DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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## 3.8 FUEL MOVEMENT AND STORAGE IN THE SPENT FUEL POOL

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.

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; or

3). Another configuration determined to be acceptable by means of an analysis to assure that $k_{eff}$ is less than or equal to 0.95.

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; or

3) Another configuration determined to be acceptable by means of an analysis to assure that $k_{eff}$ is less than or equal to 0.95.

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.

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.

Specifications 3.8.16 a.3 and 3.8.16 b.3 allow for specific criticality analyses for configurations other than those explicitly defined in Specification 3.8.16. 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.

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 $k_{eff}$ ≤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 15.1.11

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 Enrichment Weight% U-235 | Assembly Burnup (GWD/MTU) | Initial 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 |

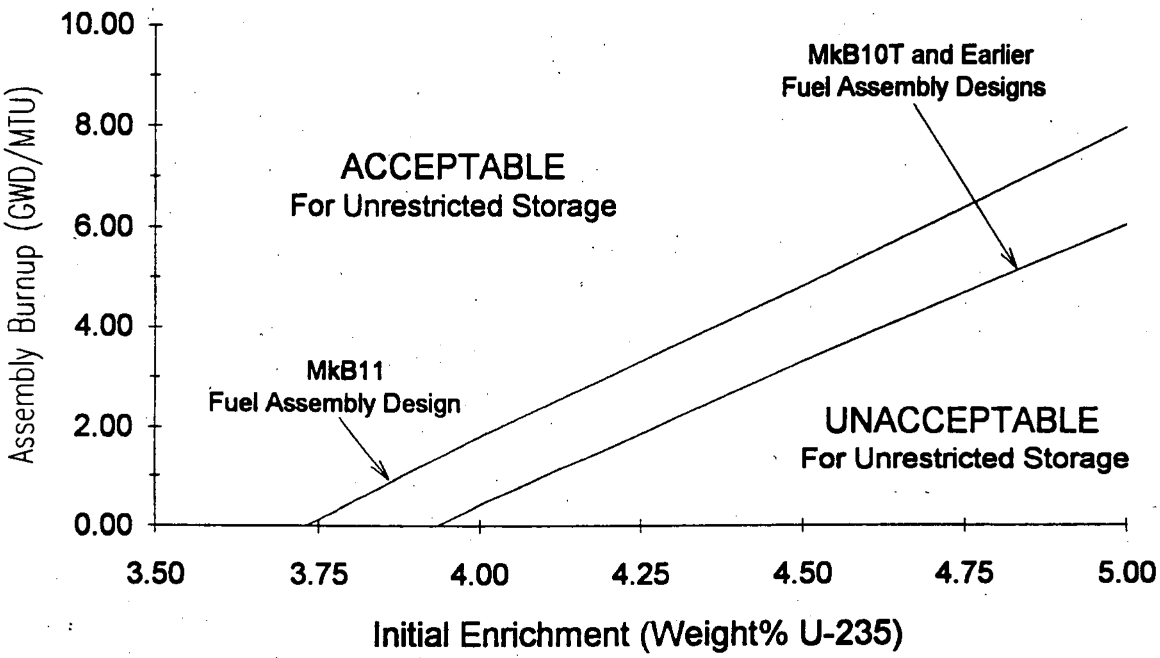


Table 3.8-2

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

| All Fuel Assembly Designs | |
|---|---|
| Initial 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 |

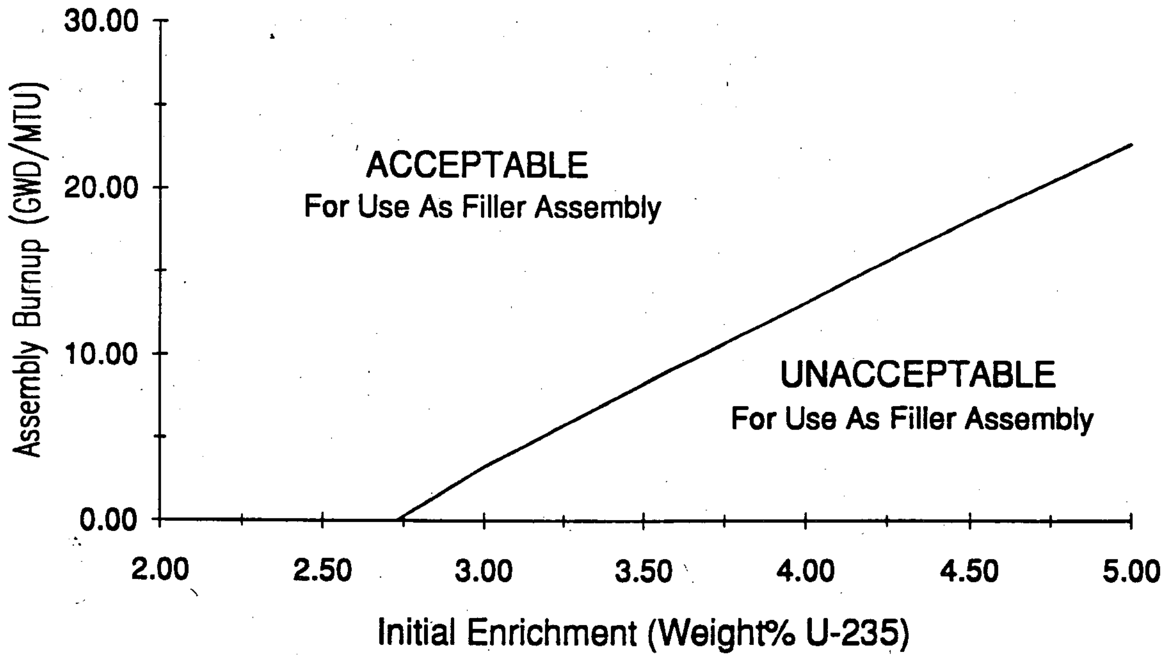


Table 3.8-3

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 Enrichment Weight% U-235 | Assembly Burnup (GWD/MTU) | Initial 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 |

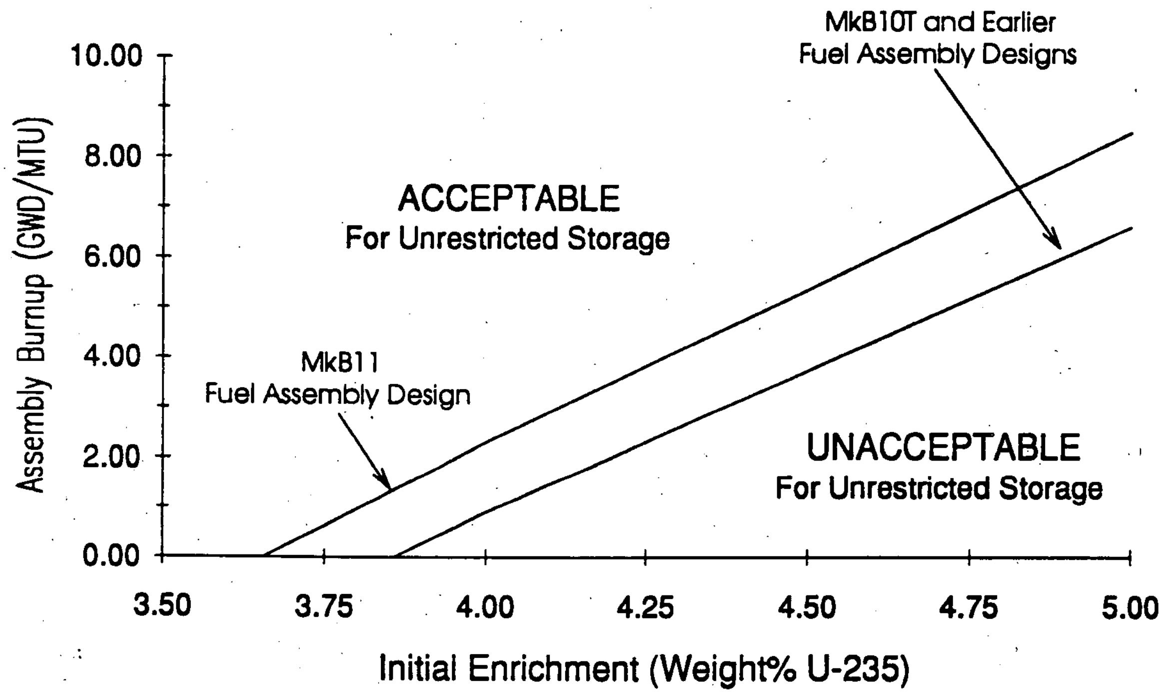


Table 3.8-4

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

| All Fuel Assembly Designs | |
|---|---|
| Initial 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 |

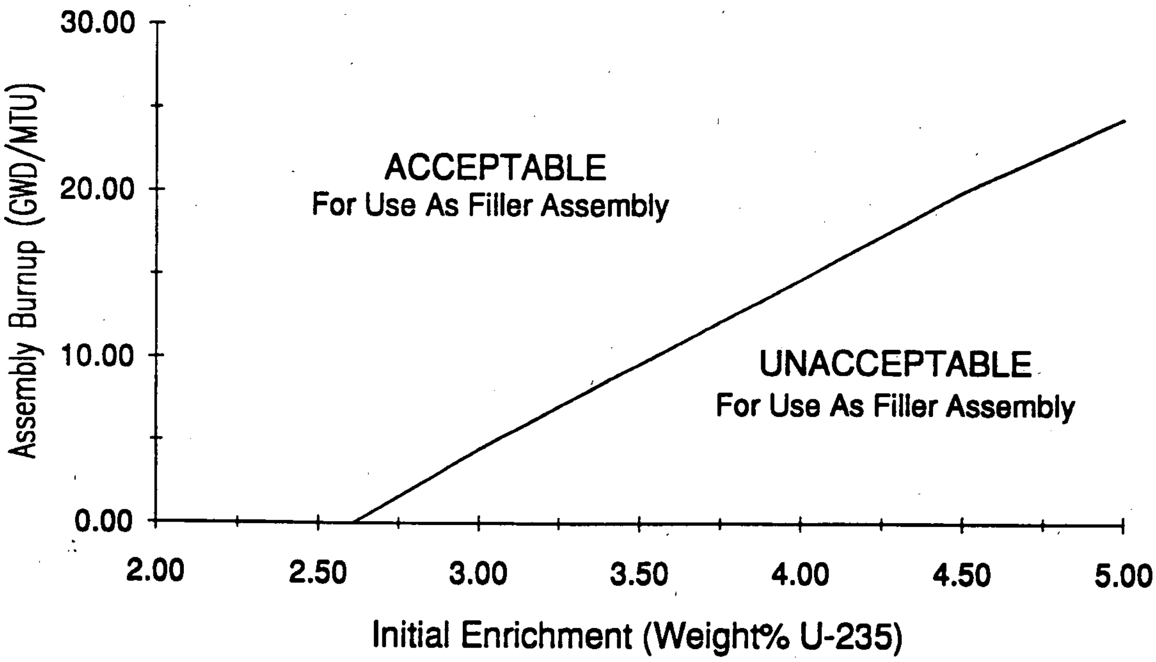
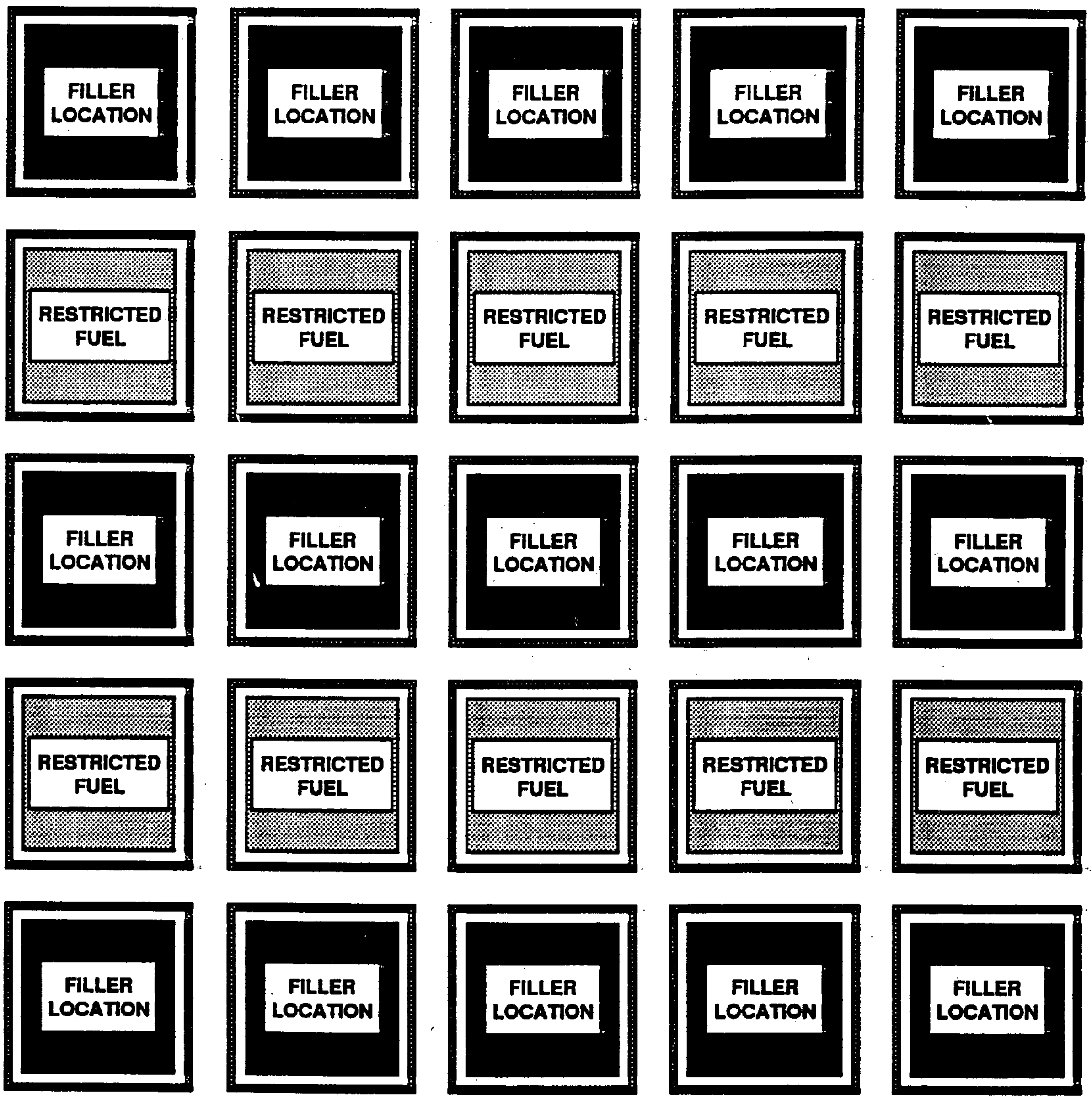


Figure 3.8-1

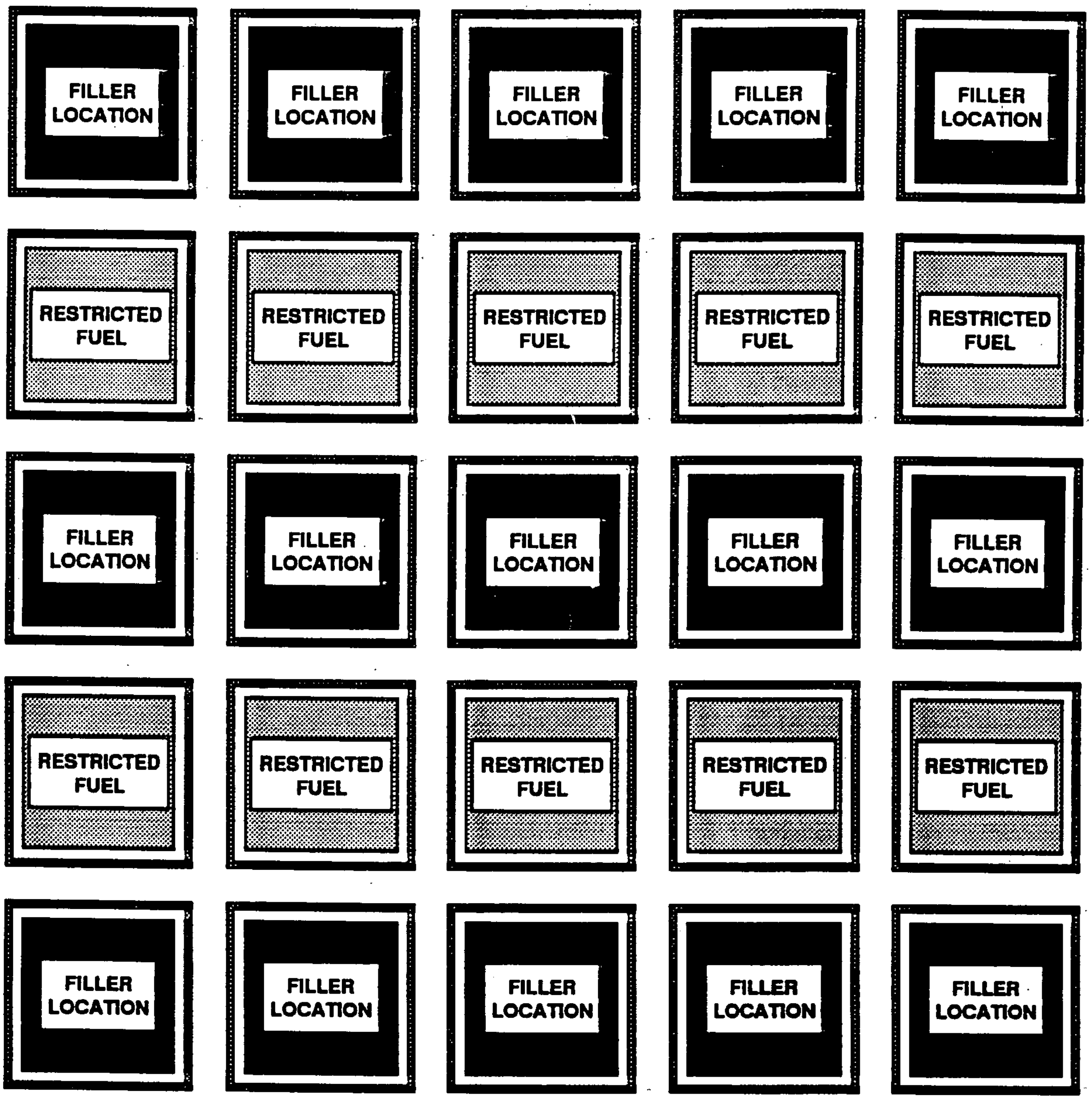
Required Loading Pattern for Restricted Storage in the Unit 1 and 2 Spent Fuel Pool




| | |
|---|---|
| Restricted Fuel: | Fuel which does not 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). |

## Figure 3.8-2

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




| | |
|---|---|
| Restricted Fuel: | Fuel which does <u>not</u> 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). |

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

## 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.

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.

# Sample COLR Pages

## 1.0 CORE OPERATING LIMITS

This Core Operating Limits Report for O3C16 has been prepared in accordance with the requirements of Technical Specification 6.9. The core operating limits within this report have been developed using NRC-approved methodology (References 1, 2, 3, and 4). The RPS protective limits and maximum allowable setpoints are documented in References 6 and 7, and validated in References 5 and 8 for O3C16. Operational limits and requirements are documented in Reference 5. The reactor coolant system design flow used in References 5 and 8 for O3C16 is 107.5 % (of 88,000 gpm per pump). The core operating limits have been developed with a radial local peaking factor ($F^N_{\Delta H}$) of 1.714 and an axial peaking factor ($F^N_Z$) of 1.5.

The following cycle-specific core operating limits are included in this report. All computations performed in setting these limits used the approved SIMULATE methodology.

1) RPS protective limits (Figures 1.1 and 1.2), and RPS maximum allowable setpoints (Figures 1.3 and 1.4),

2) Quadrant power tilt operational limits,

3) Steady state operating band,

4) Power-imbalance operational limits,

5) Rod index operational and shutdown margin-restricted limits, and

6) BWST, SFP, CBAST, and CFT boron requirements.

## 1.1 REFERENCES


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

2) DPCo, Oconee Nuclear Station, Reload Design Methodology II, DPC-NE-1002A, October 1985.

3) DPCo, Oconee Nuclear Station, Reload Design Methodology, NFS-1001A, April 1984.

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

5) O3C16 Maneuvering Analysis, DPCo calculational file, OSC-5839, September 1994.

6) Variable Low Pressure Safety Limit, DPCo calculational file, OSC-4048, Revision 0, July 1990.

7) Power-Imbalance Safety Limits and Tech. Spec. Setpoints Using Error-Adjusted Flux-Flow Ratio of 1.094, DPCo calculational file, OSC-5604, Revision 0, November 1993.

8) O3C16 Thermal-Hydraulic Evaluation, DPCo calculational file, OSC-5844, Revision 0, August 1994.

Sample COLR Pages

## Oconee 3 Cycle 16

## BWST, SFP, CBAST, and CFT BORON REQUIREMENTS

## 0 EFPD to EOC

1) The BWST boron concentration shall be greater than 2210 ppm and less than 3000 ppm (referred to by Tech Spec 3.3.4).

2) The Spent Fuel Pool boron concentration shall be greater than 2210 ppm and less than 3000 ppm (referred to by Tech Spec 3.8.15).

3) The equivalent of at least 1100 cubic feet of 11,000 ppm boron shall be maintained in the CBAST (referred to by Tech Spec 3.2.2).

4) The boron concentration in each CFT shall be greater than 1835 ppm (referred to by Tech Spec 3.3.3).

5) The refueling canal boron concentration shall be greater than 2210 ppm and less than 3000 ppm (referred to by the bases to Tech Spec 3.8.4). This concentration is large enough to maintain 1% Δk/k shutdown margin with all control rods out of the core at temperatures down to 33 deg F, and with no credit for xenon worth.

ATTACHMENT II

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## LIST OF FIGURES

## FUEL ~~LOADING-AND-REFUELING~~ MOVEMENT AND STORAGE IN THE SPENT FUEL POOL

### 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.

~~3.8.15 a. No fuel which has an enrichment greater than 4.0 weight percent $U^{235}$ (53 grams of $U^{235}$ per axial centimeter of fuel assembly) will be stored in the spent fuel pool for Unit 3.~~

~~b. No fuel which has an enrichment greater than 4.3 weight percent $U^{235}$ (57 grams of $U^{235}$ per axial centimeter of fuel assembly) will be stored in the spent fuel pool for Units 1 and 2.~~

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 not meet the criteria of Table 3.8-1; or

3). Another configuration determined to be acceptable by means of an analysis to assure that $k_{eff}$ is less than or equal to 0.95.

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 not meet the criteria of Table Table 3.8-3; or

3). Another configuration determined to be acceptable by means of an analysis to assure that $k_{eff}$ is less than or equal to 0.95.

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

3.8.17 If the limiting condition for spent fuel 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 concentraton to within its limits.

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

Bases

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.

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.

Specifications 3.8.16 a.3 and 3.8.16 b.3 allow for specific criticality analyses for configurations other than those explicitly defined in Specification 3.8.16. 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.

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 rresponding limits. This does not preclude movement of a fuel assembly to a safe sition.

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 linearly 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 15.1.11

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 Enrichment Weight% U-235 | Assembly Burnup (GWD/MTU) | Initial 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 |

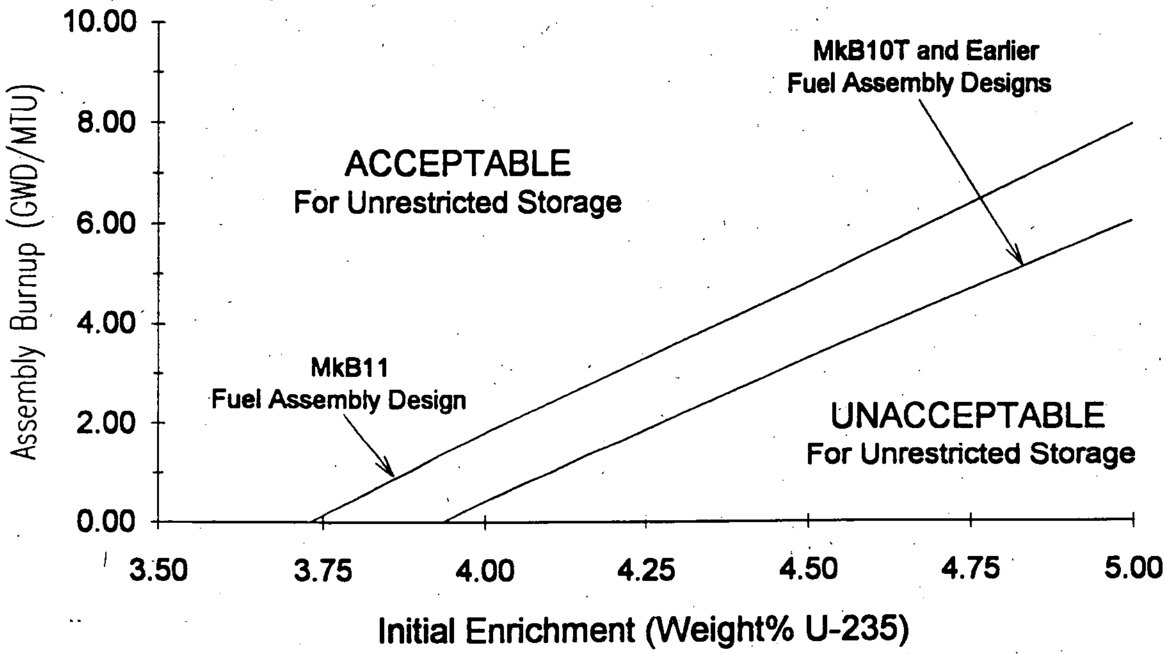


Table 3.8-2

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

| All Fuel Assembly Designs | |
|---|---|
| Initial 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 |

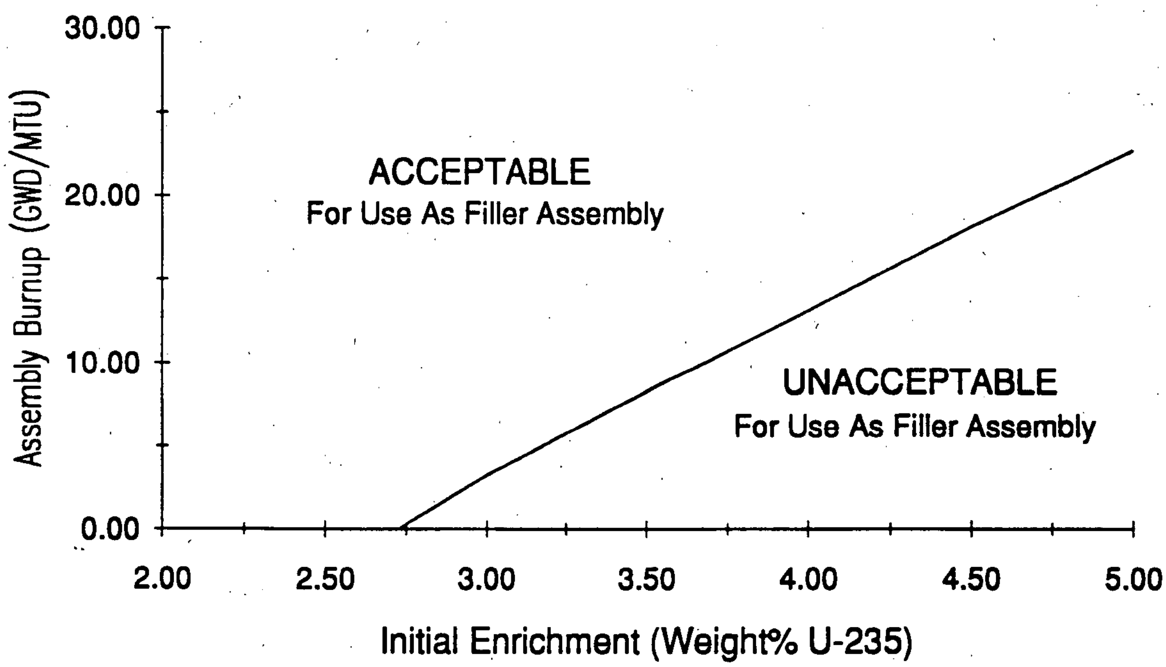


Table 3.8-3

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 Enrichment Weight% U-235 | Assembly Burnup (GWD/MTU) | Initial 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 |

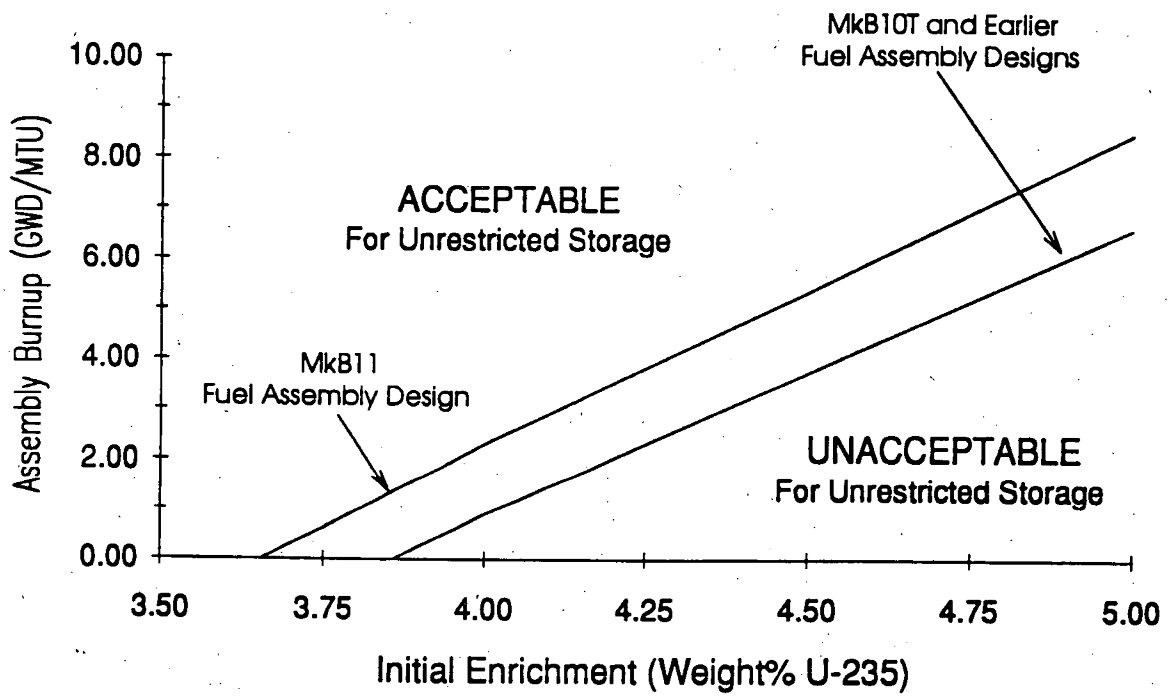
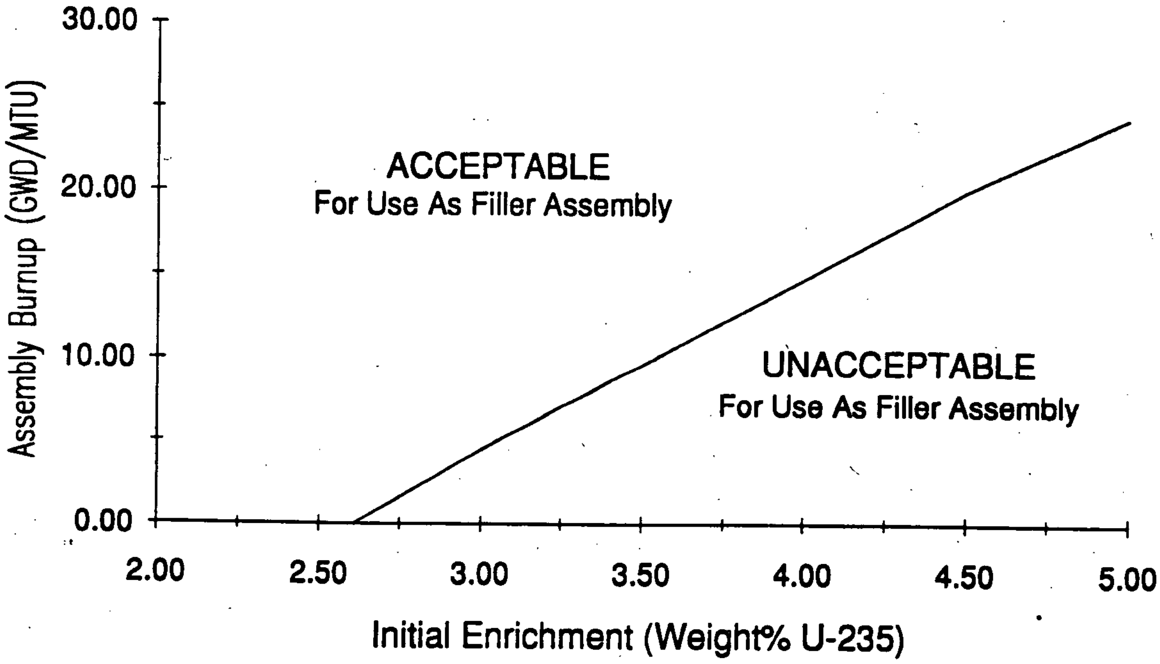


Table 3.8-4

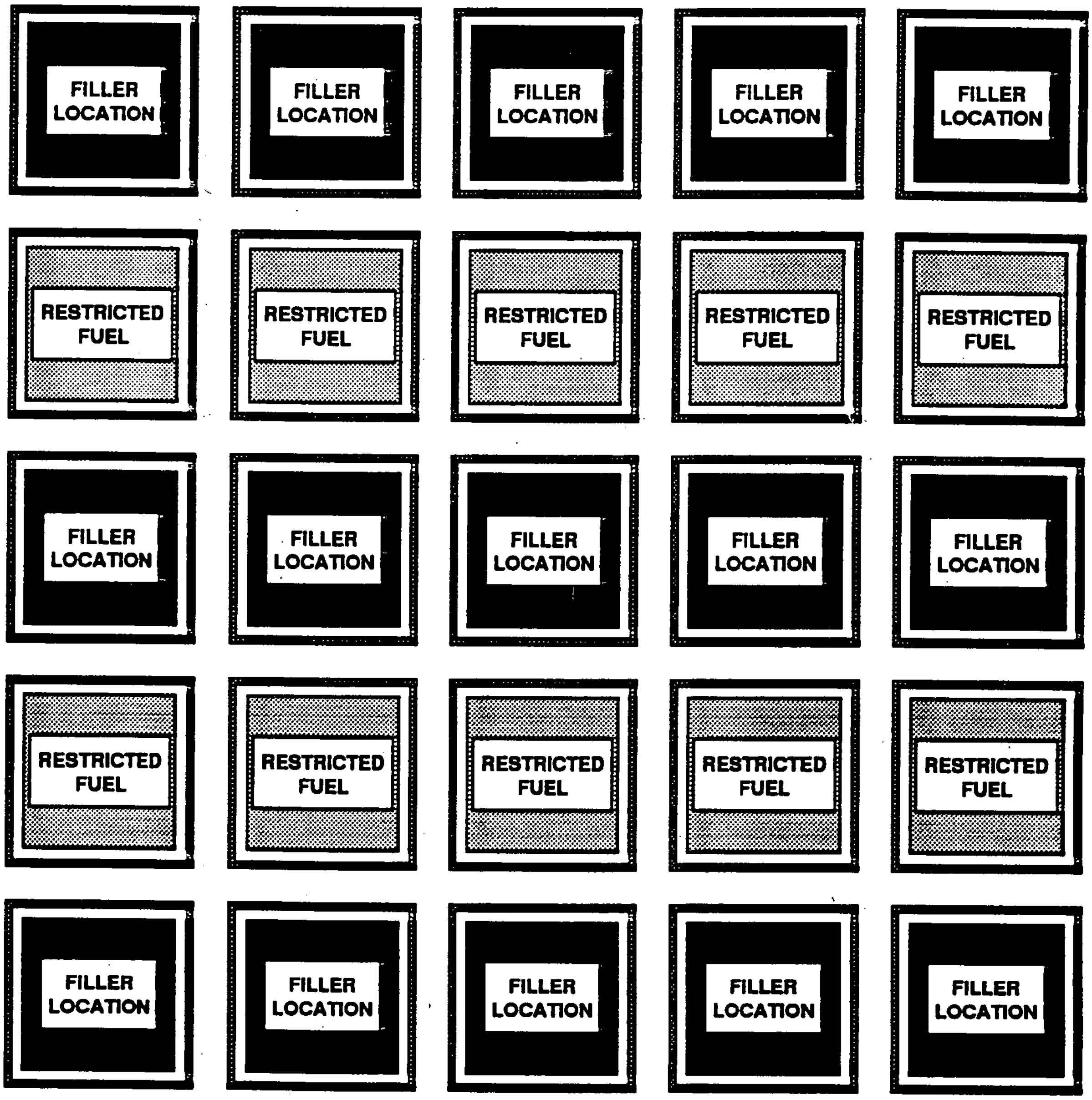
Minimum Qualifying Burnup Versus Initial Enrichment for Filler Assemblies in the Unit 3 Spent Fuel Pool

| All Fuel Assembly Designs | |
|---|---|
| Initial 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 |



<u>Figure 3.8-1</u>

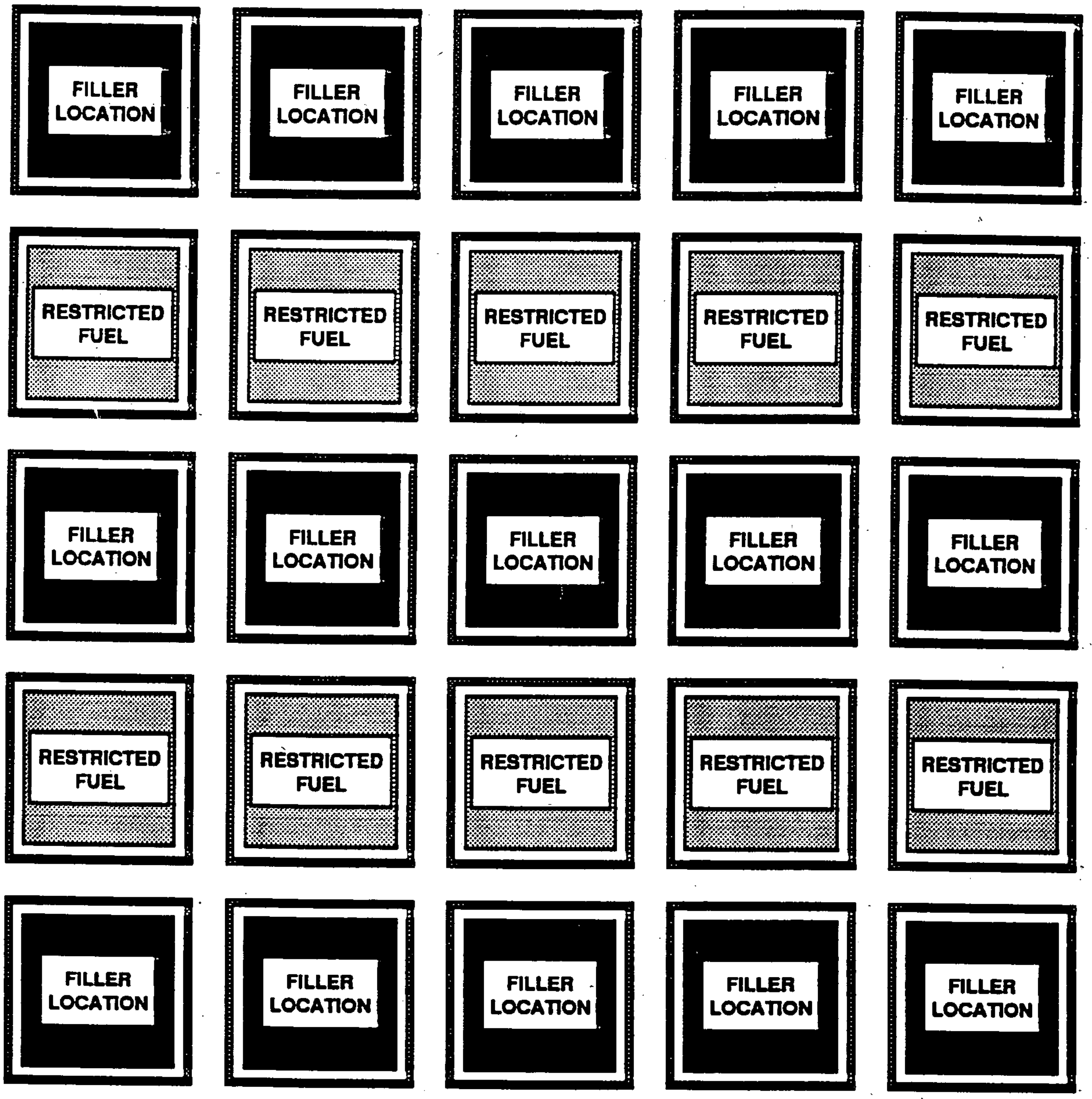
<u>Required Loading Pattern for Restricted Storage in the Unit 1 and 2 Spent Fuel Pool</u>




| | |
|---|---|
| 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). |

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). |

## 5.3 REACTOR

Specification

### 5.3.1 Reactor Core

~~5.3.1.1-----The-reactor-core-contains-approximately-93-metric-tons-of-slightly-enriched uranium-dioxide-pellets.--The-pellets-are-encapsulated-in-zircaloy-4-tubing-to form-fuel-rods.--The-reactor-core-is-made-up-of-177-fuel-assemblies,--all-of which-are-prepressurized-with-Helium-(1)~~

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-----The-fuel-assemblies-shall-form-an-essentially-cylindrical-lattice-with-an active-height-range-of-140.5-in.--to-142-in.--and-an-equivalent-diameter-of 120.9-in.--(1)~~

5.3.1~~.3~~.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)

~~5.3.1.4-----Initial-core-and-reload-fuel-assemblies-and-rods-shall-conform-to-design-and evaluation-described-in-the-FSAR.~~

### 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

~~4 NEW AND SPENT FUEL STORAGE FACILITIES~~

~~Specification~~

~~5.4.1 New Fuel Storage~~

~~5.4.1.1 New fuel will normally be stored in the spent fuel pool serving the respective unit.~~

~~In the spent fuel pool serving Units 1 and 2, the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 10.65 inches in both directions. This spacing is sufficient to maintain a $K_{eff}$ ≤0.95 when flooded with unborated water, based on fuel with an enrichment of 4.3 weight percent $U^{235}$.~~

~~In the spent fuel pool serving Unit 3, the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 10.60 inches in both directions. This spacing is sufficient to maintain a $K_{eff}$ ≤0.95 when flooded with unborated water, based on fuel with an enrichment of 4.0 weight percent $U^{235}$.~~

~~5.4.1.2 New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center to center distance of 2' 1 3/4". One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.~~

~~5.4.1.3 New fuel may also be stored in shipping containers.~~

~~5.4.2 Spent Fuel Storage~~

~~4.2.1 Irradiated fuel assemblies will be stored, prior to off site shipment, in a stainless steel lined spent fuel pool.~~

~~The spent fuel pool serving Units 1 and 2 is sized to accommodate a full core of irradiated fuel assemblies in addition to the con current storage of the largest quantity of new spent fuel assemblies predicted by the fuel management program.~~

~~Provisions are made in the Unit 1, 2 spent fuel pool to accommo date up to 1312 fuel assemblies and in the Unit 3 spent fuel pool up to 825 fuel assemblies.~~

~~5.4.2.2 Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.~~

~~5.4.2.3 Spent fuel may also be stored in Oconee Nuclear Station Independent Spent Fuel Storage Installation.~~

~~5.4.3 Whenever there is fuel in the pool, the spent fuel pool is filled with water borated to the concentration that is used in the reactor cavity and fuel transfer canal during refueling operations.~~

~~.4 The spent fuel pool and fuel transfer canal racks are designed for an earthquake force of 0.1g ground motion.~~

## 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 to 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

## 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.

~~(7)~~ (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.

~~(8)~~ (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.

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.

Sample COLR Pages

## 1.0 CORE OPERATING LIMITS

This Core Operating Limits Report for O3C16 has been prepared in accordance with the requirements of Technical Specification 6.9. The core operating limits within this report have been developed using NRC-approved methodology (References 1, 2, 3, and 4). The RPS protective limits and maximum allowable setpoints are documented in References 6 and 7, and validated in References 5 and 8 for O3C16. Operational limits and requirements are documented in Reference 5. The reactor coolant system design flow used in References 5 and 8 for O3C16 is 107.5 % (of 88,000 gpm per pump). The core operating limits have been developed with a radial local peaking factor ($F^N_{\Delta H}$) of 1.714 and an axial peaking factor ($F^N_Z$) of 1.5.

The following cycle-specific core operating limits are included in this report. All computations performed in setting these limits used the approved SIMULATE methodology.

1) RPS protective limits (Figures 1.1 and 1.2), and RPS maximum allowable setpoints (Figures 1.3 and 1.4),

2) Quadrant power tilt operational limits,

3) Steady state operating band,

4) Power-imbalance operational limits,

5) Rod index operational and shutdown margin-restricted limits, and

6) BWST, SFP, CBAST, and CFT boron requirements.

## 1.1 REFERENCES


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

2) DPCo, Oconee Nuclear Station, Reload Design Methodology II, DPC-NE-1002A, October 1985.

3) DPCo, Oconee Nuclear Station, Reload Design Methodology, NFS-1001A, April 1984.

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

5) O3C16 Maneuvering Analysis, DPCo calculational file, OSC-5839, September 1994.

6) Variable Low Pressure Safety Limit, DPCo calculational file, OSC-4048, Revision 0, July 1990.

7) Power-Imbalance Safety Limits and Tech. Spec. Setpoints Using Error-Adjusted Flux-Flow Ratio of 1.094, DPCo calculational file, OSC-5604, Revision 0, November 1993.

8) O3C16 Thermal-Hydraulic Evaluation. DPCo calculational file. OSC-5844, Revision 0, August 1994.

## Oconee 3 Cycle 16

## BWST, SFP, CBAST, and CFT BORON REQUIREMENTS

## 0 EFPD to EOC

1) The BWST boron concentration shall be greater than 2210 ppm and less than 3000 ppm (referred to by Tech Spec 3.3.4).

2) The Spent Fuel Pool boron concentration shall be greater than 2210 ppm and less than 3000 ppm (referred to by Tech Spec 3.8.15).

3) The equivalent of at least 1100 cubic feet of 11,000 ppm boron shall be maintained in the CBAST (referred to by Tech Spec 3.2.2).

4) The boron concentration in each CFT shall be greater than 1835 ppm (referred to by Tech Spec 3.3.3).

5) The refueling canal boron concentration shall be greater than 2210 ppm and less than 3000 ppm (referred to by the bases to Tech Spec 3.8.4). This concentration is large enough to maintain 1% $\Delta k/k$ shutdown margin with all control rods out of the core at temperatures down to 33 deg F, and with no credit for xenon worth.

ATTACHMENT 3
Technical Justification

This section provides the technical justification for the proposed modifications to the ONS Technical Specifications. These changes revise certain Design Features specifications to address two new fuel designs, increase the initial fuel enrichment, and establish restricted loading patterns and associated burnup criteria for qualifying fuel in both Spent Fuel Pools at the Oconee Nuclear Station. These changes are necessary to allow storage and irradiation of fuel for the upcoming Unit 3 Cycle 16 reload core. These changes will also increase design and operational flexibility, while at the same time maintaining acceptable criticality safety margins and decay heat removal capabilities. In addition, several administrative changes have been included in order to provide clarity to the Specifications and bring them more in line with the NUREG-1430 B&W Standard Technical Specification (STS) format. A description of each of the changes being requested is given below.

1. The Technical Specification Table of Contents is being changed to change the title of Specifications 3.8 and 5.4. The new titles are consistent with the STS. These changes are purely administrative.

2. The Technical Specification List of Tables and List of Figures is being changed to add Tables 3.8-1 through 3.8-4, and Figures 3.8-1 and 3.8-2 which are both part of the new Specification 3.8.16 (see number 4 below). This change is purely administrative.

3. The title for Specification 3.8 is being changed from 'Fuel Loading and Refueling' to 'Fuel Movement and Storage In the Spent Fuel Pool'. By including spent fuel pool storage, the new title better addresses the purpose of this Specification. This change is administrative.

4. Specification 3.8.15 (fuel enrichment) is being replaced with Specifications 3.8.15 (Spent Fuel Pool boron concentration), 3.8.16 (fuel storage), and 3.8.17 (action statements). These changes are being made in order to accommodate the added Spent Fuel Pool (SFP) storage restrictions and to provide increased operational flexibility. This change also provides more consistency with STS format.

    a. The SFP boron concentration has been relocated from Specification 5.4.3. This specification has also been changed to provide clarity, and to provide a specific reference for this limit in the Core Operating Limits Report (COLR). The affected pages in the O3C16 COLR are included in Attachment 1 to illustrate how the SFP boron concentration will be administratively controlled. This change is also more consistent with STS format.

    b. SFP storage limitations have been moved from Specification 3.8.15 to 3.8.16. The new SFP limit specification increases the initial enrichment limit to 5.00 weight% $U^{235}$, establishes restricted loading patterns with associated burnup criteria, and expands the previous limits to include two new fuel designs which will begin arriving on site next year. The proposed changes are necessary to allow storage of the new fuel assembly designs in the spent fuel pool. In addition, these changes increase reload design and operating flexibility while ensuring that acceptable criticality safety margin and decay heat removal capabilities, as well as fuel storage efficiency, are maintained in the spent fuel pools.

   c. Specification 3.8.17 provides the action statements associated with Specifications 3.8.15 and 3.8.16. These actions have been added to address the SFP boron concentration required by Specification 3.8.15 and the loading restrictions required by Specification 3.8.16. These actons are consistent with STS language.

5. The Bases for Specifications 3.8.15 and 3.8.16 have been updated to include the basis for the changes in these specifications. Additional Bases explain the acceptable use of non-fuel components, or empty cells, in place of designated fuel assembly locations to provide additional operational flexibility while maintaining acceptable criticality safety margin. The maximum initial fuel enrichment, 5.00 weight%, is the basis for all fuel storage requirements imposed by Specification 3.8.16. The additional bases provide linear interpolation as an acceptable method for obtaining additional data.

6. Specification 5.3.1 is revised to accommodate changes in the fuel assembly design. Specifically, the core loading in metric tons of uranium dioxide and the fuel stack height will be different for the new fuel assembly designs. To address these changes, Specifications 5.3.1.2 and 5.3.1.4 are deleted and Specification 5.3.1.1 is revised to reflect the wording in the STS. In addition, Specification 5.3.1.3 is renumbered to 5.3.1.2. These changes result in a proposed Design Features Specification for the Reactor Core that is consistent with the STS.

7. The title for Specification 5.4 is being changed from 'New and Spent Fuel Storage Facilities' to 'Fuel Storage' to be consistent with STS. This change is administrative.

8. Specification 5.4 is being changed to bring this specification more in line with the STS format.

   a. References to where new and spent fuel may be stored is being removed. This information is implied by the specification and is specified in the FSAR.

   b. The reference to fuel enrichment is being deleted from Specification 5.4.1.1. This information is being relocated to the Bases for Specifications 4.8.

   c. Additional extraneous information in Specification 5.4.2.1 is being deleted. This information is either provided in the FSAR, or is no longer applicable. This change also brings this specification more in line with STS format.

d. Specification 5.4.3, which specifies the SFP boron concentration, is being deleted. This requirement is being relocated to Specification 3.8.15. This is a more appropriate location for this limit and is also consistent with STS.

e. Specification 5.4.4, which provides the earthquake force assumed in the seismic analysis, is being deleted. This information is currently provided in the FSAR. This change is consistent with STS.

9. Specification 6.9.1 is being changed to include the Spent Fuel Pool boron concentration in the list of limits included in the Core Operating Limits Report (COLR). This change is a necessary administrative change due to the change described in number 4.a above.

The above described Technical Specification changes are necessary to accommodate the new Mark B10T and Mark B11 fuel assembly designs. Full batch implementation of the Mark B10T design will begin with Oconee 3 Cycle 16. The Mark B10T design is identical to the current MarkB10 assembly with the exception of the fuel rod design. The fuel rod design is altered to increase the uranium loading of the fuel assembly from 463.6 kg to 487 kg. This loading increase is accomplished by altering the stack height, reducing the thickness of the cladding, reducing the gap size, increasing the theoretical density and changing from a truncated cone to spherical dish on the ends of the fuel pellets. Detailed information on the new design is contained in Section 4 of the Oconee 3 Cycle 16 Reload Report. The reload report is included in the submittal as Attachment 4.

The Mark B11 fuel assembly design is scheduled for full batch implementation at ONS beginning with Oconee 3 Cycle 19. However, four lead test assemblies (LTAs) will be inserted in Oconee 1 Cycle 17. The proposed Technical Specification changes also address spent fuel pool enrichment limits for this design. Key features of the Mark B11 design are a reduction in the cladding outside diameter from 0.430 inches to 0.416 inches, a uranium loading of approximately 459 kg, and implementation of mixing vane grids. A comprehensive testing program, including incore irradiation of four LTAs, will be completed prior to full batch implementation of this design. A detailed submittal will be prepared for NRC review prior to full batch implementation of the Mark B11 design. However, in order to store the four LTAs in the spent fuel pool, proposed Specification 3.8.16 also addresses the Mark B11 design. This approach was agreed upon in an August 1, 1994 conference call with the NRC in that it avoids an additional submittal to specifically address the SFP storage restrictions for the Mark B11 design prior to shipment of the LTAs in September of 1995.

The Oconee 3 Cycle 16 Reload Report (Attachment 4) justifies the operation of Oconee 3 Cycle 16 at the rated core power of 2568 Mwth. Included are the required analyses as outlined in the US NRC document "Guidance for Proposed License Amendments Relating to Refueling," June 1975. Oconee 3 Cycle 16 was designed using the NRC approved methods delineated in Specification 6.9.2.

The Oconee 3 Cycle 16 core will have 60 fresh Mark B10T assemblies (Batch 18) enriched to 3.36 wt% U-235 with six inch axial blankets, top and bottom, enriched to 2.00 wt% U-235. Forty-four burnable poison rod assemblies (BPRAs) are inserted in the Batch 18 fuel for additional reactivity control. The acceptability of the fuel system design, nuclear design, and thermal-hydraulic design of Oconee 3 Cycle 16 are addressed in Sections 4, 5, and 6, respectively, of the Reload Report. Section 5 of the Reload Report compares the physics parameters for Cycle 16 to those of Cycle 15. Although the uranium loading of the Mark B10T is slightly higher than in previous fuel assembly designs, the overall operating characteristics of Oconee 3 Cycle 16 are very similar to previous designs.

Section 7 of the Reload Report evaluates the impact of the Oconee 3 Cycle 16 design on the accident analyses of the Oconee FSAR. This evaluation confirms that the FSAR licensing basis analyses remain bounding. As is described in Section 7 of the Reload Report, B&W has performed LOCA analyses for the Mark B10T fuel assembly design using their approved Evaluation Model. These analyses demonstrate that the LOCA linear heat rate limits for the Mark B9 fuel assembly design bound the linear heat rate limits for the Mark B10T fuel assembly design. The design of Oconee Unit 3 Cycle 16 conservatively applied the Mark B9 LOCA linear heat rate limits to the Mark B10T fuel assemblies. Operation within the LOCA linear heat rate limits given in Section 7 of the Reload Report assures that the final acceptance criteria of 10CFR 50.46 will not be violated. In summary, the analyses provided in the Oconee 3 Cycle 16 Reload Report demonstrate that the Oconee 3 Cycle 16 reload design satisfies all acceptance criteria.

Attachment 5 provides the technical justification for the fuel enrichment upgrade. Specification 3.8.16 establishes restricted loading patterns and associated burnup criteria for placement of new and irradiated fuel into the Oconee spent fuel storage pools. The analyses in Attachment 5 demonstrate that the use of these configurations for storing fuel with initial enrichments of up to 5.0 w/o U-235 will maintain sufficient criticality margins. Specifically, the analyses assure that there is a 95% probability at a 95% confidence level that the effective multiplication factor ($k_{eff}$) of the fuel assembly array will be less than or equal to 0.95. The calculated $k_{eff}$ includes all appropriate biases and uncertainties. Specification 3.8.15 establishes a minimum allowable boron concentration in the spent fuel pool to assure that the consequences of a fuel misloading accident maintain the pool configuration at or below an acceptable $k_{eff}$ of 0.95. In summary, Attachment 5 provides the technical justification to support storing the Mark B11 fuel assembly design and Mark B10T and earlier fuel assembly designs in the SFP in accordance with proposed Specifications 3.8.15 and 3.8.16.