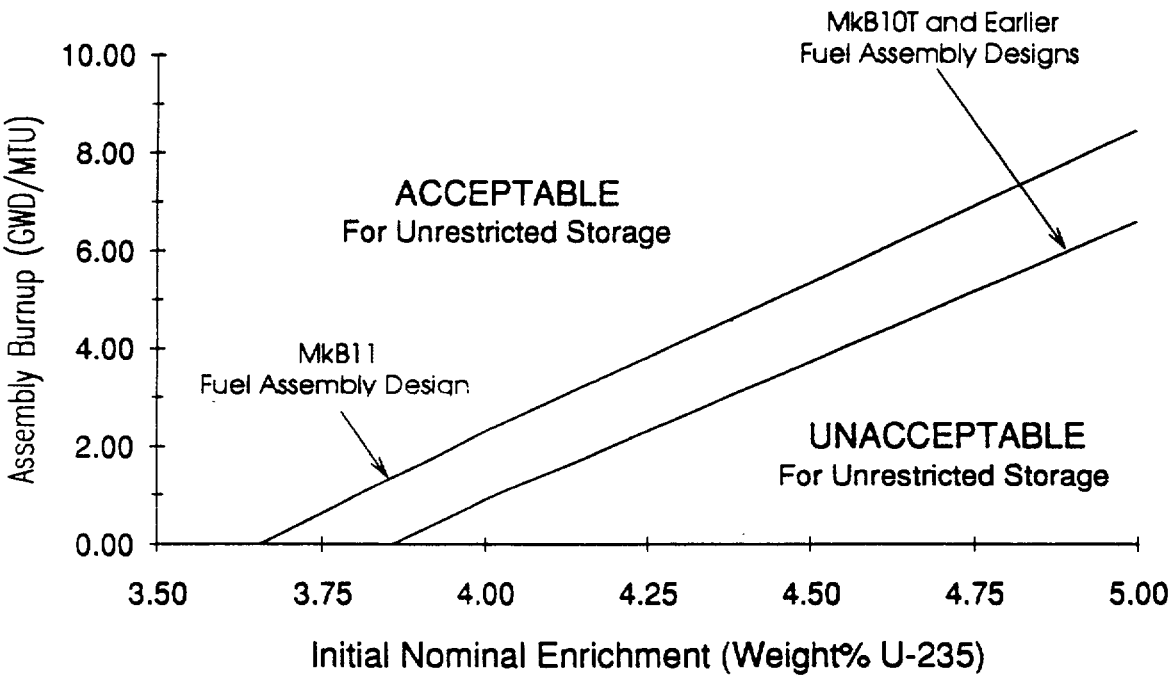


Fuel which differs from those designs used to determine the requirements of Table 3.8-2 may be qualified for use as a Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.8-3

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Storage in the Unit 3 Spent Fuel Pool

MkB10T and Earlier Fuel Assembly Designs		MkB11 Fuel Assembly Design	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)	Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
3.86 (or less)	0	3.66 (or less)	0
4.00	0.91	4.00	2.31
4.50	3.73	4.50	5.34
5.00	6.60	5.00	8.49



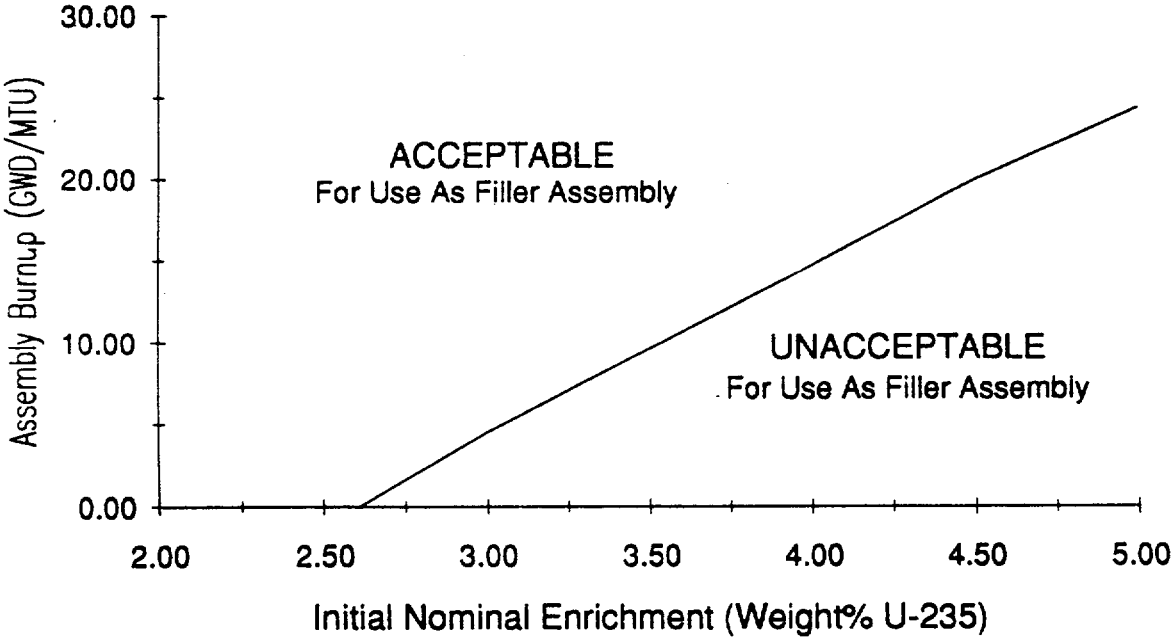
Fuel which differs from those designs used to determine the requirements of Table 3.8-3 may be qualified for Unrestricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Likewise, previously unanalyzed fuel up to 5.0 weight% U-235 may be qualified for Restricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.8-4

Minimum Qualifying Burnup Versus Initial Enrichment for Filler Assemblies in the Unit 3 Spent Fuel Pool

All Fuel Assembly Designs	
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.61 (or less)	0
3.00	4.49
3.50	9.62
4.00	14.68
4.50	19.96
5.00	24.37



Fuel which differs from those designs used to determine the requirements of Table 3.8-4 may be qualified for use as a Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.