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March 1, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
License Amendment Request to Reconcile 10 CFR 50 and 10 CFR 72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool
License Amendment Request No. 2005-009

Reference: NRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," dated March 23, 2005.

In accordance with 10 CFR 50.90, Duke Energy Corporation (Duke) proposes to amend Renewed Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55. If granted, this amendment request will allow spent fuel loading, unloading, and handling operations in the Oconee Nuclear Site (Oconee) Spent Fuel Pools (SFP) that support spent fuel transfer to an Independent Spent Fuel Storage Installation (ISFSI) licensed under 10 CFR 72.

As the Nuclear Regulatory Commission (NRC) noted in the Reference, there are differences in the criticality requirements of 10 CFR 50 for SFPs and 10 CFR 72 for ISFSIs. Duke has completed a review of the Oconee 10 CFR 72 licensing basis for the NUHOMS®-24PHB and the NUHOMS®-24P storage systems and concluded they do not meet the requirements of 10 CFR 50.68(b)(1) during loading and unloading operations in the Oconee SFP.

To demonstrate acceptable subcriticality margins for cask loading and unloading operations in accordance with 10 CFR 50 requirements, Duke proposes to revise applicable sections of the Oconee Technical Specifications (TS), TS Bases, and Updated Final Safety Analysis Report (UFSAR) to incorporate changes based on the results of a new criticality analysis. The new criticality limits for spent fuel dry storage casks are derived from a methodology that is very similar to that approved by the NRC for the McGuire Nuclear Site spent fuel storage racks. Results of this new analysis show that application of the new and revised TS and TS Bases will assure that there is acceptable subcriticality margin for cask loading and unloading operations in both Oconee SFP. The changes proposed in this amendment request will not affect the 10 CFR

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72 license, but will reconcile the differences in the 10 CFR 50 and 10 CFR 72 licenses and are hereby proposed for NRC approval.

At the meeting held between Duke and the NRC staff on November 1, 2005, the NRC indicated that Duke should address 10 CFR 50.68(b) in this submittal. Accordingly, Enclosure 4 describes how Oconee complies with 10 CFR 50.68(b). Duke will also update applicable sections of the Oconee UFSAR and submit these changes per 10 CFR 50.71(e).

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the Plant Operations Review Committee and Nuclear Safety Review Board. Additionally, a copy of this license amendment request is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements. In order to support required ISFSI transfers in Summer 2006, Duke respectfully requests that the amendment be issued by June 1, 2006, with a 90-day implementation period from the date of issuance.

Inquiries on this proposed amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 885-3364.

Sincerely,

Bruce Hamilton

B. H. Hamilton, Vice President
Oconee Nuclear Site

Enclosures:
1. Notarized Affidavit
2. Evaluation of Proposed Change
3. Oconee Nuclear Site - NUHOMS®-24P/24PHB DSC Criticality Analysis
4. Compliance with 10 CFR 50.68(b)

Attachments:
1. Technical Specifications – Mark Ups
2. Technical Specifications - Reprinted Pages
3. List of Regulatory Commitments

bc w/enclosures and attachments:

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ENCLOSURE 1

NOTARIZED AFFIDAVIT

AFFIDAVIT

B. H. Hamilton, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Bruce Hamilton

B. H. Hamilton, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 1st day of March, 2006

Sheila A Smith

Notary Public

My Commission Expires:

6/12/2013

Date

SEAL

ENCLOSURE 2

EVALUATION OF PROPOSED CHANGE

Subject: License Amendment Request to Reconcile 10 CFR 50 and 10 CFR 72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool

1.0 DESCRIPTION

Oconee Nuclear Site (Oconee) currently stores spent fuel assemblies in its Spent Fuel Pools (SFPs) and at its Independent Spent Fuel Storage Installation (ISFSI). These storage locations have been previously licensed utilizing criteria from different sections of the code of federal regulations (CFR). In the spent fuel assembly transfer process (from underwater pool to dry cask storage), there are instances where regulatory requirements of both the facility operating licenses (FOLs) and ISFSI licenses apply; however, as described in Regulatory Issue Summary (RIS) 2005-05, there are notable differences in the criticality requirements of 10 CFR 50 (for SFPs) and 10 CFR 72 (for ISFSIs). Duke Energy Corporation (Duke) has completed a review of the 10 CFR 72 licensing bases for the NUHOMS®-24PHB and the NUHOMS®-24P storage systems and concluded they do not meet the requirements of 10 CFR 50.68(b)(1) during cask loading and unloading operations in the Oconee SFP.

2.0 PROPOSED CHANGE

Duke proposes to revise applicable sections of the Oconee Units 1, 2, and 3 Technical Specifications (TS) and TS Bases to establish boron concentration limits for spent fuel cask loading and unloading operations and to restrict the burnup of spent fuel assemblies that can be loaded into a spent fuel storage cask while in a SFP. Specifically, TS 3.7.12, "Spent Fuel Boron Concentration," will be revised to add cask loading and unloading operations to the current SFP application and a new TS 3.7.18, "Dry Spent Fuel Storage Cask Loading and Unloading" will be added to address burnup limits for fuel assemblies loaded into casks while in a SFP. In addition, TS Section 4.0, "Design Features," will be revised to include spent fuel storage cask loading and unloading operations, and associated TS Bases will be revised or added as necessary. These changes are needed to ensure the requirements of 10 CFR 50.68(b)(1) are met when loading and unloading the NUHOMS® dry storage canisters (DSCs) at Oconee.

3.0 BACKGROUND

Oconee uses the NUHOMS® dry spent fuel storage system at its ISFSI. Forty NUHOMS®-24P DSCs have been loaded under Duke's specific license (SNM-2503). Another forty-four NUHOMS®-24P DSCs have been loaded under Duke's general license. Certificate of Compliance (CoC) 72-1004 is applicable to the DSCs loaded under the general license. For future loadings, Oconee will use the NUHOMS®-24PHB which has been approved by the NRC as Amendment 6 to CoC 72-1004.

The minimum dissolved boron concentration for the SFP at Oconee is provided in TS 3.7.12 for the Renewed Facility Operating License. TS Surveillance Requirement (SR)

3.7.12.1 requires verification that the pool boron concentration is within the limits of the Core Operations Limits Report (COLR) and greater than or equal to 2,220 ppm.

4.0 TECHNICAL ANALYSIS

The criticality analysis[1] of the NUHOMS®-24P/24PHB DSC, for loading and unloading operations in the Oconee SFPs, has been performed in accordance with the requirements of 10 CFR 50.68(b). This evaluation takes partial credit for soluble boron in the SFPs. Minimum burnup requirements were developed for fuel to be placed without location restrictions in the NUHOMS®-24P/24PHB DSC. These burnup requirements, applicable for eligible fuel assemblies with a minimum 5 years post-irradiation cooling time, are a function of initial U-235 enrichment.

In the DSC criticality analysis, the maximum 95/95 k_{eff} with no boron in the DSC submerged in the Oconee SFP was calculated to be 0.9980. The criticality evaluation also confirmed that with 430 ppm of partial soluble boron credit, the maximum 95/95 k_{eff} of 0.9264 remains well below the regulatory requirement that the maximum 95/95 k_{eff} be less than 0.95 for all normal conditions. Additionally, the criticality analysis demonstrated that the current minimum boron concentration required in the Oconee SFPs (2220 ppm) is adequate to maintain the maximum 95/95 k_{eff} below 0.95 for all credible accident scenarios associated with loading and unloading fuel assemblies into the NUHOMS®-24P/24PHB DSCs.

5.0 REGULATORY SAFETY ANALYSIS

5.1 <u>No Significant Hazards Consideration</u>

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The applicable accidents are the dropped fuel assembly and drop of the 100 ton spent fuel cask into the SFP. This amendment request does not change the fuel assemblies or any of the Part 50 structures, systems, or components

[1] Reference Enclosure 3

involved in fuel assembly or cask handling or any of the operations involved. Therefore, this amendment request does not affect the probability of an accident previously evaluated.

The proposed change does not increase the consequences of an accident previously evaluated for the following reasons: there is no increase in radiological source terms for the fuel; there is no change to the SFP water level; subcriticality is maintained for normal and accident conditions for the spent fuel storage racks and for cask loading and unloading; and the same boron concentrations that were previously credited for the spent fuel storage racks are assumed in the criticality analysis for cask loading and unloading.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

Handling of fuel assemblies and the NUHOMS® spent fuel cask have been previously evaluated for Oconee. These activities lead to evaluation of the fuel handling accident (dropped fuel assembly) and drop of the 100 ton spent fuel cask onto spent fuel stored in the Oconee SFP. These elements of the license amendment request (LAR) are not new, and thus do not create the potential for new or different kinds of accidents.

The new element of this LAR is the application of additional criticality controls (i.e., minimum burnup requirements for the fuel assemblies) beyond the 10 CFR 72 controls already in place for the NUHOMS® spent fuel cask. However, application of such criticality controls is not a new activity for Oconee, since similar criticality controls are currently applied to the spent fuel storage racks. Fuel assembly misloading is not a new accident; as discussed in Enclosure 3, Section 6.5, fuel assembly misloading has been considered previously for the NUHOMS® spent fuel cask and for the Oconee spent fuel pool racks. Furthermore, the criticality analysis for cask loading and unloading evaluates the same boron concentrations, moderator temperatures, and misloading scenario as previously evaluated for the spent fuel storage racks. The analysis demonstrates that a criticality accident does not occur under these conditions. It is concluded that the possibility of a criticality accident is not created since application of criticality controls is not new and the analysis demonstrates that criticality does not occur. More generally, this supports the conclusion that the potential for new or different kinds of accidents is not created.

3) Involve a significant reduction in a margin of safety.

This LAR involves the application of additional criticality controls (minimum burnup requirements) to the 10 CFR 72 controls already in place for the NUHOMS® spent fuel cask. The criticality analysis demonstrates subcriticality margins are maintained for normal and accident conditions consistent with 10 CFR 50.68(b) and other NRC guidance. Margins previously established for Oconee's spent fuel storage racks are not altered. Therefore, this LAR does not result in a reduction in a margin of safety.

5.2 Applicable Regulatory Requirements/Criteria

5.2.1 Spent Fuel Pool Storage (10 CFR 50):

As described in UFSAR Section 9.1.2.3.2, "Criticality Analysis," criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, considering possible variations, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than or equal to 0.95, with partial credit for soluble boron.

6.0 ENVIRONMENTAL CONSIDERATION

Duke has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 5.1, this proposed amendment does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

Additional criticality safety requirements do not impact effluents. Therefore, there will be no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

Additional criticality safety requirements do not impact individual or cumulative occupational radiation exposure. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

ENCLOSURE 3

OCONEE NUCLEAR SITE
NUHOMS®-24P/24PHB DSC CRITICALITY ANALYSIS

1 Introduction

This analysis examines the criticality aspects of fuel storage in the NUHOMS®-24P and NUHOMS®-24PHB dry storage canisters (DSCs), during loading and unloading operations in the spent fuel pools (SFPs) at Oconee Nuclear Site (Oconee). The analysis is intended to address the concerns documented by the Nuclear Regulatory Commission in Regulatory Issue Summary 2005-05 (Reference 1).

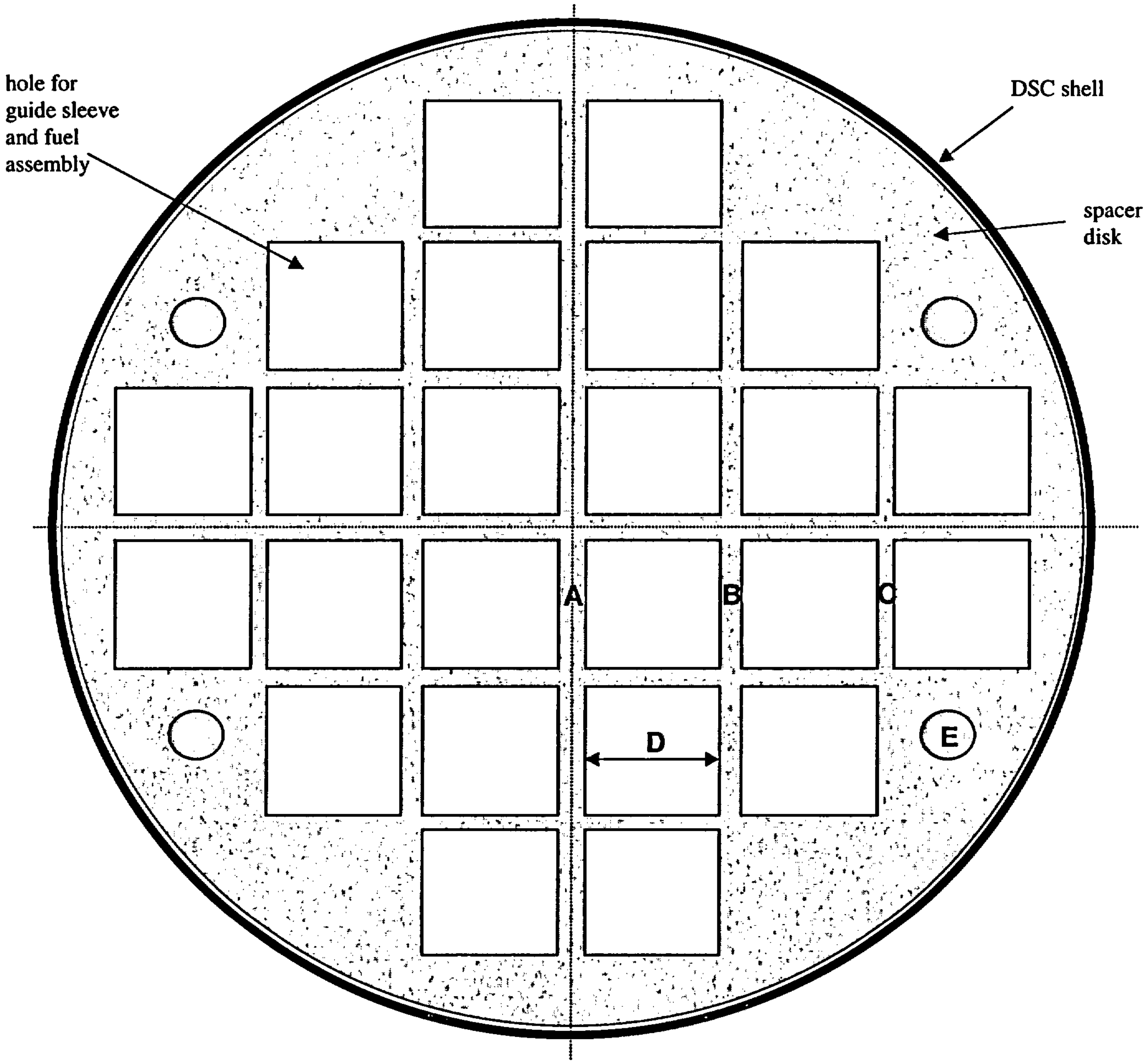
The objective of the DSC criticality evaluation is to demonstrate that eligible fuel assemblies enriched up to 5.0 wt % U-235 may be placed without location restrictions in the DSCs during loading/unloading operations in the Oconee SFPs, if specific requirements for minimum burnup, fuel assembly design, and cooling time are met.

The DSC criticality analysis examines the NUHOMS® systems in use at Oconee, to show that the placement of irradiated fuel assemblies in these DSCs complies with the requirements of 10 CFR 50.68(b). In accordance with that regulation, the DSC criticality evaluation must show subcriticality in unborated water, but may take partial credit for soluble boron in the SFP water to achieve a k_{eff} less than 0.95. The current licensing basis for fuel assembly storage in the Oconee SFPs (Reference 2) allows 430 ppm soluble boron credit for normal conditions.

2 NUHOMS®-24P/24PHB DSC Description

Three variations of the NUHOMS®-24P DSC design have been or are planned to be employed at Oconee. These include the NUHOMS®-24P (site-specific license), the NUHOMS®-24P (general license), and the NUHOMS®-24PHB (general license). Table 1 provides the nominal design data for the NUHOMS®-24P/24PHB DSC components. Note that the only significant difference among these DSC designs, from a criticality modeling perspective, is with the guide sleeves, which are full-axial length stainless steel storage cells in the DSCs. The NUHOMS®-24P (site-specific license) design has a reduced thickness for twelve of its outer guide sleeves as compared with the other two DSC designs. This design difference is evaluated in Section 6.3.

Figure 1 depicts a cross-sectional slice of the NUHOMS®-24P/24PHB DSC, through one of the 2-inch-thick spacer disks that hold the guide sleeves in place. During loading/unloading operations, the DSC sits inside of a transfer cask comprising several inches of stainless steel, lead, carbon steel, and a cementatious neutron poison material. The transfer cask effectively isolates the fuel assemblies placed in the DSC from any neighboring fuel stored in the Oconee SFP racks. DSC loading/unloading takes place in the cask pit area of the SFP. The cask pit is adjacent to the spent fuel storage racks in each of the Oconee SFPs, and is open to the rest of the SFP at all times.



Ligament Thickness **A** = 1.25 in
Ligament Thickness **B** = 1.00 in
Ligament Thickness **C** = 0.75 in
Spacer Disk Hole Width **D** = 9.28 in (square)
Support Rod Diameter **E** = 3.25 in

Figure 1. Oconee NUHOMS®-24P/24PHB DSC Radial Geometry (Spacer Disk Detail)

Table 1. Design Data for NUHOMS®-24P and -24PHB DSCs and Transfer Cask

Parameter	Nominal Dimension (inches)
Stainless Steel (SS) Guide Sleeve ID	8.90
Guide Sleeve Thickness	0.105*
SS Support Rod OD	3.25
Support Rod (x,y) coordinates	23.95, 13.92
Ligament (Spacing) Thicknesses	1.25, 1.00, 0.75
Spacer Disk Hole ID	9.28 (square)
Distances to Guide Sleeve Hole Centers from DSC radial origin (x or y coordinates)	5.265, 15.545, 25.575
Spacer Disk OD	65.50
Axial distance between Spacer Disks	21.1
SS Spacer Disk Thickness	2
DSC SS Shell Thickness	0.625
DSC Shell OD	67.19
Transfer Cask SS inner shell ID	68.00
Transfer Cask Pb shield ID	69.00
Transfer Cask Carbon Steel Support Shell ID	76.00
Transfer Cask Neutron Shield ID	79.00
Transfer Cask SS outer shell ID	85.00
Transfer Cask SS outer shell OD	85.75

* -- note that the NUHOMS®-24P site-specific design originally used at Oconee had a reduced guide sleeve thickness (0.06 inches) for twelve (12) of its outer guide sleeves

3 Fuel Assembly Designs Considered

The following Babcock & Wilcox (B&W) 15x15 fuel types that have been employed at Oconee are eligible to be loaded in the NUHOMS®-24P or NUHOMS®-24PHB DSCs: MkB2, MkB3, MkB4, MkB4Z, MkB5, MkB5Z, MkB6, MkB7, MkB8, MkB9, MkB10D, MkB10E, MkB10F, MkB10G, MkB10L. Though other fuel assembly designs have been irradiated in the Oconee reactors, they are not allowed to be loaded in the NUHOMS®-24P or NUHOMS®-24PHB DSCs – in accordance with Reference 3 – and thus are not considered in this analysis. Each of the eligible fuel assembly types can be classified as one of three bounding (in terms of criticality parameters) generic designs with the following shorthand names: **mbz**, **mbf**, and **mbl**. The important design data for these generic fuel types are listed in Table 2.

To maximize fuel assembly k_{eff}, it is assumed that irradiated fuel assemblies to be placed in the NUHOMS®-24P/24PHB DSCs contained discrete Al_2O_3-B_4C burnable poison rod assemblies (BPRAs) with the highest feasible B_4C loading during operation in the Oconee reactors. The Reference 4 analysis shows that higher BPRA boron concentrations yield greater fuel assembly k_{eff}s once the BPRAs are removed from the assemblies after reactor irradiation.

The design data for the bounding Oconee BPRAs are provided in Table 3.

Table 2. Design Data for Generic Fuel Assembly "Types" Eligible for Storage in NUHOMS®-24P/24PHB DSCs

Parameter	mbz	mbf	mbl
Average UO_2 fuel density (g/cc)	10.28	10.21	10.53
Fuel Pellet OD (inches)	0.3686	0.3700	0.3735
Cladding ID (inches)	0.377	0.377	0.380
Cladding OD (inches)	0.430	0.430	0.430
Cladding Material	Zircaloy	Zircaloy	Zircaloy
Fuel Pin Pitch (inches)	0.568	0.568	0.568
Fuel Pin Array Size	15x15	15x15	15x15
Guide Tube ID (inches)	0.498	0.498	0.498
Guide Tube OD (inches)	0.530	0.530	0.530
Specific 15x15 Fuel Designs Represented	MkB4Z, MkB5, MkB5Z, MkB6, MkB7, MkB8	MkB2, MkB3, MkB4, MkB9, MkB10D, MkB10E	MkB10F, MkB10G, MkB10L

Table 3. Design Data for Bounding BPRAs Inserted into Fuel Assemblies During Oconee Reactor Irradiation

Parameter	Value
Poison Pellet Density (g/cc)	3.38
Poison Pellet Diameter (inches)	0.340
B_{10} concentration (wt %)	0.5726
B_{11} concentration (wt %)	2.5588
C concentration (wt %)	0.8686
Al concentration (wt %)	50.807
O concentration (wt %)	45.193
Number of rodlets (fingers) in BPRA	16

4 Criticality Computer Code Validation

The main neutronics codes employed in the NUHOMS®-24P/24PHB DSC criticality analysis are SCALE 4.4/KENO V.a and CASMO-3/SIMULATE-3. These codes are well-suited to wet fuel storage criticality applications, and have been extensively benchmarked to critical experiments and reactor operational data. KENO V.a is a 3-D Monte Carlo criticality module in the SCALE 4.4 (Reference 5) package. CASMO-3 (Reference 6) is a 2-D transport code that performs fuel criticality and depletion calculations, using a 70-group cross-section library that is based on ENDF/B-IV. CASMO-3 also produces nodal macro-group cross-sections that can be used by SIMULATE-3 (Reference 7), its counterpart 3-D nodal diffusion code, for applications involving arrays of fuel assemblies with varying enrichments or burnups.

SCALE 4.4/KENO V.a is used for explicit evaluation of the 3-D NUHOMS®-24P/24PHB DSC described in Section 2. This analysis involves unirradiated fuel assemblies loaded in the DSC. The SCALE 4.4/KENO V.a computations are performed to confirm the conservatism of a simplified uniform-array DSC model, and are described and documented in Section 6.3.

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

The following subsections discuss the benchmarking validation that has been performed for both SCALE 4.4/KENO V.a and CASMO-3/SIMULATE-3. Note that the same code benchmarking results were employed in the McGuire SFP amendment request approved per Reference 16. Given the types of critical experiments with which these code systems have been validated (low-enriched uranium fuel rod lattices with configurations similar to those of fuel assemblies in SFP storage), the use of these code packages is appropriate for the NUHOMS®-24P/24PHB DSC criticality evaluations.

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

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

For NUHOMS®-24P/24PHB DSC criticality applications, the SCALE 4.4/KENO V.a benchmark biases and uncertainties are based on 58 critical experiments carried out by Pacific Northwest Laboratories (see References 8 to 10). The critical experiments evaluated cover a wide range of enrichment (2.35 and 4.31 wt % U-235), and include both over- and under-moderated lattices.

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

The 58 experiments used in the benchmarking resulted in a calculational bias of **+0.0064** Δk and an uncertainty of **±0.0066** Δk. These biases and uncertainties will be used in determining the total bounding 95/95 system k_{eff}s for each DSC configuration modeled with SCALE 4.4/KENO V.a.

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

For all of the irradiated-fuel criticality evaluations for the NUHOMS®-24P/24PHB DSC, the CASMO-3/SIMULATE-3 code set is used. All CASMO-3 calculations will be carried out with the fine-energy-group (70-group) neutron cross-section library available with that code. Duke Power has performed a benchmark analysis of 10 B&W critical experiments with CASMO-3 and SIMULATE-3. These B&W critical experiments (Reference 11) were specifically designed for reactivity benchmarking purposes. Results from the 10 B&W critical benchmark cases yielded a calculational bias of **-.0015** Δk (average over-prediction of k_{eff}) and an uncertainty of **±0.0121** Δk. Even though CASMO-3/SIMULATE-3 tends to over-predict k_{eff}, the negative bias will be conservatively ignored. The uncertainty, however, will still be used in computing the overall 95/95 k_{eff}s for the DSC irradiated-fuel storage cases described in Section 6.5.

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

For each fuel assembly design, enrichment, and burnup combination that is considered in the scope of the NUHOMS®-24P/24PHB DSC criticality analyses, a nominal k_{eff} is calculated. This k_{eff} is only the base value, however. A total k_{eff} is determined by adding several pertinent reactivity biases and uncertainties, to provide an overall 95 percent probability, at a 95 percent confidence level (95/95), that the true system k_{eff} does not exceed the 95/95 k_{eff} for that particular storage condition.

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

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

where:

$k_{nominal}$	is the k_{eff} computed for the nominal case being considered.
B_x	is a pertinent bias, as indicated in Table 4.
ks_x	is the pertinent 95/95 independent uncertainty on $k_{nominal}$, as indicated in Table 4.

Table 4 lists the various biases and uncertainties that are considered in the NUHOMS®-24P/24PHB DSC criticality analyses. Each of these biases and uncertainties is discussed in more detail below:

- **Benchmark Method Bias**

  This bias is determined from the benchmarking of the code system used (SCALE 4.4/KENO V.a or CASMO-3/SIMULATE-3), and represents how much the code system is expected to overpredict (negative bias) or underpredict (positive bias) the "true k_{eff}" of the physical system being modeled. The critical experiment benchmarks for these codes are discussed in Sections 4.1 and 4.2. The bias for SCALE 4.4/KENO V.a with its 238-group cross-section library is **+0.0064** Δk. The bias for CASMO-3/ SIMULATE-3 with its 70-group cross-section library is **-0.0015** Δk. Note that negative biases are conservatively ignored in this calculation, as recommended in Reference 12.

- **Axial Burnup Bias**

  Section 6.4 discusses the method for determining the reactivity bias associated with the difference between the system k_{eff} calculated using an average 2-D fuel assembly burnup, and the k_{eff} using the 3-D axial burnup distribution for that assembly. Section 6.4 analyzes these k_{eff} differences for a bounding set of fuel assemblies and calculates an axial burnup bias as a linear function of assembly-average burnup. This bias is of the form:

  $$\Delta k = \mathbf{0.00105912*BU - 0.02189}$$

  where BU is assembly-average burnup, in gigawatt-days/tonne uranium (GWD/MTU). Any calculated biases that are negative are conservatively ignored.

- **Benchmark Method Uncertainty**

  This uncertainty is determined from the benchmarking of the code system used (SCALE 4.4/KENO V.a or CASMO-3/SIMULATE-3), and is a measure of the expected variance (95/95 one-sided uncertainty) of predicted reactivity from the "true k_{eff}" of the physical system being modeled. The critical experiment benchmarks for these codes are discussed in Sections 4.1 and 4.2. The method uncertainty for SCALE 4.4/KENO V.a with its 238-group cross-section library is **±0.0066** Δk. The uncertainty for CASMO-3/SIMULATE-3, with its 70-group cross-section library, is **±0.0121** Δk.

- **Monte Carlo Computational Uncertainty**

  For the SCALE 4.4/KENO V.a computations performed in this analysis to determine 95/95 k_{eff}s, the Monte Carlo computational uncertainty is equal to $1.727*\sigma_{nominal}$. The $\sigma_{nominal}$ factor is the calculated standard deviation of $k_{nominal}$ (the nominal k_{eff} for that particular case). The 1.727 multiplier is the one-sided 95/95 tolerance factor for 1000 neutron generations (Reference 18). Each of the SCALE 4.4/KENO V.a cases modeled in the DSC criticality analysis counted 1000 neutron generations.

- **Mechanical Uncertainties**

  The "mechanical uncertainties" represent the total reactivity contributions of various independent DSC-related and fuel manufacturing-related uncertainties. These include reactivity effects for possible variations in fuel enrichment (±0.05 wt % U-235), fuel pellet diameter, fuel density, cladding dimensions, DSC guide sleeve thickness, DSC cell center-to-center spacing, and fuel assembly positioning within the DSC guide sleeve. The following bounding total mechanical uncertainties have been determined for the NUHOMS®-24P/24PHB DSC criticality analyses:

  - **±0.0280** Δk (no boron in SFP water)
  - **±0.0304** Δk (430 ppm boron in SFP water)

- **Burnup Computational Uncertainty**

  This burnup-related uncertainty represents the ability of the CASMO-3/SIMULATE-3 codes to accurately determine the isotopic content, and hence k_{eff}, of irradiated fuel assemblies.

  As a conservative alternative to determining this uncertainty directly, Reference 12 notes that "... In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." This approach is used for the burnup credit cases evaluated in Section 6.5, and the largest uncertainty calculated is applied to the total 95/95 k_{eff} for all assemblies loaded into the DSCs that have minimum burnup requirements. The maximum burnup computational uncertainty determined in Section 6.5 is **±0.0151 Δk.**

- **Burnup Measurement Uncertainty**

  This uncertainty represents the reactivity penalty associated with the difference between measured burnup and actual burnup. Measured burnups, which are used for Technical Specification verification, have various sources of instrumentation error that can contribute to overall measurement inaccuracies. Comparison of measured burnup data to core follow predicted burnups for a large sample of discharged Oconee fuel assemblies shows that a four (4) percent burnup measurement uncertainty is conservative. Note that Reference 17 suggests that a two (2) percent measurement uncertainty may be sufficient for pressurized-water reactor (PWR) fuel.

  The largest Δk penalty associated with a 4 percent uncertainty among the burnup credit cases considered in Section 6.5 is applied to the total 95/95 k_{eff} computations for fuel requiring any amount of burnup. The bounding burnup measurement uncertainty calculated in this manner is **±0.0104 Δk**.

Table 4. Pertinent 95/95 Biases and Uncertainties to be Considered in the NUHOMS®-24P/24PHB DSC Criticality Analysis

Biases	**Include for SCALE 4.4/ KENO V.a Calculations?**	**Include for CASMO-3/ SIMULATE-3 Calculations?**
Benchmark Method Bias	✓	✓
Axial Burnup Bias		✓
Uncertainties		
Benchmark Method Uncertainty	✓	✓
Monte Carlo Computational Uncertainty	✓	
Mechanical Uncertainties	✓	✓
Burnup Computational Uncertainty		✓
Burnup Measurement Uncertainty		✓

6 NUHOMS®-24P/24PHB Dry Storage Canister Criticality Analysis

6.1 General Analysis Requirements

In order to address the concerns documented in RIS 2005-05 (Reference 1) for the placement of fuel assemblies in the NUHOMS®-24P/24PHB DSCs, the following requirements of 10 CFR 50.68(b) are satisfied:

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

In addition, for evaluations of irradiated fuel, Reference 12 provides the following general criteria:

> "A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties."
>
> "A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution."

6.2 DSC Criticality Analysis Assumptions / Bases

The following assumptions and bases are employed for the NUHOMS®-24P/24PHB DSC criticality evaluations:

1) The main DSC criticality calculations with irradiated fuel are performed in two dimensions, using the CASMO-3 transport code. The conservative infinite-array DSC model described in Section 6.3 is used. Separate three-dimensional computations are performed with the SIMULATE-3 nodal code to determine appropriate axial burnup biases to apply to the two-dimensional CASMO-3 computational results – see Section 6.4. The 3-D SIMULATE-3 model includes 23 axial fuel zones, along with top and bottom axial reflectors containing a mix of water, steel, and Zircaloy. Reference 13 supports the assumption that using 23 axial fuel segments is sufficient to accurately capture the reactivity effects associated with axial variations in fuel burnup. Extensive historic core-follow axial burnup predictions are employed to determine a conservative axial burnup bias for use in the total 95/95 DSC k_{eff} calculations.

2) No credit is taken for any short-lived Xe-135 poisons in the fuel assemblies loaded in the DSCs, consistent with Reference 12.

3) No credit is taken for fuel assembly spacer grids.

4) No credit is taken for any BPRAs in the fuel assemblies loaded in the DSCs. As with spacer grids, even depleted BPRAs act as a modest poison in unborated or low-borated SFP water.

5) In order to ensure the most conservative isotopic content, and hence, k_{eff}, for irradiated fuel assemblies to be loaded into the NUHOMS®-24P/24PHB DSCs, conservative depletion parameters are used for the Oconee reactor fuel burnup computations performed by CASMO-3. These parameters include a high average fuel temperature (1054 °F), high outlet moderator temperature (630 °F), high cycle-average soluble boron concentration (700 ppm), and one-cycle maximum BPRA exposure (25 GWD/MTU).

6) Credit for the reactivity reduction associated with fuel burnup and 5 years of post-irradiation cooling time is employed for the DSC criticality analysis. Reactivity reduction with cooling time is primarily attributable to Pu-241 decay (~14.3 yr half-life), and Gd-155 buildup (via Eu-155 decay with ~ 4.7 yr half-life).

7) Partial soluble boron credit of 430 ppm in the Oconee SFP is taken in order to achieve a system $k_{eff} \leq 0.95$. This is in accordance with the regulatory subcriticality criteria defined in 10 CFR 50.68(b), as well as the guidance provided in Reference 12. The 430 ppm boron credit is the amount allowed for normal conditions in the current Oconee SFP criticality licensing basis (Reference 2).

8) For accident conditions in the DSC, the minimum Oconee SFP boron concentration, as specified in the Core Operating Limits Report and Technical Specification 3.7.12 (2220 ppm), is available. Per the double contingency principle (see Reference 12), it is allowable to assume that the minimum boron concentration is present in the event of an accident condition – such as a misloaded fuel assembly – in the DSC.

The assumptions and bases listed above indicate that the criticality analysis of the NUHOMS®-24P/24PHB DSC is extremely conservative. The major sources of the large quantity of Δk margin include the simplified infinite-array DSC model, the mechanical and burnup-related uncertainties discussed in Section 5, the in-reactor depletion parameters, and the axial burnup bias. The combination of these conservatisms amounts to more than 0.05 Δk.

6.3 DSC Model Simplification for Burnup Credit Computations

Note that the NUHOMS®-24P/24PHB DSC, as depicted in Figure 1, has a variable radial spacing between fuel assembly storage positions. While this variable spacing is straightforward to model with SCALE 4.4/KENO V.a, it is not really feasible using the CASMO-3/SIMULATE-3 code system.

To allow CASMO-3/SIMULATE-3 to be used for burnup credit criticality analyses of the NUHOMS®-24P/24PHB DSC, a conservative model simplification is employed, with the following features:

- Infinite array of fuel assemblies (perfect radial reflection)
- Uniform storage cell pitch (10.28 inches, corresponding to intermediate spacing "B" in Figure 1)
- Guide sleeves included in model (0.105-inch thickness per Table 1)
- Elimination of all other DSC structural material, including spacer disks, support rods, DSC shell, and surrounding transfer cask

Figure 2 illustrates the proposed infinite-array DSC model. To demonstrate the conservatism of this model for subsequent criticality evaluations, k_{eff} calculations are performed for both the full-detail radial DSC model in Figure 1 (using SCALE 4.4/KENO V.a) and the simplified DSC layout in Figure 2 (using both SCALE 4.4/KENO V.a and CASMO-3). These calculations are all performed using perfect axial reflection. The Figure 1 calculations are performed for both the "site-specific" NUHOMS®-24P design as well as the general license NUHOMS®-24P/24PHB DSC. The difference between these designs is discussed in Section 2.

Table 5 presents the results of the criticality evaluations of the full-detail and simplified DSC models, which were performed using unirradiated "mbl" fuel assemblies, as described in Table 2, in the DSC storage cells. Calculations were carried out for two different enrichments (2.0 and 5.0 wt % U-235) and SFP water temperatures (68 and 150 °F).

As the results in Table 5 demonstrate, the infinite-array Figure 2 DSC model is extremely conservative, with calculated k_{eff}s more than 0.04 Δk higher than those associated with the full-detail Figure 1 DSC. Such a large amount of conservatism provides assurance that the Figure 2 DSC model is appropriate for the CASMO-3/SIMULATE-3 burnup-credit calculations described in Sections 6.4 and 6.5.

The 2-D infinite-array k_{eff} values in Table 5 show good agreement between SCALE 4.4/KENO V.a and CASMO-3. Note that some of the relative conservatism in the CASMO-3 model is attributable to ignoring its negative method bias (see Section 5).

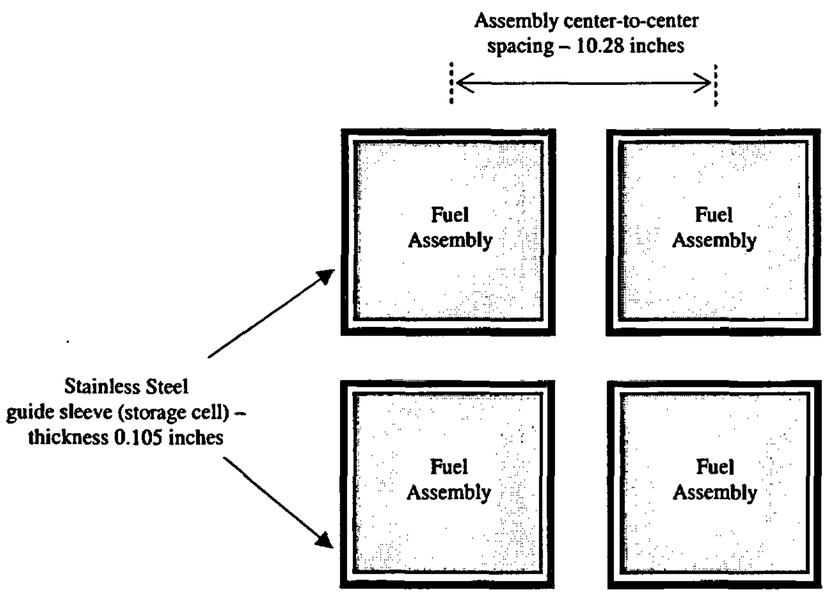


Figure 2. Simplified Infinite-Array DSC Model

Table 5. Comparison of 95/95 k_{eff} Results for NUHOMS®-24P/24PHB DSC Models (3-D SCALE 4.4/KENO V.a full-detail DSCs vs. 2-D Infinite-Array DSC models) {fresh "mbl" fuel, unborated SFP water}

DSC Model Evaluated	3-D General License DSC Model	3-D Site-Specific DSC Model	2-D Infinite-Array DSC Model	2-D Infinite-Array DSC Model
Code System Used	SCALE 4.4/ KENO V.a	SCALE 4.4/ KENO V.a	SCALE 4.4/ KENO V.a	CASMO-3
68 °F SFP water temp, 2.0 wt % U-235 "mbl" fuel, total 95/95 k_{eff}	0.9932	0.9980	1.0386	1.0437
150 °F SFP water temp, 2.0 wt % U-235 "mbl" fuel, total 95/95 k_{eff}	0.9977	1.0011	1.0439	1.0499
68 °F SFP water temp, 5.0 wt % U-235 "mbl" fuel, total 95/95 k_{eff}	1.2114	1.2180	1.2669	1.2713
150 °F SFP water temp, 5.0 wt % U-235 "mbl" fuel, total 95/95 k_{eff}	1.2200	1.2239	1.2756	1.2808

6.4 Axial Burnup Bias Evaluation

As noted in Section 6.1, it is important to quantify any positive reactivity effects attributable to the axial burnup distribution in stored fuel assemblies. In order to properly assess an appropriate axial burnup bias to apply to fuel stored in the NUHOMS®-24P/24PHB DSCs, the following steps were carried out:

- Axial burnup distributions were obtained for Oconee fuel assemblies that may be stored in the NUHOMS®-24P/24PHB DSCs per Section 3. A large sample size (> 1000 assemblies) was used for the evaluation. The assembly burnup distributions (in 23 axial nodes) were available from SIMULATE-3 Oconee core follow predictions. End-of-cycle axial burnup data from Oconee 1 Cycles 16-20, and Oconee 3 Cycles 15-18, were chosen for this analysis, as these cycles provided sufficient non-blanketed and blanketed fuel assembly burnup information for assessment.

- CASMO-3 core depletion cases were carried out with the fuel assembly types listed in Table 2, for several different enrichments that bounded the core follow operational data. The conservative depletion parameters identified in Section 6.2 were used, and the fuel assemblies were cooled for 5 years following reactor irradiation.

- A SIMULATE-3 nodal model of the NUHOMS®-24P/24PHB DSC was constructed, using the nominal infinite-array CASMO-3 DSC data described in Section 6.3. Axial reflectors containing a mixture of water, Zircaloy, and steel were specified.

- SIMULATE-3 cases were performed to calculate k_{eff}s for the core follow fuel assemblies compiled in the first step of this procedure. For a given fuel type considered, two SIMULATE calculations were carried out for each core follow fuel assembly: 1) using the real (predicted) axial burnup profile; and 2) using a flat axial profile, with each of the 23 nodes at the equivalent assembly-average burnup. These SIMULATE-3 cases showed that the "mbz" fuel type from Table 2 yielded the highest (most positive) axial burnup biases among the three fuel types considered.

- The individual "mbz" fuel assembly k_{eff} differences between real and flat axial profile cases were computed, and these Δk_{eff}s were then plotted as a function of burnup.

Figure 3 shows all of the Δk_{eff}s that were calculated in the last step of the above procedure. A 95/95 axial burnup bias as a linear function of enrichment is determined by drawing a line through the data points circled on Figure 3.

The resulting linear equation for the bias is:

$$\Delta k = \mathbf{0.00105912*BU - 0.02189}$$

where BU is assembly-average burnup, in GWD/MTU.

Note that this 95/95 "bounding" line does not include a few of the data points in Figure 3. Table A-31 in Reference 14 confirms that the "bounding" axial bias line in Figure 3 provides 95% probability, at a 95% confidence level, that the true axial burnup bias for a particular assembly does not exceed the value calculated per the above equation.

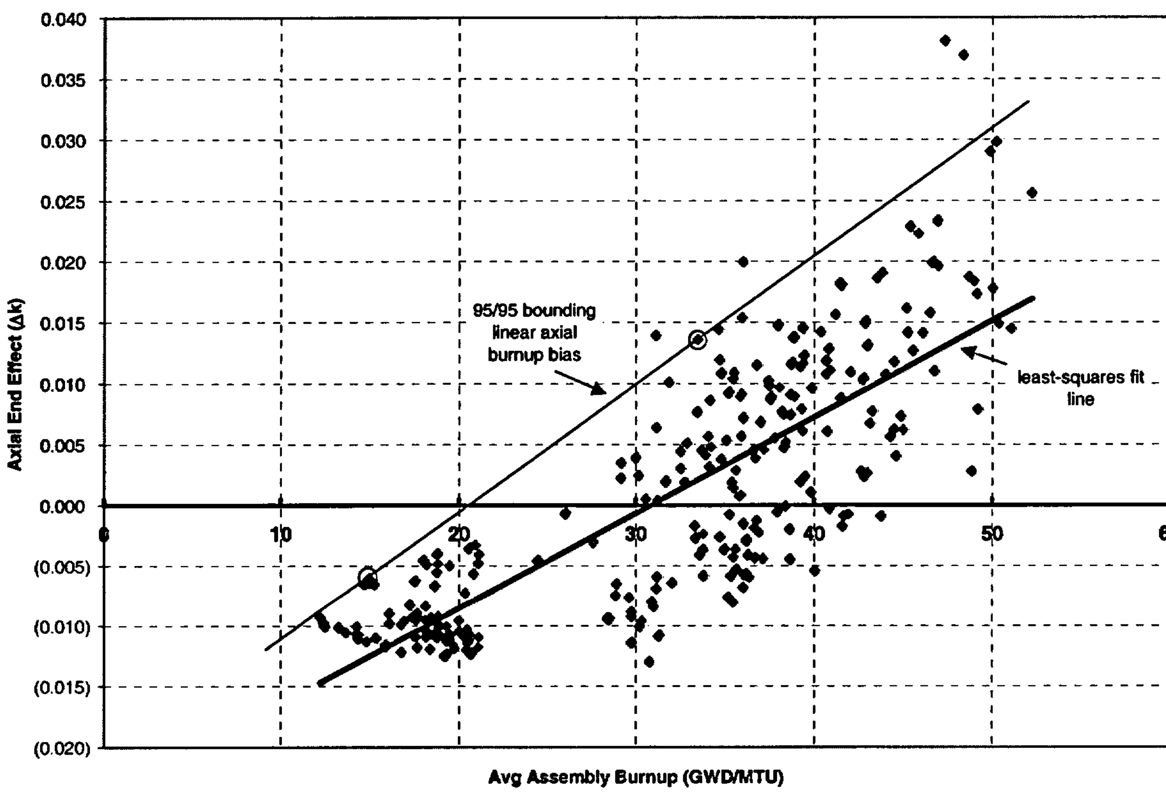


Figure 3. Individual Fuel Assembly Axial End Effect Biases {from Oconee Core Follow Data}

6.5 DSC Criticality Analysis Results

Using the infinite-array DSC model validated in Section 6.3, the main criticality calculations were performed with CASMO-3, using the three generic fuel assembly designs described in Section 3. The normal-condition criticality calculations for the NUHOMS®-24P/24PHB DSC were performed with no boron in the SFP water [to satisfy the 95/95 $k_{eff} < 1.0$ criterion of 10 CFR 50.68(b)], and with 430 ppm of soluble boron credit (to satisfy the 95/95 $k_{eff} < 0.95$ criterion of the same regulation).

Table 6 documents the maximum k_{eff} results, at various enrichments and corresponding burnups, for the no-boron normal-condition CASMO-3 DSC criticality calculations. These results are provided for the fuel assembly type (mbl) and SFP water temperature (150 °F) that yielded the highest k_{eff}s. The biases and uncertainties in this table are taken from Section 5. Note that the maximum computed 95/95 k_{eff} is 0.9980, meeting the subcriticality criterion identified in the previous paragraph.

Using the fuel assembly burnup requirements shown in Table 6, DSC calculations with 430 ppm boron yield the highest nominal k_{eff} (0.8645) at 5.0 wt % U-235 (as expected per the Reference 15 report). Applying the mechanical uncertainty documented in Section 5 for 430 ppm conditions (±0.0304 Δk), and assuming the remaining biases and uncertainties are the same as the no-boron values listed in Table 6, the maximum 95/95 k_{eff} for the NUHOMS®-24P/24PHB DSC in 430 ppm water is 0.9264.

The minimum fuel assembly burnup requirements shown in Table 6 are plotted as a function of enrichment in Figure 4. Because the data points in this figure show a high degree of linearity (the coefficient of determination, or R^2, is 0.9994), it is appropriate to perform a linear interpolation between neighboring data points in Figure 4, when determining the minimum burnup requirement for a fuel enrichment not specifically evaluated in Table 6.

Among the Reference 12 accident conditions that need to be considered, (abnormal water temperatures, water voiding, fuel assembly drop, misload, and placement immediately adjacent to the DSC), the fuel assembly misload is the worst-case event from a criticality perspective. The most severe type of misload is the placement of an unirradiated 5.0 wt % U-235 fuel assembly in the DSC. Among all the fuel types that have been used at Oconee (including those not eligible to be stored in the NUHOMS®-24P/24PHB DSCs), conservative criticality computations – using the SCALE 4.4/KENO V.a code with the simplified infinite-array model described in Section 6.3 – show that a misloaded MkB11 fuel assembly requires the most soluble boron, 630 ppm, to maintain DSC system k_{eff} below 0.95. This is still much less than the amount of soluble boron available for accident conditions (2220 ppm, as discussed in Section 6.2).

Note that a fuel assembly misloading event in the SFP, whether it occurs in the fuel storage racks or a NUHOMS®-24P/24PHB DSC, is not a new type of accident. Reference 2 mentions the misloading of a fresh 5.0 wt % U-235 assembly in the Oconee SFP storage racks as the most severe of the criticality accident scenarios. Reference 3 discusses the misloading of an unqualified high enrichment fuel assembly in the NUHOMS®-24P DSC as a postulated "off-normal" condition.

Table 6. Maximum 95/95 k_{eff}s for NUHOMS®-24P/24PHB DSC (Infinite-Array CASMO-3/SIMULATE-3 DSC Model) – Normal (non-accident) Conditions – {bounding "mbl" fuel, unborated SFP water at 150 °F}

Enrichment (wt % U-235)	**1.60**	**2.00**	**2.50**	**3.00**	**3.50**	**4.00**	**4.50**	**5.00**
Burnup (GWD/MTU)	**0**	**8.93**	**15.34**	**21.02**	**27.12**	**32.78**	**38.33**	**43.77**
Nominal CASMO-3 k_{eff}	0.9673	0.9624	0.9625	0.9620	0.9556	0.9496	0.9437	0.9380
Benchmark Method Bias	(0.0015)*	(0.0015)*	(0.0015)*	(0.0015)*	(0.0015)*	(0.0015)*	(0.0015)*	(0.0015)*
Axial Burnup Bias	0	(0.0124)*	(0.0056)*	0.0004	0.0068	0.0128	0.0187	0.0245
Benchmark Method Uncert	0.0121	0.0121	0.0121	0.0121	0.0121	0.0121	0.0121	0.0121
Mechanical Uncerts	0.0280	0.0280	0.0280	0.0280	0.0280	0.0280	0.0280	0.0280
Burnup Comp Uncert	0	0.0151	0.0151	0.0151	0.0151	0.0151	0.0151	0.0151
Burnup Meas Uncert	0	0.0104	0.0104	0.0104	0.0104	0.0104	0.0104	0.0104
Total 95/95 k_{eff}	**0.9978**	**0.9980**	**0.9980**	**0.9980**	**0.9980**	**0.9980**	**0.9980**	**0.9980**

* -- negative bias conservatively ignored

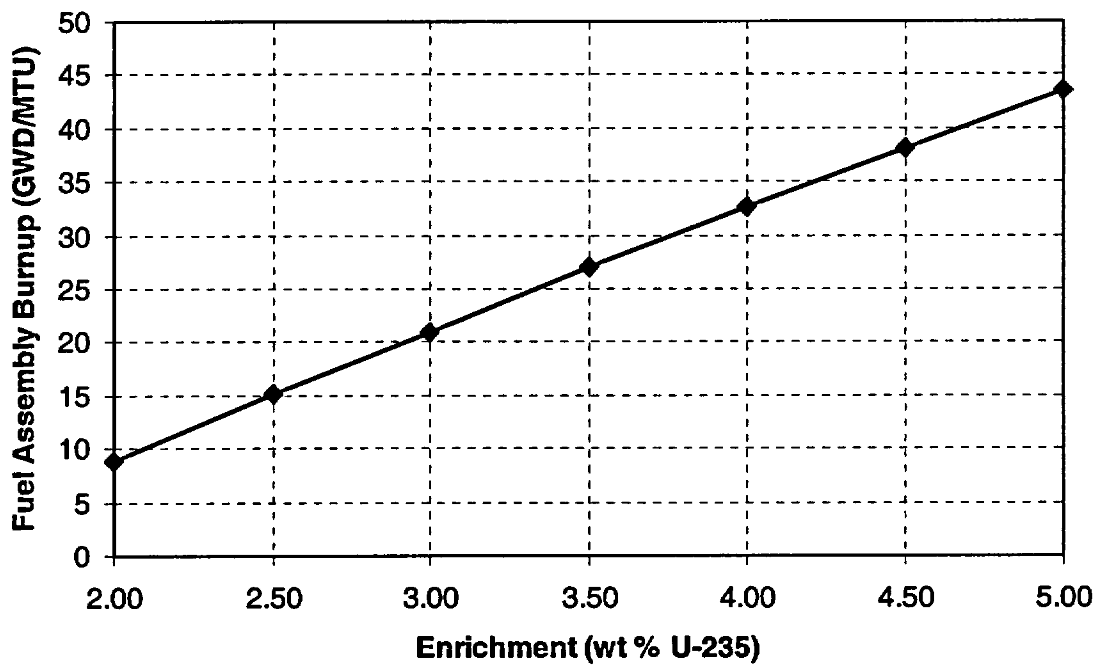


Figure 4. NUHOMS®-24P/24PHB DSC Minimum Burnup Requirements to meet 10 CFR 50.68(b) Criteria (Minimum 5 Years Post-Irradiation Cooling Time) {MkB2-B8, MkB9, MkB10 Fuel Types}

7 Conclusions

The criticality analysis of the NUHOMS®-24P/24PHB DSC, for loading and unloading operations in the Oconee SFPs, has been performed in accordance with the requirements of 10 CFR 50.68(b). This evaluation takes partial credit for soluble boron in the SFPs.

Minimum burnup requirements were developed for fuel to be placed without location restrictions in the NUHOMS®-24P/24PHB DSC. These burnup requirements, applicable for eligible fuel assemblies with a minimum 5 years post-irradiation cooling time, are a function of initial U-235 enrichment.

In the DSC criticality analysis, the maximum 95/95 k_{eff} with no boron in the Oconee SFP was calculated to be **0.9980**. This meets the no-boron 95/95 $k_{eff} < 1.0$ criterion in 10 CFR 50.68(b). The criticality evaluation also confirmed that with 430 ppm of partial soluble boron credit, the maximum 95/95 k_{eff} of **0.9264** remains well below the regulatory requirement that the maximum 95/95 k_{eff} be less than 0.95 for all normal conditions.

Finally, the criticality analysis demonstrated that the current minimum boron concentration required in the Oconee SFPs (2220 ppm) is adequate to maintain the maximum 95/95 k_{eff} below 0.95 for all credible accident scenarios associated with loading fuel assemblies into the NUHOMS®-24P/24PHB DSCs.

8 References

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16. **"McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC NOs MC0945 and MC0946),"** Letter from J. Shea (U.S. NRC) to G. Peterson (Duke), March 17, 2005.

17. **Determination of the Accuracy of Utility Spent-Fuel Burnup Records**, EPRI TR-112054, Electric Power Research Institute, July 1999.

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ENCLOSURE 4

COMPLIANCE WITH 10 CFR 50.68(b)

The eight criteria in 10 CFR 50.68(b) are listed below, along with a discussion of how Oconee complies with each criterion. This information is being provided for information only purposes and may be revised in the future consistent with 10 CFR 50.59. Therefore, this information does not represent specific commitments:

(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

All storage locations have been evaluated to ensure that fuel can be stored safely at all times, under the most adverse moderator conditions as specified by appropriate regulations. Controls and procedures are in place to ensure that fuel assemblies are only placed into an acceptable storage configuration as allowed by appropriate regulatory criteria. Procedures and other administrative controls for the handling of fuel assemblies ensure that the movement of a fuel assembly is performed safely and that the fuel assembly being moved remains subcritical even under the most adverse moderation conditions.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

Oconee has no fresh fuel storage racks, so this criterion is not applicable.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

Oconee has no fresh fuel storage racks, so this criterion is not applicable.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded

with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Duke is currently taking credit for 430 ppm soluble boron in the Oconee SFPs, per Technical Specification (TS) 4.3.1.c and proposed TS 4.4.1.c.

For fuel assemblies in the Oconee spent fuel storage racks, the NRC Safety Evaluation for the current licensing basis ["Oconee Nuclear Station Units 1, 2, and 3 Re: Issuance of Amendments (TAC NOs MB0894, MB0895, and MB0896)," Letter from L. Olshan (U.S. NRC) to W. McCollum (Duke), April 22, 2002] noted the 10 CFR 50.68(b)(4) requirement, and approved the methodology Duke employed to meet the dual subcriticality criteria in this regulation.

For fuel assemblies to be loaded into the NUHOMS®-24P and NUHOMS®-24PHB dry storage canisters (DSCs), in accordance with the proposed new TS 4.4 and TS 3.7.18 requirements, the Enclosure 3 DSC criticality analysis shows that the maximum 95/95 k_{eff} with no boron in the Oconee SFP is 0.9980. The DSC criticality analysis also confirms that with 430 ppm soluble boron credit, the maximum 95/95 k_{eff} is 0.9264, well below the 0.95 criterion.

(5) The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

Excluding the nuclear fuel, there is a limited amount of SNM stored at various locations onsite at Oconee. The total quantity of non-fuel SNM (< 200 grams of fissile material) is below the amount for a critical mass defined in 10 CFR 70.4.

(6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

Fuel assemblies are stored and handled in areas of the plant discussed below. Radiation monitoring is provided for these areas to detect excessive radiation levels and will provide an alarm to alert personnel if a potential radiation hazard is present.

1. Unit 1 and 2 Fuel Building; includes the fuel receiving area.
2. Unit 1 and 2 Spent Fuel Pool; includes the cask loading pit, the decon area, the new fuel elevator, the fuel transfer tube area and the spent fuel storage area/racks.
3. Unit 1 Reactor Building; includes the fuel transfer tube area, the reactor core and the refueling canal.

4. Unit 2 Reactor Building; includes the fuel transfer tube area, the reactor core and the refueling canal.
5. Unit 3 Fuel Building; includes the fuel receiving area.
6. Unit 3 Spent Fuel Pool; includes the cask loading pit, the decon area, the new fuel elevator, the fuel transfer tube area and the spent fuel storage area/racks.
7. Unit 3 Reactor Building; includes the fuel transfer tube area, the reactor core and the refueling canal.

Another area in which fuel assemblies are stored at Oconee is the ISFSI, which has been licensed in accordance with 10 CFR Part 72. As such, this area of Oconee is not addressed by this response.

(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

This limit is provided in TS 3.7.13 and 4.3.

(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).

The Oconee UFSAR is updated on an annual basis and a submittal made to the NRC at the end of June. Following approval of this amendment request, applicable sections of the Oconee Nuclear Site UFSAR will be updated to fully credit 10 CFR 50.68(b) criteria no later than June 30, 2007.

ATTACHMENT 1

TECHNICAL SPECIFICATIONS – MARK UPS

TABLE OF CONTENTS

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Boron Concentration

LCO 3.7.12 The spent fuel pool boron concentration limit shall be within limits.

APPLICABILITY: ~~When fuel assemblies are stored in the spent fuel pool~~, and when fuel assemblies are in a dry cask located in the spent fuel pool. spent fuel storage

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-----NOTE----- LCO 3.0.3 is not applicable.	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

ADD TO TS

3.7 PLANT SYSTEMS

3.7.18 Dry Spent Fuel Storage Cask Loading and Unloading

LCO 3.7.18 The combination of initial enrichment, burnup and post-irradiation cooling time of each fuel assembly in a dry spent fuel storage cask shall meet the criteria of Table 3.7.18-1.

APPLICABILITY: Whenever any fuel assembly is in a dry spent fuel storage cask located in the spent fuel pool.

ACTIONS

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

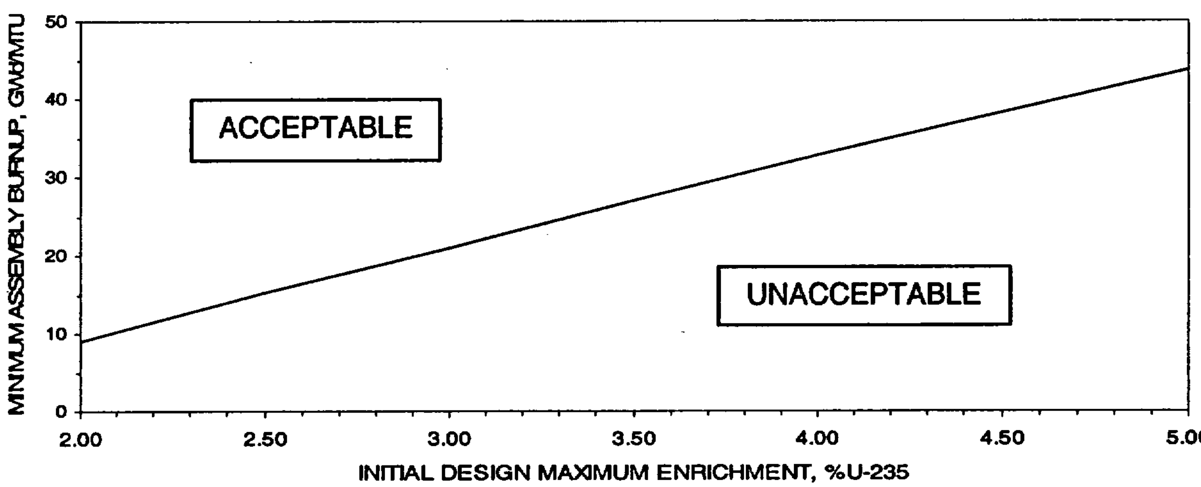
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify by administrative means the initial enrichment, burnup, and post-irradiation cooling time of the fuel assembly is in accordance with Table 3.7.18-1.	Prior to placing the fuel assembly into a dry spent fuel storage cask for loading AND Prior to placing a dry spent fuel storage cask into the spent fuel pool for unloading.

ADD TO TS

Table 3.7.18-1 (page 1 of 1)
Minimum Qualifying Burnup versus Design Maximum Enrichment
for Dry Spent Fuel Storage Cask Loading and Unloading

Initial Design Maximum Enrichment (Weight% U-235)	Minimum Assembly Burnup (GWD/MTU)
1.60 (or less)	0
2.00	8.93
2.50	15.34
3.00	21.02
3.50	27.12
4.00	32.78
4.50	38.33
5.00	43.77



NOTES:

The Design Maximum enrichment indicated above is the nominal maximum enrichment of any fuel pin in the fuel assembly being considered. The as-built enrichment of a fuel assembly may exceed its specified Design Maximum by up to 0.05 wt % U-235 and still be loaded in accordance with the above burnup limits for that Design Maximum enrichment. The minimum burnup requirements indicated above are based on a minimum post-irradiation cooling time of 5 years.

Fuel which differs from those designs used to determine the requirements of Table 3.7.18-1 may be qualified by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 430 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR. Maintaining the normal spent fuel pool boron concentration within the TS limits assures $k_{eff} \leq 0.95$ for any accident condition;

d. A nominal 10.65 inch center to center distance between fuel assemblies placed in spent fuel storage racks serving Units 1 and 2;

e. A nominal 10.60 inch center to center distance between fuel assemblies placed in spent fuel storage racks serving Unit 3;

f. A nominal 25.75 inch center to center spacing between fuel assemblies placed in the fuel transfer canal.

4.3.2 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.

INSERT 3

ADD to TS Insert 3

4.0 DESIGN FEATURES

4.4 Dry Spent Fuel Storage Cask Loading and Unloading

4.4.1 Criticality

Dry spent fuel storage cask loading or unloading in the spent fuel pool shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 430 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR. Maintaining the normal spent fuel pool boron concentration within the TS limits assures $k_{eff} \leq 0.95$ for any accident condition;

d. Dry spent fuel storage cask designs limited to NUHOMS®-24P or NUHOMS®-24PHB.

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

Insert 1

~~The Oconee spent fuel storage racks contain Boraflex neutron-absorbing panels that surround each storage cell on all four sides (except for peripheral sides). The function of these Boraflex panels is to ensure that the reactivity of the stored fuel assemblies is maintained within required limits. Boraflex, as manufactured, is a silicon rubber material that retains a powder of boron carbide (B_4C) neutron absorbing material. The Boraflex panels are enclosed in a formed stainless steel wrapper sheet that is spot-welded to the storage tube. The wrapper sheet is bent at each end to complete the enclosure of the Boraflex panel. The Boraflex panel is contained in the plenum area between the storage tube and the wrapper plate. Since the wrapper plate enclosure is not sealed, spent fuel pool water is free to circulate through the plenum. It has been observed that after Boraflex receives a high gamma dose from the stored irradiated fuel ($>10^{10}$ rads) it can begin to degrade and dissolve in the wet environment. Thus, the B_4C poison material can be removed, thereby reducing the poison worth of the Boraflex sheets. This phenomenon is documented in NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks".~~

~~To address this degradation, the Oconee spent fuel storage racks have been analyzed taking credit for soluble boron as allowed in Reference 1. The methodology ensures that the spent fuel rack multiplication factor, k_{eff}, is less than or equal to 0.95 as recommended in ANSI/ANS-57.2-1983 (Ref. 2) and NRC guidance (Ref. 3). The spent fuel storage racks are analyzed to allow storage of fuel assemblies with enrichments up to a maximum nominal enrichment of 5.00 weight percent (wt %) Uranium-235 while maintaining $k_{eff} \leq 0.95$, including uncertainties, tolerances, biases, and credit for soluble boron. Note that the criticality analysis accounts for a maximum as-built enrichment tolerance of 0.05 wt % U-235. For example, for a specified maximum design enrichment of 5.00 wt % U-235, an as-built enrichment up to 5.05 weight percent is acceptable. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions and to provide subcritical margin such that the spent fuel pool $k_{eff} \leq 0.95$. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal conditions is 430 ppm. In addition, sub-criticality of the pool ($k_{eff} < 1.0$) is assured on a 95/95 basis without the presence of the soluble boron in the pool (excluding certain burnup-related uncertainties described in the criticality analysis). The criticality~~

Insert 1.

Each Oconee spent fuel pool (SFP) contains racks for fuel assembly storage and a cask pit area for loading assemblies into a NUHOMS®-24P/24PHB dry storage canister (DSC). Criticality analyses have been performed for both SFP rack storage and DSC loading/unloading operations, in accordance with the regulation (Ref. 1) and the guidance in References 2 and 3. The SFP and DSC criticality analyses each take credit for 430 ppm soluble boron during normal conditions, in order to achieve system $k_{eff} \leq 0.95$. This partial soluble boron credit is included in Specifications 4.3.1 c. (SFP storage racks) and 4.4.1 c. (DSC).

The SFP storage rack criticality analysis yields fuel assembly storage configuration requirements and associated minimum burnup values (as a function of initial U-235 enrichment), which are specified in LCO 3.7.13. The DSC criticality evaluation establishes minimum burnup requirements for the loading of fuel assemblies into a NUHOMS®-24P/24PHB DSC without location restrictions. The DSC burnup requirements are provided in LCO 3.7.18.

The minimum SFP boron concentration of 2220 ppm (per SR 3.7.12.1) allows sufficient time to detect and mitigate all credible boron dilution scenarios, well before the SFP boron concentration drops to 430 ppm. The minimum 2220 ppm boron is available for all accident conditions evaluated in the SFP rack and DSC criticality analyses, per the double contingency principle (Ref. 4).

BASES

~~BACKGROUND (continued)~~

~~analysis performed shows that the acceptance criteria for criticality are met for the storage of fuel assemblies when credit is taken for reactivity depletion due to fuel burnup, no credit for the Boraflex neutron absorber panels, and storage configurations and enrichment limits specified by LCO 3.7.13.~~

APPLICABLE SAFETY ANALYSES

~~Most accident conditions do not result in an increase in reactivity in the spent fuel pool. Examples of these accident conditions are the drop of a fuel assembly on top of a rack, the drop of a fuel assembly between rack modules (rack design precludes this condition), and the drop of a fuel assembly between rack modules and the pool wall. However, four accidents can be postulated which could result in an increase in reactivity in the spent fuel storage pools. The first is a drop or placement of a fuel assembly into the cask loading area. The second is a loss of normal cooling to the spent fuel pool water which causes an increase in the pool water temperature. The third is the misloading of a fuel assembly into a location in which the restrictions on location, enrichment and burnup are not satisfied. The fourth is a drop of a heavy load onto the spent fuel racks.~~

Insert 2

~~For an occurrence of these postulated accidents, the double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 430 ppm required to maintain~~ $k_{eff} \leq$ ~~to 0.95 under normal storage conditions) can be assumed as a realistic initial condition, since not assuming its presence would be a second unlikely event.~~

~~Calculations were performed to determine the amount of soluble boron required to offset the highest reactivity increase caused by these postulated accidents, to maintain~~ $k_{eff} \leq 0.95$~~. It was found that a spent fuel pool boron concentration of 2220 ppm was sufficient to maintain~~ $k_{eff} \leq 0.95$ ~~for the worst-case postulated criticality-related accident (the heavy load drop event). Specification 3.7.12 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by these postulated accidents.~~

~~The minimum boron concentration limit ensures the SFP boron concentration is adequate to meet the sub-criticality requirements of fuel stored in the SFP for the most limiting accident in the SFP: A cask drop onto fuel in the SFP.~~

Insert 2.

Reference 3 discusses several criticality accident conditions that should be considered in SFP storage rack criticality analyses. Applicable accidents for the Oconee SFP storage racks include: 1) drop of a fuel assembly on top of the SFP storage rack; 2) drop of a fuel assembly outside of the storage rack modules; 3) abnormal SFP water temperatures outside the normal temperature range; 4) the misloading of a fuel assembly in a storage cell for which restrictions on location, enrichment or burnup are not satisfied; and 5) the drop of a heavy load (transfer cask) onto the SFP storage racks (NUREG-0612). Of these SFP storage rack accidents, the heavy load drop event requires the largest amount of soluble boron (almost 2200 ppm) to maintain SFP $k_{eff} \leq 0.95$.

The accident scenarios (Ref. 3) that are valid for the loading/unloading of a NUHOMS®-24P/24PHB DSC include: 1) drop of a fuel assembly on top of the DSC storage cells; 2) drop of a fuel assembly immediately outside of the transfer cask containing the DSC; 3) abnormal SFP water temperatures beyond the normal temperature range; and 4) the misloading of a fresh 5.0 wt % U-235 fuel assembly in one of the DSC storage cells. Of these DSC accidents, the misload event requires the largest amount of soluble boron (630 ppm) to achieve a system $k_{eff} \leq 0.95$.

Note that it is plausible to consider a loss of normal SFP cooling accident occurring in conjunction with a boron dilution event in the Oconee SFPs. In this unlikely scenario, with SFP water temperatures up to 212 ºF, the largest concentration of soluble boron required to maintain system $k_{eff} \leq 0.95$ is 500 ppm (for the SFP storage racks). This amount of soluble boron is still much less than that remaining after the worst-case credible dilution event (825 ppm).

Therefore, maintaining the SFP boron concentration ≥ 2220 ppm per SR 3.7.12.1 ensures that $k_{eff} \leq 0.95$ for any accident conditions in the SFP storage rack or NUHOMS®-24P/24PHB DSC. This minimum boron concentration limit includes allowance for analytical, mechanical, and instrument measurement uncertainties.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Note that it is plausible that the "loss of normal cooling" accident could occur in conjuction with a spent fuel pool boron dilution event. Criticality calculations show that the soluble boron needed to maintain $k_{eff} \le 0.95$ for the "loss of normal cooling" accident (500 ppm) is still less than the boron concentration following the worst-case credible dilution event (825 ppm).~~

Insert 2 (cont'd)

~~Therefore, maintaining the spent fuel pool boron concentration within the limits assures $k_{eff} \le 0.95$ for any accident condition. For normal storage conditions, Specification 4.3.1 c. requires that the spent fuel rack k_{eff} be $\le$ 0.95 when flooded with water borated to 430 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 430 ppm is not a credible event.~~

~~The numerical value of the SR 3.7.12.1 minimum boron concentration limit ($\ge$ 2220 ppm) includes allowance for analytical mechanical and instrument measurement uncertainties.~~

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

LCO

Insert 3

~~The dissolved boron concentration limits for in spent fuel pool preserves the assumption used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.~~

APPLICABILITY

Insert 4

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

ACTIONS

A.1 and A.2

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

Insert 3.

The minimum concentration of dissolved boron in the SFP (2220 ppm) preserves the assumptions used in the analyses of the potential accident scenarios described above. This minimum boron concentration ensures that the system k_{eff} for the SFP storage rack or the NUHOMS®-24P/24PHB DSC will remain below 0.95 for all credible criticality accident scenarios and boron dilution events.

Insert 4.

This LCO applies whenever fuel assemblies are stored in the SFP storage racks, or whenever fuel assemblies are being loaded into a NUHOMS®-24P/24PHB DSC in the SFP.

BASES

ACTIONS

A.1 and A.2 (continued)

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

When the concentration of boron in the SFP ~~fuel storage pool~~ is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. Immediate action is also required to initiate action to restore the SFP boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the SFP ~~fuel storage pool~~ is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time. The COLR revision process assures that the minimum boron concentration specified in the COLR bounds the limit specified by this SR.

REFERENCES

1. 10CFR50.68(b) ~~WCAP-14416-NP-A, Westinghouse Spent Fuel Rack Criticality Analysis Methodology, Revision 1, November 1996.~~

2. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.

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

BASES

REFERENCES (continued)	4.	Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
	5.	10 CFR 50.36.

ADD TO TSB

~~B~~ 3.7 PLANT SYSTEMS

B 3.7.18 Dry Spent Fuel Storage Cask Loading and Unloading

BASES

BACKGROUND

Fuel loading and unloading operations for the NUHOMS®-24P and NUHOMS®-24PHB dry storage canisters (DSCs) take place in the cask pit area of the spent fuel pool. The cask pit is adjacent to the spent fuel storage racks in each of the Oconee spent fuel pools, and is open to the rest of the spent fuel pool at all times. The NUHOMS®-24P and NUHOMS®-24PHB DSCs contain storage cells for 24 fuel assemblies. Eligible B&W 15x15 fuel assemblies (MkB2-B8, MkB9, and MkB10) with initial enrichments ≤ 5.0 wt % U-235 may be stored in the NUHOMS®-24P or NUHOMS®-24PHB DSC, as long as the fuel assemblies meet the minimum burnup and cooling time requirements specified in Table 3.7.18-1.

For normal conditions in the spent fuel pool, the NUHOMS®-24P and NUHOMS®-24PHB DSCs have been analyzed using credit for soluble boron as allowed in Reference 1. This ensures that the system multiplication factor, k_{eff}, is $\leq$ 0.95 as recommended in ANSI/ANS-57.2-1983 (Ref. 2) and NRC guidance (Ref. 3). The DSC is analyzed to allow loading/unloading of eligible fuel assemblies while maintaining $k_{eff} \leq 0.95$, including uncertainties, tolerances, biases, and credit for 430 ppm soluble boron. Note that the criticality analysis accounts for a maximum as-built enrichment tolerance of 0.05 wt % U-235. For example, for a specified maximum design enrichment of 5.00 wt % U-235, an as-built enrichment up to 5.05 weight percent is acceptable. The 430 ppm soluble boron credit must provide sufficient subcritical margin to maintain the DSC $k_{eff} \leq$ 0.95. In addition, sub-criticality of the DSC ($k_{eff} < 1.0$) must be assured on a 95/95 basis, without the presence of any soluble boron in the spent fuel pool.

The dual k_{eff} criteria identified in the above paragraph are satisfied for fuel assemblies meeting the minimum burnup and post-irradiation cooling time requirements specified in Table 3.7.18-1. Reactivity reduction with cooling time is primarily attributable to Pu-241 decay and Gd-155 buildup (via Eu-155 decay).

Specification 4.4.1 c. requires that the DSC k_{eff} be $\leq$ 0.95 when flooded with water borated to 430 ppm. A spent fuel pool boron dilution analysis has been performed that confirms that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff}

ADD TO TSB

BASES

BACKGROUND (continued)

design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 430 ppm is not a credible event.

APPLICABLE SAFETY ANALYSES

Several accident conditions (Ref. 3) are considered that could result in an increase in system k_{eff} for a DSC being loaded or unloaded in the spent fuel pool. These accident conditions include the drop of a fuel assembly on top of the DSC storage cells, the drop of a fuel assembly just outside the transfer cask containing the DSC, a higher than normal spent fuel pool water temperature, and the misloading of a fresh 5.0 wt % U-235 assembly in one of the DSC storage cells.

For an occurrence of these postulated accidents, the double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) can be applied. This double contingency principle does not require assuming two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 430 ppm required to maintain $k_{eff} \leq 0.95$ under normal DSC loading/unloading conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed to determine the amount of soluble boron required to offset the highest reactivity increase associated with these postulated accidents, in order to maintain $k_{eff} \leq 0.95$. It was found that a spent fuel pool boron concentration of 630 ppm was sufficient to maintain $k_{eff} \leq 0.95$ for the worst-case postulated criticality-related accident (the fresh fuel assembly misloaded in a DSC storage cell). Specification 3.7.12 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by these postulated accidents.

For normal storage conditions, Specification 4.3.1 c. requires that the spent fuel rack k_{eff} be ≤ 0.95 when flooded with water borated to 430 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 430 ppm is not a credible event.

ADD TO TSB

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The configuration of fuel assemblies in the DSC and the concentration of dissolved boron in the spent fuel pool satisfy Criterion 2 of 10 CFR 50.36 (Ref. 5)
LCO	The k_{eff} of the dry spent fuel storage cask (NUHOMS®-24P or NUHOMS®-24PHB DSC), during loading and unloading operations in the spent fuel pool, will always remain $\leq$ 0.95, assuming the spent fuel pool is flooded with water borated to at least 430 ppm, and that each loaded fuel assembly meets the initial enrichment, burnup, and post-irradiation cooling time of Table 3.7.18-1.
APPLICABILITY	This LCO applies whenever any fuel assembly is in a dry spent fuel storage cask located in the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies loaded in the NUHOMS®-24P or NUHOMS®-24PHB DSC is not in accordance with the LCO, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with the LCO.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment, burnup, and post-irradiation cooling time of the fuel assembly to be loaded into or removed from the NUHOMS®-24P or NUHOMS®-24PHB DSC is in accordance with Table 3.7.18-1.

ADD TO TSB

BASES

REFERENCES

1. 10 CFR 50.68(b)(4)

2. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.

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

4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

5. 10 CFR 50.36

ATTACHMENT 2

TECHNICAL SPECIFICATIONS– REPRINTED PAGES

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Boron Concentration

LCO 3.7.12 The spent fuel pool boron concentration limit shall be within limits.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and when fuel assemblies are in a dry spent fuel storage cask located in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	------------------NOTE----------------- LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

3.7 PLANT SYSTEMS

3.7.18 Dry Spent Fuel Storage Cask Loading and Unloading

LCO 3.7.18 The combination of initial enrichment, burnup and post-irradiation cooling time of each fuel assembly in a dry spent fuel storage cask shall meet the criteria of Table 3.7.18-1.

APPLICABILITY: Whenever any fuel assembly is in a dry spent fuel storage cask located in the spent fuel pool.

ACTIONS

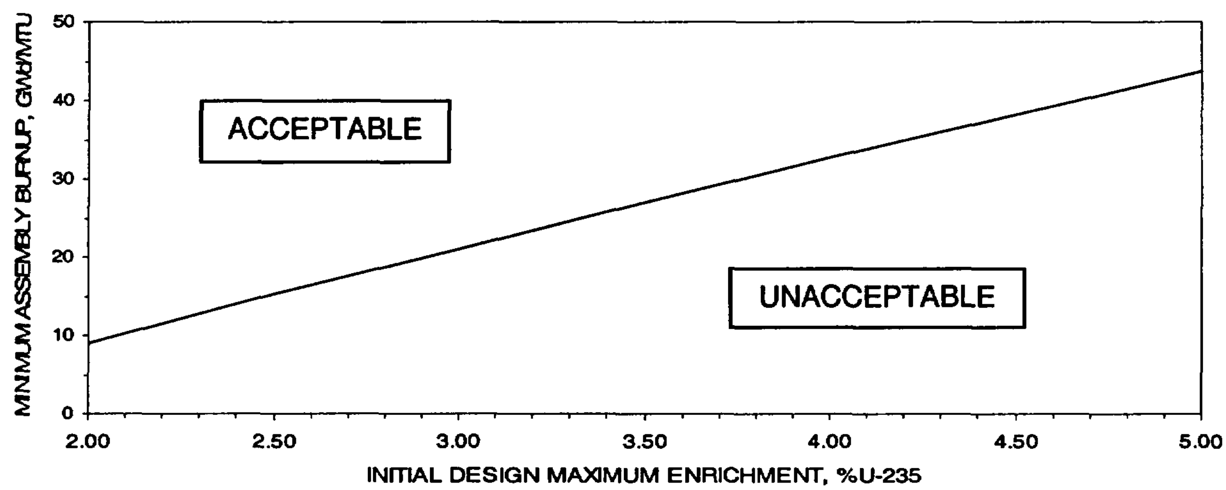
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -----------NOTE----------- LCO 3.0.3 is not applicable. ------------------------------ Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify by administrative means the initial enrichment, burnup, and post-irradiation cooling time of the fuel assembly is in accordance with Table 3.7.18-1.	Prior to placing the fuel assembly into a dry spent fuel storage cask for loading <u>AND</u> Prior to placing a dry spent fuel storage cask into the spent fuel pool for unloading.

Table 3.7.18-1 (page 1 of 1)
Minimum Qualifying Burnup versus Design Maximum Enrichment
for Dry Spent Fuel Storage Cask Loading and Unloading

Initial Design Maximum Enrichment (Weight% U-235)	Minimum Assembly Burnup (GWD/MTU)
1.60 (or less)	0
2.00	8.93
2.50	15.34
3.00	21.02
3.50	27.12
4.00	32.78
4.50	38.33
5.00	43.77



NOTES:

The Design Maximum enrichment indicated above is the nominal maximum enrichment of any fuel pin in the fuel assembly being considered. The as-built enrichment of a fuel assembly may exceed its specified Design Maximum by up to 0.05 wt % U-235 and still be loaded in accordance with the above burnup limits for that Design Maximum enrichment. The minimum burnup requirements indicated above are based on a minimum post-irradiation cooling time of 5 years.

Fuel which differs from those designs used to determine the requirements of Table 3.7.18-1 may be qualified by means of an analysis using NRC approved methodology to assure that k_{eff} is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron.

4.0 DESIGN FEATURES

4.4 Dry Spent Fuel Storage Cask Loading and Unloading

4.4.1 Criticality

Dry spent fuel storage cask loading or unloading in the spent fuel pool shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 430 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR. Maintaining the normal spent fuel pool boron concentration within the TS limits assures $k_{eff} \leq 0.95$ for any accident condition;

d. Dry spent fuel storage cask designs limited to NUHOMS®-24P or NUHOMS®-24PHB.

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

Each Oconee spent fuel pool (SFP) contains racks for fuel assembly storage and a cask pit area for loading assemblies into a NUHOMS® - 24P/24PHB dry storage canister (DSC). Criticality analyses have been performed for both SFP rack storage and DSC loading/unloading operations, in accordance with the regulation (Ref. 1) and the guidance in References 2 and 3. The SFP and DSC criticality analyses each take credit for 430 ppm soluble boron during normal conditions, in order to achieve system $k_{eff} \leq 0.95$. This partial soluble boron credit is included in TS 4.3.1 c. (SFP storage racks) and 4.4.1 c. (DSC).

The SFP storage rack criticality analysis yields fuel assembly storage configuration requirements and associated minimum burnup values (as a function of initial U-235 enrichment), which are specified in LCO 3.7.13. The DSC criticality evaluation establishes minimum burnup requirements for the loading of fuel assemblies into a NUHOMS® -24P/24PHB DSC without location restrictions. The DSC burnup requirements are provided in LCO 3.7.18.

The minimum SFP boron concentration of 2220 ppm (per SR 3.7.12.1) allows sufficient time to detect and mitigate all credible boron dilution scenarios, well before the SFP boron concentration drops to 430 ppm. The minimum 2220 ppm boron is available for all accident conditions evaluated in the SFP rack and DSC criticality analyses, per the double contingency principle (Ref. 4).

APPLICABLE SAFETY ANALYSES

Reference 3 discusses several criticality accident conditions that should be considered in SFP storage rack criticality analyses. Applicable accidents for the Oconee SFP storage racks include: 1) drop of a fuel assembly on top of the SFP storage rack; 2) drop of a fuel assembly outside of the storage rack modules; 3) abnormal SFP water temperatures outside the normal temperature range; 4) the misloading of a fuel assembly in a storage cell for which restrictions on location, enrichment, burnup, or post-irradiation cooling time are not satisfied; and 5) the drop of a heavy load (transfer cask) onto the SFP storage racks (NUREG-0612). Of these SFP storage rack accidents, the heavy load drop event requires the largest amount of soluble boron (almost 2200 ppm) to maintain SFP $k_{eff} \leq 0.95$.

BASES

ACTIONS

A.1 and A.2 (continued)

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

When the concentration of boron in the SFP is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. Immediate action is also required to initiate action to restore the SFP boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the SFP is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time. The COLR revision process assures that the minimum boron concentration specified in the COLR bounds the limit specified by this SR.

REFERENCES

1. 10 CFR 50.68(b).

2. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.

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

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).

BASES

REFERENCES (continued)	5.	10 CFR 50.36.

B 3.7 PLANT SYSTEMS

B 3.7.18 Dry Spent Fuel Storage Cask Loading and Unloading

BASES

BACKGROUND

Fuel loading and unloading operations for the NUHOMS®-24P and NUHOMS®-24PHB dry storage canisters (DSCs) take place in the cask pit area of the spent fuel pool. The cask pit is adjacent to the spent fuel storage racks in each of the Oconee spent fuel pools, and is open to the rest of the spent fuel pool at all times. The NUHOMS®-24P and NUHOMS®-24PHB DSCs contain storage cells for 24 fuel assemblies. Eligible B&W 15x15 fuel assemblies (MkB2-B8, MkB9, and MkB10) with initial enrichments ≤ 5.0 wt % U-235 may be stored in the NUHOMS®-24P or NUHOMS®-24PHB DSC, as long as the fuel assemblies meet the minimum burnup and cooling time requirements specified in Table 3.7.18-1.

For normal conditions in the spent fuel pool, the NUHOMS®-24P and NUHOMS®-24PHB DSCs have been analyzed using credit for soluble boron as allowed in Reference 1. This ensures that the system multiplication factor, k_{eff}, is $\leq$ 0.95 as recommended in ANSI/ANS-57.2-1983 (Ref. 2) and NRC guidance (Ref. 3). The DSC is analyzed to allow loading/unloading of eligible fuel assemblies while maintaining $k_{eff} \leq 0.95$, including uncertainties, tolerances, biases, and credit for 430 ppm soluble boron. Note that the criticality analysis accounts for a maximum as-built enrichment tolerance of 0.05 wt % U-235. For example, for a specified maximum design enrichment of 5.00 wt % U-235, an as-built enrichment up to 5.05 weight percent is acceptable. The 430 ppm soluble boron credit must provide sufficient subcritical margin to maintain the DSC $k_{eff} \leq$ 0.95. In addition, sub-criticality of the DSC ($k_{eff} < 1.0$) must be assured on a 95/95 basis, without the presence of any soluble boron in the spent fuel pool.

The dual k_{eff} criteria identified in the above paragraph are satisfied for fuel assemblies meeting the minimum burnup and post-irradiation cooling time requirements specified in Table 3.7.18-1. Reactivity reduction with cooling time is primarily attributable to Pu-241 decay and Gd-155 buildup (via Eu-155 decay).

Specification 4.4.1 c. requires that the DSC k_{eff} be $\leq$ 0.95 when flooded with water borated to 430 ppm. A spent fuel pool boron dilution analysis has been performed that confirms that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff}

BASES

BACKGROUND (continued)

design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 430 ppm is not a credible event.

APPLICABLE SAFETY ANALYSES

Several accident conditions (Ref. 3) are considered that could result in an increase in system k_{eff} for a DSC being loaded or unloaded in the spent fuel pool. These accident conditions include the drop of a fuel assembly on top of the DSC storage cells, the drop of a fuel assembly just outside the transfer cask containing the DSC, a higher than normal spent fuel pool water temperature, and the misloading of a fresh 5.0 wt % U-235 assembly in one of the DSC storage cells.

For an occurrence of these postulated accidents, the double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) can be applied. This double contingency principle does not require assuming two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of additional soluble boron in the spent fuel pool water (above the 430 ppm required to maintain $k_{eff} \leq 0.95$ under normal DSC loading/unloading conditions) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

Calculations were performed to determine the amount of soluble boron required to offset the highest reactivity increase associated with these postulated accidents, in order to maintain $k_{eff} \leq 0.95$. It was found that a spent fuel pool boron concentration of 630 ppm was sufficient to maintain $k_{eff} \leq 0.95$ for the worst-case postulated criticality-related accident (the fresh fuel assembly misloaded in a DSC storage cell). Specification 3.7.12 ensures the spent fuel pool contains adequate dissolved boron to compensate for the increased reactivity caused by these postulated accidents.

For normal storage conditions, Specification 4.3.1 c. requires that the spent fuel rack k_{eff} be ≤ 0.95 when flooded with water borated to 430 ppm. A spent fuel pool boron dilution analysis was performed which confirmed that sufficient time is available to detect and mitigate a dilution of the spent fuel pool before the 0.95 k_{eff} design basis is exceeded. The spent fuel pool boron dilution analysis concluded that an unplanned or inadvertent event which could result in the dilution of the spent fuel pool boron concentration to 430 ppm is not a credible event.

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The configuration of fuel assemblies in the DSC and the concentration of dissolved boron in the spent fuel pool satisfy Criterion 2 of 10 CFR 50.36 (Ref. 5)
LCO	The k_{eff} of the dry spent fuel storage cask (NUHOMS®-24P or NUHOMS®-24PHB DSC), during loading and unloading operations in the spent fuel pool, will always remain $\leq$ 0.95, assuming the spent fuel pool is flooded with water borated to at least 430 ppm, and that each loaded fuel assembly meets the initial enrichment, burnup, and post-irradiation cooling time of Table 3.7.18-1.
APPLICABILITY	This LCO applies whenever any fuel assembly is in a dry spent fuel storage cask located in the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies loaded in the NUHOMS®-24P or NUHOMS®-24PHB DSC is not in accordance with the LCO, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with the LCO.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment, burnup, and post-irradiation cooling time of the fuel assembly to be loaded into or removed from the NUHOMS®-24P or NUHOMS®-24PHB DSC is in accordance with Table 3.7.18-1.

BASES

REFERENCES

1. 10 CFR 50.68(b)(4)

2. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.

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

4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

5. 10 CFR 50.36

ATTACHMENT 3

LIST OF REGULATORY COMMITMENTS

The following commitment table identifies those actions committed to by Duke Energy Corporation (Duke) in this submittal. Other actions discussed in the submittal represent intended or planned actions by Duke. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Implementation Date
Upon NRC approval of this license amendment request, applicable sections of the Oconee Nuclear Site Updated Final Safety Analysis Report (UFSAR) will be updated and submitted in the annual report.	Prior to June 30, 2007