NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

December 21, 2006

NRC 2006-0089
10 CFR 50.90
10 CFR 50.68

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

License Amendment Request Number 247: Spent Fuel Pool Storage Criticality Control

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) proposes to revise the Point Beach Nuclear Plant (PBNP) Unit 1 and 2 licensing basis to reflect a revision to the spent fuel pool (SFP) criticality analysis methodology. The revised criticality analysis determined acceptable fuel storage configurations in the spent fuel pool storage racks with credit for fuel burnup, integral fuel burnable absorber (IFBA) pins, Plutonium-241 decay, and soluble boron, where applicable. Associated changes are proposed to Technical Specifications (TS) 3.7.12, "Spent Fuel Pool Storage," and 4.3.1, "Criticality," to reflect the results of the new criticality analysis.

Enclosure 1 provides a description and analysis of the proposed changes and includes the technical evaluation and associated no significant hazards and environmental considerations. Enclosure 2 provides the existing TS pages marked up to indicate the proposed changes. Enclosure 3 provides the revised (clean) TS pages. Enclosure 4 provides draft proposed TS Bases changes, for information. Enclosure 5 provides a copy of the boron dilution analysis performed in support of this amendment request. Enclosure 6 provides a non-proprietary version of the criticality analysis with soluble boron credit performed for the PBNP Units 1 and 2, by the Westinghouse Electric Corporation. Enclosure 7 provides a proprietary version of the criticality analysis (and associated affidavit).

As Enclosure 7 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-06-2108 and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

NMC currently holds an exemption for the PBNP units from the requirements of 10 CFR 70.24. Upon approval of this license amendment request to credit soluble boron and fuel storage patterns in lieu of Boraflex® the criticality licensing basis for PBNP Units 1 and 2 will be 10 CFR 50.68.

NMC requests approval of the proposed license amendment by December 20, 2007. While this license amendment request is neither exigent nor emergency, a prompt review is requested to resolve a TS non-conformance regarding the storage of several spent fuel assemblies as described herein. Once approved the license amendment will be implemented within 90 days.

<u>Summary of Commitments</u>

The plant-specific criticality analysis does not credit the Boraflex® neutron absorbing material in the fuel storage racks. Consequently, the associated commitments to maintain a Boraflex® surveillance and aging management program and perform "blackness" testing in response to Generic Letter 96-04 and license renewal are no longer necessary and will be cancelled.

Approval of this license amendment and adoption of 10 CFR 50.68 as the criticality licensing basis will result in the cancellation of the 10 CFR 70.24 exemption. As required by 10 CFR 50.68(b)(8) NMC is making the following commitment for the PBNP Units 1 and 2:

> Following approval of this amendment the PBNP Unit 1 and 2, Final Safety Analysis Report will be revised no later than the next update required under 10 CFR 50.71(e) to reflect the adoption of 10 CFR 50.68(b).

In accordance with 10 CFR 50.91, a copy of this amendment application, with the non-proprietary enclosures, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 21, 2006.

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures:
1) - Description and Analysis of Change
2) - Proposed Technical Specification Changes
3) - Revised Technical Specification Pages
4) - Proposed Technical Specification Bases
5) - Boron Dilution Analysis
6) - Westinghouse PBNP criticality analysis (Non-proprietary)
7) - Westinghouse PBNP criticality analysis (Proprietary)

cc: Administrator, Region III, USNRC (w/o Enclosure 6)
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC (w/o Enclosure 6)
Public Service Commission of Wisconsin (w/o Enclosure 6)

bcc: w/o Enclosures 5, 6 and 7 (except File, which gets all Enclosures)

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ENCLOSURE 1
DESCRIPTION AND ANALYSIS OF CHANGE

LICENSE AMENDMENT REQUEST NUMBER 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

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LICENSE AMENDMENT REQUEST NUMBER 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

1.0 SUMMARY

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) proposes to revise the Point Beach Nuclear Plant (PBNP) Unit 1 and 2 licensing bases to reflect the application of the Westinghouse Electric Corporation (Westinghouse) methodology described in WCAP-14416, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," (Reference 1) for the criticality analysis of the spent fuel pool. Westinghouse performed a plant-specific analysis entitled, "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis," WCAP-16541 (Reference 2). The criticality analysis determined acceptable fuel storage configurations in the spent fuel pool (SFP) storage racks with credit for fuel burnup, integral fuel burnable absorber (IFBA) pins, Plutonium-241 (Pu-241) decay, and soluble boron, where applicable. The results provide the basis for the necessary changes to the Technical Specifications (TS) for each unit. Changes to the following specifications (and/or associated bases) are proposed to conform to the results of the new criticality analysis.

- Specification 3.7.11 - Fuel Storage Pool Boron Concentration (Bases)
- Specification 3.7.12 - Spent Fuel Pool Storage
- Specification 4.3.1 - Criticality

The Technical Analysis portion of this letter (Section 6.0) describes the spent fuel rack criticality analysis methodology and boron dilution evaluation and provides the basis for the acceptability of the proposed TS changes.

The criticality analysis for PBNP Units 1 and 2 determined:

1. Fuel assembly burnup versus initial enrichment limits for safe storage in the following fuel storage configurations:

   - All-Cell,
   - 1-out-of-4 for 5.0 weight-percent Uranium-235 (U-235) fresh fuel with no IFBA, and
   - 1-out-of-4 for 4.0 weight-percent (w/o) U-235 fresh fuel with IFBA fuel storage configurations;

   with credit taken for 5, 10, 15, and 20 years of Pu-241 decay.

2. The number of IFBA pins versus initial enrichment limits required for safe storage in the '1-out-of-4 for 4.0 w/o U-235 fresh fuel with IFBA' storage configuration.

3. The fuel assembly loading requirements at the interfaces between fuel storage configurations.

4. The amount of soluble boron required to maintain the effective multiplication factor (k_{eff}) less than or equal to 0.95 in the SFP, including all biases and uncertainties, assuming the most limiting plausible reactivity accident.

The boron dilution analysis for PBNP Units 1 and 2 determined that 10 hours would be required to dilute the SFP from the minimum allowed TS boron concentration of 2,100 parts-per-million," (ppm) in Specification 3.7.11, "Fuel Storage Pool Boron Concentration," to the criticality analysis determined minimum concentration of 805 ppm. This time duration demonstrates that sufficient time is available for operators to recognize and terminate an inadvertent dilution event.

Currently, soluble boron in the SFP is not credited to maintain k_{eff} less than or equal to 0.95. The Boraflex® neutron absorber panels in the high-density fuel storage racks are presently credited in the criticality analyses to maintain subcritical conditions and the PBNP units hold an exemption from the requirements of 10 CFR 70.24, "Criticality Accident Requirements" (Reference 3). Upon approval of this license amendment request (LAR) the criticality licensing basis for the PBNP units will become 10 CFR 50.68, "Criticality Accident Requirements," and the 10 CFR 70.24 exemption will be superceded. Section 7.2 of this enclosure provides a discussion of how the PBNP intends to comply with the requirements of 10 CFR 50.68.

The PBNP Units 1 and 2 maintain a surveillance program to detect and monitor Boraflex® degradation in accordance with Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," (Reference 4). The new plant-specific criticality analysis documented in WCAP-16541 for the PBNP units does not credit the Boraflex® in the fuel storage racks obviating "blackness" testing and associated commitments in response to Generic Letter 96-04 and license renewal (Reference 28) as they are no longer necessary, as demonstrated by the results of the analysis. See Section 7.5 for additional detail for a list of commitments.

Additionally, with approval of this LAR, the associated TS changes will resolve an issue reported in Licensee Event Report (LER) 266/301/2006-002-00, "Fuel Assemblies in Spent Fuel Pool do not Meet Technical Specification Requirements," (Reference 5) concerning storage of twelve (12) spent fuel assemblies that met prior TS criticality requirements for spent fuel assemblies, but exceed the presently specified TS initial fuel enrichment limit of 4.60 weight-percent U-235 with no IFBA pins.

2.0 BACKGROUND

Approval of this LAR will resolve potential future issues with Boraflex® degradation and a current non-compliance with the PBNP TS concerning storage of 12 spent fuel assemblies in the SFP.

Resolution of Boraflex® Degradation

Boraflex® is a silicone-based polymer material containing boron-10 in the form of small particles of boron carbide as the neutron absorber. As discussed in Generic Letter 96-04, Boraflex® has degraded under the SFP environment in light-water reactors. Boraflex® is presently credited in the PBNP Unit 1 and 2 criticality analysis to ensure the subcriticality of the SFP. This LAR proposes to eliminate reliance on Boraflex® by revising the fuel storage related specifications and design features sections of the TS to reflect the results of the plant-specific criticality analysis (WCAP-16541) to maintain subcriticality in the fuel storage racks. The Boraflex® will be replaced by a combination of soluble boron and administrative controls requiring that the fuel (or pins) stored in the cells meet the requirements of the TS enforcing storage configurations, initial enrichment, burnup, IFBAs, P-241 decay, and soluble boron requirements of the criticality analysis to preclude criticality. No changes were determined necessary to the SFP boron concentration.

This LAR does not change or modify the fuel, fuel handling processes, spent fuel racks, number of fuel assemblies that may be stored in the SFP, decay heat generation rate, or the spent fuel pool cooling and cleanup system. The Boraflex® panels may be removed in the future but presently will remain in place. However, for the purposes of the licensing basis criticality analyses, it was assumed that no Boraflex® was present in the racks for reactivity control.

Resolution of LER 266/301/2006-002-00

On June 26, 2006 (Reference 5), it was discovered that 12 spent fuel assemblies stored in the SFP did not meet the current requirements of Specification 3.7.12. The current criticality analysis (approved on September 4, 1997) requires fuel assemblies with an initial U-235 enrichment greater than 4.60 w/o to have an acceptable number of IFBA rods based on Figure 3.7.12-1. The 12 assemblies had a nominal initial enrichment of 4.70 w/o and no IFBA rods (meeting the prior TS requirement of 4.75 w/o initial enrichment and IFBA rods were not required). The current criticality analysis, and the TS, formerly included an alternate analysis methodology to accommodate these 12 assemblies. The alternate criticality analysis methodology allowed assemblies with an initial enrichment greater than 4.60 w/o to be stored if they had a k-infinity (k_{inf}) less than a specified value. On February 26, 1999, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 99-003 (Reference 6) which stated they were abandoning the k_{inf} methodology because it could lead to IFBA requirements lower than those required by the IFBA enrichment curve. PBNP LAR 214, approved by the NRC on March 20, 2000, removed the k_{inf} methodology from the TS without recognizing the affect on the criticality provisions for storage of the 12 assemblies. When this LAR is approved the revised criticality analysis and proposed TS changes will restore compliance with the criticality analysis licensing basis and remedy this condition.

3.0 LICENSING BASIS

The current NRC regulatory requirements for maintaining subcritical conditions in SFPs are provided in 10 CFR 50.68, "Criticality Accident Requirements." Each holder of an operating license is required to either comply with 10 CFR 70.24, "Criticality Accident Requirements," to maintain a monitoring system capable of detecting a criticality event or comply with the 10 CFR 50.68 requirements to prevent a criticality event (or obtain an exemption to the regulation). NMC currently holds an exemption from the requirements of 10 CFR 70.24 (Reference 3) for PBNP. The cover letter to the exemption summarizes the pertinent parts of the exemption and states in-part:

> Based upon the information provided, there is reasonable assurance that irradiated and unirradiated fuel will remain subcritical during fuel handling and storage; furthermore, you maintain radiation monitors in accordance with PBNP's General Design Criterion 18 which is analogous to 10 CFR Part 50, Appendix A, Criterion 63. The low probability of a criticality together with your adherence to PBNP's General Design Criterion 18 constitute good cause for granting an exemption from 10 CFR 70.24.

The spent fuel rack criticality analysis methodology applied to the criticality analyses discussed herein invokes the requirements of 10 CFR 50.68. Approval of this license amendment request as discussed in NRC Regulatory Issue Summary 2005-05 (Reference 7) will invalidate the existing 10 CFR 70.24 exemption. The pertinent 10 CFR 50.68 regulatory requirements are:

10 CFR 50.68(b)(1) states:

> 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.

10 CFR 50.68(b)(4) states:

> 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.

10 CFR 50.68(b)(7) states:

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

Following approval of this proposed LAR, NMC will follow the requirements of 10 CFR 50.68 for PBNP with respect to the prevention of an inadvertent criticality event in the SFP. A discussion of how PBNP intends to comply with the requirements of 10 CFR 50.68 is provided in Section 7.2 of this enclosure. A discussion with respect to other pertinent NRC regulatory requirements and guidance is provided in Section 7.3 of this enclosure.

4.0 PROPOSED CHANGES

A description of the proposed changes to the TS (and associated TS Bases) is provided below. Mark-ups of the proposed TS wording and figures being added/removed are provided in Enclosure 2. Re-typed versions of the proposed TS wording and figure changes are provided in Enclosure 3. Draft proposed changes to the associated TS Bases are provided (for information only) in Enclosure 4.

4.1 TS 3.7.11 – Fuel Storage Pool Boron Concentration

To mitigate postulated criticality related accidents, boron is dissolved in the SFP water and controlled by Limiting Condition for Operation (LCO) 3.7.11, "Fuel Storage Pool Boron Concentration." The specified concentration of 2100 ppm provides significant margin to the concentration of 805 ppm assumed in the postulated limiting criticality accident event and boron dilution scenarios evaluated. This concentration is the minimum required concentration for fuel assembly storage and movement within SFP.

Surveillance Requirement (SR) 3.7.11.1 requires verification of the boron concentration every 7 days, consistent with SR 3.7.16.1, in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants" (Reference 8). No changes are proposed to Specification 3.7.11 or the dissolved boron concentration specified therein. However, the Bases for Specification 3.7.11 are proposed to be revised to reflect the results of the new criticality analysis.

4.2 TS 3.7.12 – Spent Fuel Pool Storage

The present TS 3.7.12, "Spent Fuel Pool Storage," defines acceptable conditions for fuel storage in the SFP based on fuel assembly initial enrichments of ≤ 4.6 w/o U-235 without IFBA, or for the combinations of initial enrichment and number of IFBA rods specified in current Figure 3.7.12-1.

Based on the results of the new criticality analysis, NMC proposes to revise LCO 3.7.12 to re-define the conditions for fuel storage as a function of initial fuel assembly enrichment, burnup, and decay time. This will allow any fuel assembly meeting the required conditions to be stored at any storage location (cell) within the SFP. This configuration is referred to as the All-Cell configuration and is the least restrictive storage configuration for the PBNP SFP. The following changes are proposed:

- Revise LCO 3.7.12 to remove LCO 3.7.12.a, which specifies a maximum fuel assembly initial enrichment of ≤ 4.60 w/o U-235 for fuel without IFBA.
- Revise LCO 3.7.12 to remove LCO 3.7.12.b which refers to Figure 3.7.12-1 to specify the acceptable number of, and poison material loadings, of the IFBA pins as a function of fuel assembly enrichment.
- Revise the text of LCO 3.7.12 and replace existing Figure 3.7.12-1 with a new figure that defines the acceptable range for fuel storage in the All-Cell configuration as a function of initial fuel assembly enrichment, burnup and decay time.
- Revise the text of LCO 3.7.12 to indicate that fuel must meet the conditions specified in Figure 3.7.12-1 or meet the storage configurations specified in Specification 4.3.1.1 by adding a reference to Specification 4.3.1.1.

Existing Figure 3.7.12-1 and the present LCO 3.7.12 wording (restated below),

LCO 3.7.12 (Old)	*Fuel assembly storage in the spent fuel pool shall be as follows:*
	a. Fuel assembly initial enrichment ≤ 4.6% w/o U-235; or
	b. Fuel assembly contains Integral Fuel Burnable Absorber (IFBA) rods within the "acceptable" range of Figure 3.7.12-1.

will be replaced by a new Figure 3.7.12-1 and revised LCO 3.7.12 wording as stated below.

LCO 3.7.12 (New)	*The combination of initial enrichment, burnup and decay time of each fuel assembly stored in the spent fuel pool shall be within the Acceptable range of Figure 3.7.12-1 or in accordance with Specification 4.3.1.1.*

The Bases for Specification 3.7.12 are proposed to be revised to reflect the results of the new criticality analysis (see Enclosure 4).

4.3 TS 4.3.1 – Criticality

TS 4.3, "Fuel Storage," provides the criteria for PBNP fuel storage. TS 4.3.1.1 specifies design features providing criticality control for the SFP fuel storage racks. Based on the new criticality analysis, NMC proposes to revise Specification 4.3.1.1 to credit soluble boron in the SFP for fuel storage in accordance with 10 CFR 50.68(b)(4). Specification 4.3.1.1 will now define more restrictive new and spent fuel storage configurations in the SFP allowing storage based upon a combination of burnup, initial enrichment, Pu-241 decay time, number of IFBA pins and poison material loading, in conjunction with specifying the fuel loading requirements at the interfaces between various storage configurations. This will allow any fuel assembly meeting the required conditions to be stored in the All-Cell, 1-out-of-4 for 5.0 w/o U-235 fresh fuel with no IFBA, or 1-out-of-4 for 4.0 w/o U-235 fresh fuel with IFBA storage configurations. The following changes are proposed to Specification 4.3.1 (items are renumbered as indicated below):

- Replace Specification 4.3.1.1.a with a statement indicating that fuel assemblies may have a maximum U-235 enrichment of 5.0 weight-percent.

- Revise Specification 4.3.1.1.b to increase the maximum k_{eff} from ≤ 0.95 to < 1.0 for when the SFP is fully flooded with unborated water in accordance with 10 CFR 50.68(b)(4).

- Add Specification 4.3.1.1.c to specify the allowable k_{eff} as ≤ 0.95 when the SFP is fully flooded with borated water to the required accident concentration in accordance with 10 CFR 50.68(b)(4).

- Renumber existing Specification 4.3.1.1.c as Specification 4.3.1.1.d.

- Add a new Specification 4.3.1.1.e stating that new or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time that is within in the "Acceptable" range of new Figure 3.7.12-1 may be allowed unrestricted storage in the fuel storage racks.

- Add a new Specification 4.3.1.1.f stating that new or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time that is in the "Unacceptable" range of new Figure 3.7.12-1 will be stored in compliance with the additional requirements specified in new Figures 4.3.1-1 through 4.3.1-8.

ENCLOSURE 1

- Add the following figures to Specification 4.3.1:

Figure	Description
4.3.1-1	1-Out-of-4 for 5 w/o with no IFBA Storage Configuration.
4.3.1-2	1-Out-of-4 for 4 w/o with IFBA Storage Configuration.
4.3.1-3	1-Out-of-4 for 5 w/o with no IFBA / "All Cell" Interface.
4.3.1-4	1-Out-of-4 for 4 w/o with IFBA / "All Cell" Interface.
4.3.1-5	1-Out-of-4 for 4 w/o with IFBA / 1-Out-of-4 for 5 w/o with no IFBA.
4.3.1-6	Spent Fuel Assembly Burnup Requirements for 1-Out-of-4 for 5.0 w/o with no IFBA.
4.3.1-7	Spent Fuel Assembly Burnup Requirements for 1-Out-of-4 for 4.0 w/o with IFBA.
4.3.1-8	Fresh Fuel IFBA Requirements.

Following the proposed changes to add new Figures 4.3.1-1 through 4.3.1-8, Specification 4.3.1 will be revised to read (changes underlined below):

4.3.1 Criticality (Old)

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies meeting at least one of the following storage limits may be stored in the spent fuel storage racks:

1. Fuel assemblies with an enrichment ≤4.6% weight percent U-235; or

2. Fuel assemblies which contains Integral Fuel Burnable Absorber (IFBA) pins in the "acceptable range" of Figure 3.7.12-1.

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR;

c. A nominal 9.825 inch center to center distance between fuel assemblies placed in the fuel storage racks;

to,

4.3.1 *Criticality*

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum 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 Reference 1;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 805 ppm, which includes an allowance for uncertainties as described in Reference 1;

d. A nominal 9.825 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "Acceptable" range of Figure 3.7.12-1 may be allowed unrestricted storage in the fuel storage racks; and

f. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "Unacceptable" range of Figure 3.7.12-1 will be stored in compliance with Figures 4.3.1-1 through 4.3.1-8.

There are no bases associated with the Section 4.0 specifications; therefore, there are no TS Bases changes for the above.

5.0 DESCRIPTION OF THE SFP AND FUEL STORAGE CELLS

PBNP has a single SFP divided into north and south halves connected through a divider wall. For analysis purposes the SFP was treated as two separate pools. Each pool has inside dimensions of approximately 220 inches by 408 inches. Figure 2-1 in Section 2.2 of the criticality analysis report depicts the locations of the fuel storage racks in the SFP. Table 2-1 summarizes the overall geometry data for the SFP.

Fuel storage cell dimensions are summarized in Table 2-2 and Figure 2-2 of the criticality analysis report. For analysis purposes the Boraflex® neutron absorbing material (poison) was assumed not to exist. The geometry of the Boraflex® was represented as water in the KENO model, thus no credit was taken for the presence of this neutron absorbing material in the criticality analysis. A failed fuel rod storage basket designed to accommodate individual spent and/or fresh fuel pins in a 7x7 array was conservatively modeled in the criticality analysis.

6.0 TECHNICAL ANALYSIS

On July 28, 1995, as supplemented on October 23, 1996, the Westinghouse Owners Group (WOG) submitted licensing topical report WCAP-14416-P-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," (Reference 1) providing the methodology for calculating the k_{eff} of spent fuel storage racks in which no credit was taken for soluble boron except under accident conditions. The report also presented a new procedure allowing partial credit for soluble boron in the SFP water when performing storage rack criticality analysis. Approval was documented in an NRC Safety Evaluation issued for the methodology (Reference 9).

General Design Criterion (GDC) 62, "Prevention of criticality in fuel storage and handling," in Appendix A to 10 CFR 50 (Reference 10) states:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The NRC established a 5-percent subcriticality margin (spent fuel pool multiplication factor k_{eff} less than 0.95) to comply with GDC 62 as recommended by ANS-57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," (Reference 11) and the guidance in a letter entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," (Reference 12). In accordance with this guidance, all of the applicable biases and uncertainties should be combined with k_{eff} to provide a one-sided, upper tolerance limit on k_{eff} such that the true value will be less than the calculated value with a 95-percent probability at a 95-percent confidence level.

The new procedure in WCAP-14416 allowed credit for soluble boron in the SFP to offset these uncertainties to maintain k_{eff} less than 0.95. However, the spent fuel rack k_{eff} calculation was required in accordance with 10 CFR 50.68(b)(4) to remain less than 1.0 (subcritical) when flooded with unborated water with a 95-percent probability at a 95-percent confidence level. Implementation of the proposed changes in the required fuel storage configurations and the associated fuel assembly reactivity requirements will continue to satisfy the requirements of GDC 62.

Specifically, the revised design basis for preventing criticality in the PBNP Unit 1 and 2 SFP in accordance with 10 CFR 50.68(b)(4) will be:

1. the k_{eff} of the fuel rack array shall be < 1.0 in unborated water, with a 95 percent probability at a 95 percent confidence level, including uncertainties; and

2. the k_{eff} of the fuel rack array shall be < 0.95 in the pool containing borated water, with a 95 percent probability at a 95 percent confidence level, including uncertainties.

Westinghouse issued in 2000, NSAL 2005-15, "Axial Burnup Shape Reactivity Bias," (Reference 13) to advise clients that the methodology in WCAP-14416 could be non-conservative with respect to the axial reactivity bias used to account for three-dimensional (3D) burnup effects in the two-dimensional (2D) model. The NRC in a letter to Westinghouse dated July 27, 2001 (Reference 14), stated the following:

> Although this approach may lead to sufficient margin to account for the identified non-conservatism(s) on a plant specific basis, it departs from the Westinghouse methodology approved in WCAP-14416-NP-A. Therefore, the staff concludes that the methodology of WCAP-14416 can no longer be relied upon as an "approved methodology" by the NRC staff or the licensees. For future licensing actions, licensees will need to submit plant-specific criticality calculations for spent fuel pool configurations that include technically supported margins.

Westinghouse has performed a new criticality analysis for the PBNP to eliminate reliance on Boraflex® in the fuel storage racks. The revised criticality analysis is based on the original methodology presented in WCAP-14416, but the 2D-to-3D axial burnup biasing methodology was not used. Instead, the 3D axial burnup distribution effects were explicitly modeled. The methodology used for the new PBNP criticality analysis is analogous to that for several recent analyses that have been reviewed and approved by the NRC. See Section 7.4 for some of the applicable precedents.

This LAR does not propose any physical changes to the SFP fuel storage racks or other plant systems which may have an impact on storage of fuel in the SFP. The proposed changes to the TS in this LAR implement the results of the revised analysis.

6.1 Methodology Summary

The Westinghouse plant-specific criticality analysis for PBNP determined the acceptable fuel storage conditions for the SFP fuel storage racks with credit for burnup, IFBA pins, Pu-241 decay and soluble boron, where applicable. Once the greatest reactivity caused by the limiting single postulated accident is determined the corresponding amount of soluble boron needed to mitigate is compared against the TS soluble boron limit. The methodology involved determining the:

- Acceptable fresh and spent fuel storage configurations with no soluble boron in the pool such that the 95/95 upper tolerance limit value of k_{eff}, including applicable biases and uncertainties, was less than 0.995.

  The actual NRC k_{eff} limit in accordance with 10 CFR 50.68(b)(4) is unity. An additional margin of 0.005 Δk_{eff} units was included for conservatism.

- Amount of soluble boron (in ppm) necessary to reduce the k_{eff} of all fuel storage configurations by at least 0.05 Δk_{eff} units and to compensate for 5 percent of the maximum burnup credited in any storage configuration.

- Largest increase in reactivity caused by postulated accidents and the amount of soluble boron needed to offset this reactivity increase.

- Fuel assembly burnup versus initial enrichment limits, crediting Pu-241 decay, required for storage of assemblies for each storage configuration.

- Number of IFBA pins versus initial enrichment limits required for storage of assemblies in the 1-out-of-4 for 4.0 w/o U-235 fresh fuel with IFBA configuration.

- Fuel assembly loading requirements at the interface between the different fuel storage configurations.

6.2 Analytical Criteria, Modeling Approaches and Assumptions

As described in Sections 1.3 and 1.5 of the criticality analysis report, the Westinghouse soluble boron credit methodology provides reactivity margin in SFP criticality analysis that may be applied for added flexibility in fuel storage. This allows the Boraflex® in the fuel storage racks to not be credited as a neutron absorber for any of the fuel storage configurations. Following are selected assumptions made in the criticality analysis:

- The following Westinghouse fuel types were conservatively modeled to represent the fuel assemblies residing in the storage configurations:

  - 14 x 14 Standard fuel represented depleted fuel assemblies
  - 14 x 14 Optimized Fuel Assemblies (OFA) represented fresh fuel

  Westinghouse standard fuel is bounding even though V422+ fuel is the present design for the PBNP. This is due to standard fuel being longer than V422+ fuel and use of Zirc-4 as the cladding material (for the standard fuel) which is less absorbent than the Zirlo used in the V422+ fuel design.

- Enrichment of the fresh fuel pellets was assumed to be up to 5.0 w/o U-235. Fresh fuel was conservatively modeled with a UO_2 density of 10.686 g/cc (97.5 percent of theoretical density).

- Fuel assemblies (fresh and depleted) were modeled as containing right-solid cylindrical pellets uniformly enriched over the entire length of the fuel stack height. This assumption bounds assembly designs incorporating annular or lower enrichment fuel pellets such as those used for axial or lower enrichment fuel pellets such as those used for axial blankets.

- The Boraflex® in the fuel storage racks was conservatively omitted and replaced by water in the analysis. The stainless steel material encasing the Boraflex®, however, was modeled.

- Modeling of the IFBA pins in the analysis is proprietary (see proprietary version of criticality analysis report).

- The design basis limit k_{eff} for the zero soluble boron condition was conservatively reduced from 1.0 to 0.995 for the analysis.

- The most reactive SFP temperature (full moderator density of 1 g/cc) was used for each fuel storage configuration such that the analysis results are valid over the nominal spent fuel temperature range (50 to 180°F).

- Infinite lattice analyses were used to evaluate the fuel storage racks reactivity characteristics. This approach was applied to evaluate burnup limits versus initial enrichment and physical tolerances and uncertainties.

- A two (2) foot thick water reflector was modeled above and below the storage cell geometry. The pool water was simulated to be full density (1 g/cc) at room temperature (68°F) conditions. The top and bottom surfaces of the water reflector had reflected boundary conditions.

Section 5 of this enclosure summarizes the design of the SFP and fuel storage rack cells (see Sections 2.2 and 2.3 of the criticality analysis report).

Fuel storage configurations were modeled in KENO as a repeating 2 x 2 arrays of storage cells. For the All-Cell storage configuration all four cells in the 2 x 2 array contained depleted standard fuel assemblies. For the '1-out-of-4 5.0 w/o U-235 fresh fuel with no IFBA' storage configuration one of the four depleted standard fuel assemblies in the array was replaced with a fresh 5.0 w/o U-235 OFA fuel assembly. For the '1-out-of-4 4.0 w/o U-235 fresh fuel with IFBA' storage configuration one of the four depleted standard fuel assemblies in the array was replaced with a fresh 4.0 w/o U-235 OFA fuel assembly. Fresh OFA fuel assemblies with enrichments greater than 4.0 w/o U-235 contain IFBA pins. Depictions of the three proposed fuel storage configurations are provided by Figures 3-1, 3-2, and 3-3 in the criticality analysis report.

6.3 Computer Codes

This section describes the analysis methodology used to assure the criticality safety of the SFP and to define the limits to be placed on fresh and spent (depleted) fuel assembly storage configurations. The criticality analyses employed the SCALE-PC and the Discrete Integral Transport computer codes. SCALE-PC was used for the calculations involving infinite arrays for all storage configurations in the SFP. The two-dimensional Discrete Integral Transport code was used for simulation of in-reactor fuel assembly depletion.

SCALE-PC

The SCALE computer code (Reference 15) was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC includes the KENO V.a code, a three-dimensional Monte Carlo criticality code. SCALE-PC is a personal computer version of the SCALE-4.4a code system which includes the updated SCALE-4.4a version of the 44-group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross section library. SCALE-PC was used both for benchmarking and for the calculations involving infinite arrays for storage configurations in the SFP. The issues with the KENO code described in NRC Information Notices 91-26, 2005-13 and 2005-31 (References 16, 17 and 18 respectively) are not applicable for the reasons discussed in Section 7.3 of this enclosure.

Validation of SCALE-PC

As discussed in Section 1.4.2 of the criticality analysis report, validation of the SCALE-PC computer code for fuel storage rack analyses was based on analysis of selected critical experiments from two experimental programs: the Babcock and Wilcox (B&W) experiments in support of Close Proximity Storage of Power Reactor Fuel (Reference 19) and the Pacific Northwest Laboratory (PNL) Program in support of the design of Fuel Shipping and Storage Configurations. Nineteen experimental configurations from the B&W and eleven configurations from the PNL experimental programs were selected. For both the B&W and PNL experiments, the full environment of the active fuel region was represented explicitly. Tables 1-1 and 1-2 of Reference 2 summarize the results of these analyses performed with both the 44-group and 238-group libraries. The NRC has previously accepted the use of this data for benchmarking the KENO V.a code under storage conditions similar to those proposed in this amendment request (see Section 7.4).

Determination of the mean calculational bias and variance is described in detail in Section 1.4.2 of the criticality analysis report. The magnitude of $k_{95/95}$ is computed for a given KENO-calculated value of k_{eff} (and associated one sigma uncertainty) by the equation below. This approach provides a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than of $k_{95/95}$.

$$k_{95/95} = k_{keno} + \Delta k_{bias} + M_{95/95} (\sigma_m^2 + \sigma_{keno}^2)^{1/2}$$

where,

k_{keno} is the KENO-calculated multiplication factor

Δk_{bias} is the mean calculational method bias

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses

σ_m^2 is the mean calculational method variance deduced from the validation analyses

σ_{keno}^2 is the square of the KENO standard deviation

Determination of the mean calculational methods bias, the mean calculational variance, and the 95/95 confidence level multiplier were determined as described in detail in Section 1.4.2 of the criticality analysis report.

Discrete Integral Transport Code

The Discrete Integral Transport (DIT) code (Reference 20) performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The DIT code is used for simulation of in-reactor fuel assembly depletion. The multigroup cross sections utilized in DIT are based on the ENDF/B-VI neutron cross section library.

The DIT code and its cross section library are used in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data. For the purpose of SFP criticality analysis calculations, the DIT code was used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment.

6.4 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology was the inclusion of an axial burnup profile correlated with the feed enrichment and discharge burnup of the depleted fuel assemblies. This effect is important since the majority of spent fuel assemblies stored in the SFP have a discharge burnup well beyond that for which the assumption of a uniform axial burnup shape is conservative.

Burnt fuel assemblies were divided up into multiple axial zones and represented with a limiting axial burnup profile in the analysis. The DIT computer code generated isotopic concentrations for each segment of the axial burnup profile. Fuel temperatures for each axial zone were calculated based on a representative fuel temperature correlation while the moderator temperatures were based on a linear relationship with axial position. This moderator, fuel temperature, and power profile data were used in the DIT code to deplete the fuel to the desired burnup for each initial enrichment and axial zone.

6.5 Tolerance / Uncertainty Calculations

Analytical models were developed to perform evaluations to demonstrate that the k_{eff} of the SFP will be less than 0.995 with zero soluble boron present in the pool water. Applicable biases factored into the evaluation included: 1) methodology bias deduced from the validation analyses of pertinent critical experiments, and 2) any reactivity bias, relative to the reference analysis conditions associated with operation of the SFP over a temperature range of 50° to 180°F. Cases for nominal conditions were performed assuming full moderator density (1 g/cc), corresponding to 40°F, which is less than the normal SFP temperature range and more conservative. A 95 / 95 confidence level assessment of the various tolerances and uncertainties was performed which included:

Confidence level calculational uncertainties and methods variance: Described in Subsection 1.4.2 of the criticality analysis report.

Fuel pin manufacturing tolerance: Was assumed to consist of an increase in fuel enrichment of 0.05 w/o U-235. An increase in UO_2 density was not assumed since the calculations used 97.5 percent of theoretical density (highest credible density for PWR fuel). Individual contributions were combined by taking the square root of the sum of the squares of each component.

Storage rack fabrication tolerances: The inner stainless steel canister inside dimension was increased, the thickness of the canister was decreased, and the storage cell pitch was decreased; as described in Section 3.4 of the criticality analysis to determine the effects on reactivity.

Tolerance due to positioning the fuel assembly in the storage cell: Nominal and off-center positioning cases were performed for each fuel assembly storage configuration as described in Section 3.4 of the criticality analysis.

Burnup and IFBA manufacturing uncertainty: The uncertainty for the storage configurations and the manufacturing tolerance and calculational uncertainties associated with the IFBA pins B-10 poison loading (proprietary) were determined.

Tables 3-4 through 3-6 in the criticality analysis report provide a tabular summary of the KENO computer code results used in the calculation of the biases and uncertainties for the fuel assembly storage configurations.

6.6 No Soluble Boron 95/95 k_{eff} Calculational Results

Burnup versus initial enrichment limits for a target k_{eff} value at zero soluble boron were determined for each of the following fuel storage configurations.[1] and for the interfaces between fuel storage configurations.

- All-Cell
- 1-out-of-4 for 5.0 w/o U-235 fresh fuel with no IFBA
- 1-out-of-4 for 4.0 w/o U-235 fresh fuel with IFBA fuel storage;

For each storage configuration, the k_{eff} values over a range of initial enrichments and assembly average burnups, with an axially distributed burnup profile, were determined. The sum of the biases and uncertainties was determined for each configuration and subtracted from the design basis limit k_{eff} (reduced from 1.0 to 0.995 to cover analytical biases and uncertainties) for the zero soluble boron condition to arrive at the target k_{eff} value.

Storage Configurations	Range of Initial Enrichment (w/o U-235)	Assembly Average Burnups (MWD/MTU)	Sum of Biases / Uncertainties (Δk_{eff} Units)	Target k_{eff}
All-Cell	Up to 5.0	Up to 45,000	0.02492	0.97008
1-out-of-4 5.0 w/o U-235 fresh fuel with no IFBA	Up to 5.0	Up to 55,000	0.01909	0.97591
1-out-of-4 4.0 w/o U-235 fresh fuel with IFBA	Up to 5.0	Up to 55,000	0.02079	0.97421

Based on the target k_{eff} value, the interpolated enrichment for no burnup was determined. The derived burnup limits, for enrichments greater than the target k_{eff} value, were based on the k_{eff} values for 3.0, 4.0, and 5.0 w/o U-235. For each of these enrichments KENO calculations were performed at several assembly average burnup values for an axially distributed burnup profile. A second degree fit of burnup versus k_{eff} data was used to determine the burnup required to meet the target k_{eff} value.

Resulting burnup versus initial enrichment storage limits for 0, 5, 10, 15, and 20 years of decay time were determined. Limiting burnups as a function of initial enrichment were fitted to a third degree polynomial for each decay period. These polynomials were used to determine the burnup as a function of initial enrichment for each fuel storage configuration.

For fuel designs containing IFBAs, for each fresh fuel enrichment and number of IFBA pins, k_{eff} was evaluated for different burnups of the depleted fuel assemblies with an initial enrichment of 5.0 w/o U-235. A second degree fit of burnup versus k_{eff} data was performed using the target k_{eff} value which was the

1 Requirements were also determined for storage of fuel pins in the Failed Fuel Rod Storage Basket and within the guide tubes in the All-Cell storage configuration.

same as that used to determine the burnup requirements for the depleted fuel assemblies for that configuration. Accordingly, the required number of IFBA pins as a function of initial enrichment was fitted to a second degree polynomial, which was used to determine the number of IFBA pins as a function of initial enrichment for this, the '1-out-of-4 4.0 w/o U-235 fresh fuel with IFBA' fuel storage configuration.

Interface Requirements between Fuel Storage Configurations: KENO computer models were constructed to analyze fuel assembly loading requirements at the interfaces between different configurations. For storage configurations involving high and low reactivity assemblies, i.e. for 1-out-of-4 configurations, the assemblies with the lower reactivity must be placed at the interface. Interface requirements are depicted in the TS figures. It is acceptable to leave a storage cell in a fuel rack empty.

Burnup Requirements for Intermediate Decay Time Points: At least a second order polynomial is used to determine burnup requirements for intermediate decay time points for those storage configurations crediting Pu-241 decay.

Empty Cells: An empty cell is permitted in any location for all storage configurations since the water filled cell decouples the neutronic interaction between the spent fuel assemblies in the pool and, therefore, not cause any increase in reactivity. Non-fissile material and debris canisters may be stored in empty cells of All-Cell storage configuration provided that the canister does not contain fissile materials.

6.7 Soluble Boron

In an enclosure to the safety evaluation for WCAP-14416 (Reference 9) the total soluble boron requirement was defined as the sum of three quantities:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

The soluble boron required to maintain k_{eff} less than or equal to 0.95, account for *burnup and reactivity uncertainties, and mitigate various accidents is summarized* below.

Soluble Boron Credit for 95/95 k_{eff} Less Than or Equal to 0.95 ($SBC_{95/95}$)

The initial enrichment and burnup were chosen to represent the storage configuration based on minimizing the soluble boron worth. Soluble boron worth decreases with increasing burnup. The KENO model assumed the SFP was filled with the All-Cell configuration containing depleted fuel at 45,000 MWD/MTU with 5.0 w/o U-235 initial enrichment. SFP k_{eff} values were calculated for soluble boron in increments. The soluble boron required to reduce k_{eff} by 0.05 Δk_{eff} units was determined to be 270.6 ppm.

Soluble Boron Credit to Account for Burnup and Reactivity Uncertainties (SBC_{RE})

Fuel assembly reactivity uncertainty was calculated by employing a depletion reactivity uncertainty and multiplying by the maximum burnup credited for a storage configuration. The maximum burnup was 47,000 MWD/MTU for the '1-out-of-4 5.0 w/o fresh fuel with no IFBA' storage configuration resulting in a depletion reactivity uncertainty of 0.015667 Δk_{eff}.

Fuel burnup uncertainty was conservatively calculated as 5 percent of the maximum fuel burnup credited in a storage configuration analysis. The maximum reactivity change associated with a 5 percent change in burnup was 0.007143 Δk_{eff} units for the All-Cell storage configuration.

The soluble boron credit required for uncertainties in reactivity and burnup effects was determined to be 0.022810 Δk_{eff} to the corresponding soluble boron concentration of 118.9 ppm.

Soluble Boron Credit Required to Offset Most Limiting Single Accident (SBC_{PA})

The soluble boron concentration required to mitigate accidents was determined by identifying possible accidents or conditions that could increase the k_{eff} of the SFP and determining that which produced the largest increase (maximize the required concentration). The accidents / conditions considered included:

- Dropping a fresh fuel assembly on top of the fuel storage racks,
- Misloading a fresh assembly into an incorrect fuel storage rack location,
- Placing a fresh assembly into a location outside the fuel storage racks,
- A SFP temperature greater than 180°F.

Fuel mishandling events were simulated with the KENO model to assess the k_{eff} increase in the SFP. A fresh Westinghouse 14 x 14 OFA fuel assembly enriched to 5.0 w/o U-235 (with no burnable poisons) was assumed misloaded into a fuel storage rack cell or lowered into the cask area between the storage racks. These accidents / conditions are discussed below.

Fresh fuel assembly dropped on top of the fuel storage racks

The physical separation between fuel assemblies in the fuel storage racks and a dropped fuel assembly lying on top of the racks, is sufficient to neutronically decouple the accident, i.e., it does not result in a positive reactivity increase.

The design of the fuel storage rack modules and fuel handling equipment in the SFP is such that it precludes the insertion of a fuel assembly between the fuel storage rack modules.

Misloading a fresh fuel assembly into an incorrect fuel storage rack location, location outside the fuel storage racks, or SFP temperature greater than the operating range

Cool-down events produce less positive reactivity change compared to heat-up events. Therefore, heat-up events were evaluated. Nominal cases were developed by filling up the SFP with one of the fuel storage configurations and then the accident scenarios, described above, were applied. This was repeated for each storage configuration. Both nominal cases and accident scenarios were simulated assuming zero ppm boron using depleted fuel isotopics in the SFP. The results are shown below.

Accident Scenarios	**All-Cell (ppm)**	**1-out-of-4 5.0 w/o Fresh with no IFBA (ppm)**	**1-out-of-4 4.0 w/o Fresh with IFBA (ppm)**
Misloaded fresh fuel assembly into burnup fuel storage rack location	402.9	306.0	365.4
Misloaded fresh fuel assembly in the cask area between fuel storage racks	1.0	17.1	---
SFP temperature greater than normal operating range (240°F)	125.8	---	46.0

The accident / condition that produced the largest increase in the SFP k_{eff} was the misloading of a fresh 5.0 w/o U-235 enrichment fuel assembly in an incorrect storage rack location in the 'All-Cell' configuration with the configuration containing depleted fuel assemblies with a 5.0 w/o U-235 initial enrichment at a burnup of 45,000 MWD/MTU. The required soluble boron concentration necessary to mitigate this accident (and therefore all other less severe accidents) was determined to be 402.9 ppm.

Total Soluble Boron Requirement (SBC_{TOTAL})

The total soluble boron required is summation of the following factors:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

$$= 270.6 \text{ ppm} + 118.9 \text{ ppm} + 402.9 \text{ ppm} = 792.4 \text{ ppm}$$

The soluble boron credit (SBC_{TOTAL}) determined assuming a Boron-10 concentration of 19.9 atom percent was 792.4 ppm. A more conservative total soluble boron credit assuming a Boron-10 concentration of 19.6 atom percent (expected lowest value crediting depletion) was 804.5 (rounded up to 805) ppm.

A soluble boron concentration of 805 ppm assures that the k_{eff} is less than or equal to 0.95 accounting for burnup and reactivity depletion uncertainties and including the most limiting postulated single accident. For an occurrence of the postulated accident conditions, the double contingency principle discussed in ANSI/ANS-57.2 and an April 1978 NRC letter (References 11 and 12, respectively) can be applied. The double contingency principle states that the criticality analyses are not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for the postulated accident conditions, the presence of the soluble boron in the SFP can be assumed as a realistic initial condition, since not assuming its presence would be a second unlikely event.

Specification 3.7.11, "Fuel Storage Pool Boron Concentration," requires the SFP boron concentration to be maintained greater than or equal to 2100 ppm whenever fuel is stored in the SFP and requires verification of the concentration every 7 days, consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," and the NRC safety evaluation for WCAP-14416. Therefore, margin is maintained to the SFP boron concentration of 805 ppm determined in the PBNP plant-specific criticality analysis (WCAP-16541) as that required to maintain k_{eff} less than 0.95 in the event of the most limiting postulated single accident (including uncertainties). This margin provides sufficient time to detect and mitigate a dilution event before the spent fuel rack criticality analysis of $k_{eff} < 0.95$ would be exceeded (if an event were to occur).

6.8 Spent Fuel Pool Dilution Analysis

The NRC approved the Westinghouse soluble boron credit methodology described in WCAP-14416 with a requirement that all licensees applying this methodology provide a plant-specific boron dilution analysis. The NRC safety evaluation (Reference 9) for the licensing topical report states:

> All licensees proposing to use the new method described above for soluble boron credit should identify potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit ...and should quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. The effects of incomplete boron mixing, such as boron stratification, should be considered. This analysis should be submitted for NRC review and should also be used to justify the surveillance interval used for verification of the technical specification minimum pool boron concentration.

A SFP boron dilution analysis to support the new PBNP Units 1 and 2 criticality analysis is provided in Enclosure 5. The SFP dilution analysis included an evaluation of the following.

- Boron Dilution Initiating Events
- Boron Dilution Volumes and Times

- Dilution Sources
- Spent Fuel Pool Instrumentation
- Administrative Procedures

The boron dilution analysis assumed thorough mixing of the non-borated water postulated to be added with the contents of the SFP. A conservatively determined SFP volume of approximately 236,400 gallons is turned over approximately every 3 hours with one SFP pump running (the normal alignment). It is unlikely with this cooling flow and convection from the spent fuel decay heat that thorough mixing would not occur, but if mixing were inadequate, a localized pocket of non-borated water could conceivably form. Results of the PBNP plant-specific criticality analysis (WCAP-16541) demonstrated that the spent fuel rack k_{eff} is less than 1.0 even if the pool were filled with non-borated water. Thus if a non-borated pocket formed, k_{eff} would not exceed 1.0 anywhere in the pool.

The plant-specific PBNP criticality analysis determined the soluble boron concentration required to maintain the SFP $k_{eff} < 0.95$, including uncertainties was 805 ppm. Existing Specification 3.7.11 requires a minimum boron concentration of 2,100 ppm. (No change to Specification 3.7.11 is requested.)

Summary of Dilution Events

Based on the evaluation of potential dilution sources in the analysis, the following five dilution sources (in order from least to most likely) were determined capable of providing a significant amount of non-borated water to the SFP. The required dilution volume exceeds the volume of all unborated water sources in the plant used for normal makeup with the exception of the Demineralized Water System.

- Fire Protection (from a hose station)
- Monitor Tanks (through demineralizer flushing)
- Reactor Makeup Water (RMUW) Storage Tank
- Chemical and Volume Control System (CVCS) Holdup Tank
- Demineralized (DI) Water

Fire protection was determined to be the least likely source since it would only be used as a measure of last resort for a loss of inventory accident and because the makeup hose is not located in the vicinity of the SFP. The monitor tanks were the next least likely source since they are used only for demineralizer flushing and are not for normal or emergency makeup. The RMUW Storage Tank was the next least likely source because there is not a direct makeup path to the SFP (and it is not the preferred source because of the required valve lineup). Dilution from a CVCS Holdup Tank was the second most likely, but the tanks are normally borated to some degree and the volume of all three tanks is less than that required to dilute the SFP from 2,100 to 805 ppm.

The DI Water System was determined to be the most likely source because it has a direct connection to the SFP (is the preferred makeup source) and there is an unlimited supply from the water treatment plant. Maximum flow is 400 gpm with two pumps running. Typically only one pump and demineralizer are in service, limiting output to 200 gpm.

Note the following discussion outlines the SFP overflow scenario. At the maximum DI flow rate, it would take 249 minutes to fill the SFP to the high level alarm assuming the SFP level was initially at the low level alarm. If the fuel transfer canal were full, as is the normal case, the high level alarm would alert the control room much sooner. Assuming that the high level alarm were to fail, the SFP would overflow, spilling onto the refueling floor, resulting in water filling the Plant Auxiliary Building sump. If the flow exceeded the capacity of the drains, it would flow onto the refueling deck and into other parts of the building. The water would eventually end up in the Waste Holdup Tank (24,000 gallon approximate capacity) where the high and high-high alarms would act as a secondary backup to the SFP high level alarm. By procedure, the operator must inform the water treatment operator prior to filling the SFP. Continued makeup to the SFP should be noticed by the water treatment operator. In addition, it would take 10 hours to reach the required dilution volume and routine operator rounds of the area would identify the overflow of the SFP.

<u>Spent Fuel Pool Instrumentation</u>

SFP water level and temperature is available and alarms on a common annunciator in the control room. The alarm actuates on high SFP temperature (120°F) and high or low SFP level. The temperature and level alarms are powered from the vital DC power supply. Additional instrumentation is provided to monitor the pressure and flow of the SFP cleanup system, and pressure, flow and temperature of the SFP cooling system. Two area radiation monitors are available in the SFP area for low range and high range area monitoring.

<u>Administrative Controls</u>

The following administrative controls are in place to control the SFP boron concentration and water inventory:

1. Procedures to identify and terminate dilution events.

2. Procedures for loss of inventory ordered so borated water sources are used first.

3. The procedure for makeup allows use of DI water, RMUW, and CVCS hold-up tank water provided additional requirements for sampling and/or initial boron concentration are met.

4. Plant personnel perform rounds at the SFP at least once every 8 hours and records temperature and level (level once per day).

5. Administrative controls, locked closed valves are used on the RMUW flow paths to the SFP cooling system.

6. Procedural controls are placed on potential dilution paths.

7. Procedures require SFP initial boron concentration to be greater than or equal to 2,500 ppm when using RMUW or DI water for makeup, SFP cooling system to be operating (flow rate greater than or equal to 1,000 gpm) and makeup is limited to 12 inches of level. Prior to adding additional water, the concentration must be re-verified to be greater than 2,500 ppm.

8. The CVCS holdup tanks must meet chemistry requirements including boron concentration before being used as a makeup source.

9. SFP boron concentration is administratively maintained at greater than 2,300 ppm (typically around 3,000 ppm) and sampled every 7 days per TS.

As stated above, the SFP boron concentration is typically maintained around 3,000 ppm. If the concentration decreases to less than the TS 3.7.11 limit of 2,100 ppm, an Action statement is immediately entered to restore the concentration. The dilution analysis determined the limiting source / time to reduce the boron concentration from the TS limit of 2,100 to 805 ppm. The SFP volume was conservatively determined to be approximately 236,400 gallons. Approximately 251,000 gallons of non-borated water would be required to reduce the SFP volume from the TS minimum concentration limit to the concentration corresponding to a SFP $k_{eff} < 0.95$, i.e., 805 ppm. To provide this dilution volume, an operator would have to initiate the dilution flow, abandon monitoring SFP level, ignore administrative procedures, and ignore a SFP high-level alarm for a minimum period of at least 10 hours. The required dilution volume exceeds the volume of all unborated water sources in the plant used for normal makeup with the exception of the DI Water System.

The boron dilution analysis concluded that an unplanned or inadvertent dilution reducing the SFP concentration from 2,100 to 805 ppm was not credible. It also demonstrated that sufficient time is available to detect and mitigate a dilution event before the spent fuel rack criticality analysis $k_{eff} < 0.95$ were exceeded.

6.9 Fuel Assembly Burnup Determination

Fuel assembly burnup is a key input for determining how and where a fuel assembly may be stored in the SFP. Fuel assembly burnup values are determined using software approved under the software quality assurance (SQA) program. The software uses fuel vendor generated isotopic data, data from incore detector readings, and reactor operating history data. Qualified personnel use the software and independent reviews of input and output data are performed by another qualified individual.

Burnup data input into software for planning fuel movements is reviewed by qualified personnel. Fuel movement planning software is approved under the

SQA program. Updates to the software data files are controlled by procedure and require independent review. Fuel movement sequences are planned by qualified individuals and reviewed independently by another qualified individual. Fuel assembly movements are verified administratively through use of a plant procedure separate from the fuel movement planning software.

6.10 Results Summary and Safety Assessment

The proposed TS changes and resulting fuel (both fresh and depleted) storage limits ensure that the stored fuel assembly array remains subcritical for all the analyzed fuel storage configurations. The computer codes, methods and techniques described in the Westinghouse criticality analysis methodology licensing topical, WCAP-14416, were used to satisfy the regulatory requirements for criticality control of fuel in storage, i.e., 10 CFR 50.68 and 10 CFR 50, Appendix A, General Design Criterion 62. The Westinghouse methodology uses industry accepted analysis codes and methods which have been benchmarked for SFP criticality analyses crediting soluble boron. The specific PBNP fuel storage requirements delineated in the TS are based on a plant-specific criticality analysis (WCAP-16541) performed in accordance with the Westinghouse spent fuel rack criticality analysis methodology described in licensing topical report WCAP-14416;crediting soluble boron, fuel burnup and initial enrichment, integral fuel burnable absorber pins, and Plutonium-241 decay, where applicable. The revisions to the specifications (and/or associated Bases) listed below incorporate the results of the analysis.

- Specification 3.7.11 - Fuel Storage Pool Boron Concentration (Bases)
- Specification 3.7.12 - Spent Fuel Pool Storage
- Specification 4.3.1 - Criticality

Specification 3.7.12 defines the acceptable range for fuel storage in the All-Cell configuration via new Figure 3.7.12-1 (as a function of initial fuel assembly enrichment, burnup and decay time). Specification 4.3.1.1 has been revised to be consistent with 10 CFR 50.68(b). For fuel assemblies not meeting the requirements of Specification 3.7.12, Figures 4.3.1-1 through 4.3.1-8 in Specification 3.7.12 provide criteria for the approved analyzed fuel storage configurations, interfaces between configurations, burnup versus initial enrichment requirements as a function of Pu-241 decay time, and IFBA requirements.

The plant-specific criticality analysis determined the 'All-Cell,' '1-out-of-4 for 5.0 w/o U-235 fresh fuel with no IFBA,' and '1-out-of-4 for 4.0 w/o U-235 fresh fuel with IFBA' were all acceptable fuel storage configurations to ensure that the SFP would remain subcritical, with credit taken for Pu-241 decay and the number of IFBA pins (as applicable). All of the proposed TS fuel storage configurations / requirements correspond explicitly to those determined by the plant-specific criticality analysis, and hence have been determined to be safe and to meet the requirements of 10 CFR 50.68.

The required soluble boron credit determined for the bounding cases for the plant-specific fuel storage configurations / conditions by the PBNP criticality analysis (WCAP-16541) were:

• Soluble Boron for 95/95 $k_{eff} \leq 0.95$ ($SBC_{95/95}$)	270.6
• Soluble Boron to Account for Burnup and Reactivity Uncertainties (SBC_{RE})	118.9
• Soluble Boron to Offset Most Limiting Single Accident (SBC_{PA})	402.9
• Total Soluble Boron Credit (SBC_{TOTAL}) (w/B-10 depl.)	792.4(805)

The plant-specific soluble boron concentration of 805 ppm ensures that the SFP k_{eff} will be less than or equal to 0.95 accounting for burnup and reactivity depletion uncertainties and including the most limiting postulated single accident. Specification 3.7.11 (unchanged) requires the boron concentration to be maintained greater than or equal to 2100 ppm. This is verified via SR 3.7.11.1 every 7 days (consistent with NUREG-1431 and the safety evaluation for WCAP-14416). Thus, margin is maintained to SFP criticality analysis limit boron concentration of 805 ppm.

The loss of substantial amounts of soluble boron from the SFP that could lead to k_{eff} exceeding 0.95 was evaluated as part of the analyses in support of this license amendment request. The dilution analysis demonstrates that a dilution of the SFP boron concentration from the minimum TS concentration of 2100 to 805 ppm was not credible. The results of the dilution analysis demonstrate that sufficient time is available for operators to detect and mitigate a dilution event before the spent fuel rack criticality analysis limit of $k_{eff} < 0.95$ would be exceeded (if an event were to occur). Also, the plant-specific criticality analysis results demonstrate, in accordance with the requirements of 10 CFR 50.68, that even if a complete dilution of the SFP were to occur the spent fuel rack k_{eff} would remain < 1.0 (at a 95/95 percent probability and confidence level) with the pool flooded with unborated water.

The PBNP plant-specific analyses (WCAP-16541), performed in accordance with the Westinghouse soluble boron credit methodology, provide a safe, conservative approach demonstrating that the SFP will remain subcritical during normal and postulated accident conditions. Operation of PBNP in accordance with the proposed TS resulting from these analyses ensures that the requirements of 10 CFR 50.68 are met with no Boraflex® assumed in the fuel storage racks. The licensing basis changes and associated TS changes will continue to protect the health and safety of the public.

7.0 REGULATORY ANALYSIS

7.1 No Significant Hazards Determination

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) is proposing to revise the Technical Specifications for the Point Beach Nuclear Plant, Units 1 and 2, to reflect the application of the Westinghouse soluble boron credit methodology to the spent fuel pool (SFP) criticality analysis. The Westinghouse criticality analysis determined acceptable storage conditions for fuel in the SFP fuel storage racks with credit for burnup, Integral Fuel Burnable Absorber pins, Plutonium-241 decay and soluble boron, where applicable. Associated changes are proposed to the Technical Specifications for storage of fuel in the SFP and design features for criticality control to reflect the results of the criticality analysis.

NMC has evaluated the proposed amendments in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92, "Issuance of Amendment," and has determined that the operation of the Point Beach Nuclear Plant in accordance with the proposed amendments presents no significant hazards. The NMC evaluation against each of the criteria in 10 CFR 50.92 follows.

1) **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   Operation of the facility in accordance with the proposed amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated. The presence of soluble boron in the Spent Fuel Pool (SFP) water being used for criticality control does not increase the probability of a dropped fuel assembly accident within the pool. The handling of the fuel assemblies in the SFP has always been performed and will continue to be performed in borated water.

   There is no increase in the probability of the accidental misloading of fuel assemblies into the SFP fuel storage racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and in accordance with the spent fuel storage rack limitations specified in the Technical Specifications (TS). There is no increase in the consequences for an accidental misloading of fuel assemblies in the SFP fuel storage racks because the criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading.

Soluble boron credit is used to provide margin to offset uncertainties, tolerances, and off-normal / accident conditions, and to provide subcritical margin such that the SFP k_{eff} is maintained less than or equal to 0.95. The plant-specific criticality analysis results demonstrate that the spent fuel rack k_{eff} will remain < 1.0 (at a 95/95 percent probability and confidence level) even with the SFP flooded with unborated water.

There is no increase in the probability of the loss of normal cooling to the SFP water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

A loss of normal cooling to the SFP water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density, which would result in a net increase in reactivity when soluble boron is present in the water. However, the additional negative reactivity provided by the 2100 ppm boron concentration limit, above that provided by the concentration required (805 ppm) to maintain k_{eff} less than or equal to 0.95, will compensate for the increased reactivity which could result from a loss of SFP cooling event. Because adequate soluble boron will be maintained in the SFP water the consequences of a loss of normal cooling to the SFP will not be increased.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No

Under the proposed amendment, no changes are being made to the fuel storage racks themselves, to any other systems, or to the physical structures of the Primary Auxiliary Building. Therefore, there are no changes proposed to the plant configuration, equipment design, or installed equipment.

Criticality accidents in the SFP are not new or different types of accidents. They have been analyzed in the FSAR and in fuel storage criticality analysis reports associated with specific licensing amendments. The proposed new SFP storage limitations are consistent with the assumptions made in the new criticality analysis, and will not have any significant effect on normal SFP operations and maintenance, and do not create the possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the SFP loading configuration meets specified requirements.

The current TS includes a SFP boron concentration limit that conservatively bounds the boration assumption of the new criticality analysis. Since soluble boron has always been maintained in the SFP water, implementation of this requirement for SFP criticality control purposes has have no effect on normal pool operations and maintenance. Also, since soluble boron has always been present in the SFP, a dilution event has always been a possibility. The loss of substantial amounts of soluble boron from the SFP that could lead to k_{eff} exceeding 0.95 was evaluated as part of the analyses in support of this license amendment request. The evaluation demonstrates that a dilution of the SFP boron concentration from the minimum TS concentration of 2100 to 805 ppm is not credible.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3) **Does the proposed amendment result in a significant reduction in a margin of safety?**

Response: No

The proposed Technical Specification changes providing the resulting spent fuel storage operation limits provide adequate safety margin to ensure that the stored fuel assembly array always remains subcritical. These limits are based on a plant-specific criticality analysis performed in accordance with the present Westinghouse spent fuel rack criticality analysis methodology which allows credit for soluble boron.

The criticality analysis takes credit for soluble boron to ensure that k_{eff} will be less than or equal to 0.95 under normal circumstances. While the criticality analysis used credit for soluble boron, storage configurations have been defined using 95/95 k_{eff} calculations to ensure that the spent fuel rack k_{eff} is less than unity (0.995) with no soluble boron. Soluble boron credit is used to provide safety margin to offset uncertainties, tolerances, and off-normal / accident conditions, and to provide subcritical margin such that the SFP k_{eff} is maintained less than or equal to 0.95.

The loss of substantial amounts of soluble boron from the SFP that could lead to k_{eff} exceeding 0.95 was evaluated as part of the analyses in support of this license amendment request. The evaluation demonstrates that a dilution of the SFP boron concentration from the minimum TS concentration of 2100 to 805 ppm is not credible. Also, the plant-specific criticality analysis results demonstrate that even if a complete dilution were to occur the spent fuel rack k_{eff} would remain < 1.0 (at a 95/95 percent probability and confidence level) with the SFP flooded with unborated water. The plant-specific criticality analysis performed in accordance with the conservative analysis methodology of the Westinghouse licensing topical report demonstrates that the requirements of 10 CFR 50.68 and 10 CFR 50, Appendix A, General Design Criterion 62 will be satisfied. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion

Operation of the Point Beach Nuclear Plant Units 1 and 2 in accordance with the proposed license amendment will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and, does not result in a significant reduction in any margin of safety. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not result in a significant hazards determination.

7.2 10 CFR 50.68 Compliance

The following summarizes the NMC plan for compliance with 10 CFR 50.68, "Criticality Accident Requirements" for PBNP. Prior to approval of this amendment request, an exemption from the requirements of 10 CFR 70.24 applies to criticality control at PBNP.

10 CFR 50.68(b)(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.

Plant procedures do not allow movement of fuel assemblies when the soluble boron concentration is below the TS limit. Maintaining the SFP at the required boron concentration ensures the pool will remain subcritical, even in the event of a mispositioned or dropped fuel assembly.

Plant procedures for new fuel receipt prohibit more than one fuel assembly out of a shipping container or new fuel vault storage location at a time. During refueling operations only one fuel assembly can be moved at a time in the SFP as this is a physical limitation of the crane. A second hoist installed on the crane is not qualified for fuel handling and is administratively restricted from being used for fuel handling.

Movement of fuel assemblies is controlled by plant procedures. Qualified personnel are responsible for planning fuel movement sequences. Fuel movement sequences are reviewed independently by qualified personnel. Fuel movements are performed by qualified plant operators. All fuel handling operations are overseen by a senior reactor operator (SRO) and a qualified individual. Fuel movements are independently verified by the SRO and the qualified individual.

Storage of fuel assemblies will be procedurally controlled to assure the k_{eff} remains below 1.0, at a 95 percent probability, 95 percent confidence level, when flooded with unborated water. The fuel storage patterns assure subcriticality under the most adverse moderation conditions by unborated water. The proposed fuel storage patterns ensure reactivity is less than 1.0 (analyzed at 0.995 for additional conservatism) at a 95 percent probability with a 95 percent confidence level when filled with unborated water and is less than or equal to 0.95 at a 95 percent probability with a 95 percent confidence level when credit is taken for 805 ppm boron.

10 CFR 50.68(b)(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.

Criticality analyses performed for the fresh fuel storage racks have demonstrated that the k_{eff} does not exceed 0.95 at a 95 percent probability, 95 percent confidence level. The fresh fuel storage racks are used at the PBNP when required for unloading new fuel assemblies or control rods. The fresh fuel storage racks is designed to hold new fuel assemblies and is utilized primarily for the storage of the replacement fuel assemblies. This requirement of 10 CFR 50.68(b)(2) is already included in the plant TS under Specification 4.3.1.2(b).

10 CFR 50.68(b)(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.

Criticality analyses performed for the fresh fuel storage racks have demonstrated that k_{eff} will not exceed 0.98 at a 95 percent probability, 95 percent confidence level under optimum moderation. This requirement of 10 CFR 50.68(b)(3) is already included in the plant TS under Specification 4.3.1.2(c).

10 CFR 50.68(b)(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.

> This new criticality analysis demonstrates through the use of administrative controls that the SFP k_{eff} will not exceed 0.95, at a 95 percent probability, 95 percent confidence level when flooded with water borated to 805 ppm; and will remain below 1.0 (analyzed at 0.995) at a 95 percent probability, 95 percent confidence level if flooded with unborated water.

10 CFR 50.68(b)(5) - The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

> All non-fuel SNM stored onsite is tracked in a database. Plant procedures for tracking non-fuel SNM will be updated to ensure the site has a quantity less than that necessary for critical mass as defined by 10 CFR 150.11.

10 CFR 50.68(b)(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.

> Two area radiation monitors are installed in the SFP area. These radiation monitors are connected to the plant radiation monitoring system and will alarm locally and in the control room. In addition, an area monitor is required to be operating on the SFP bridge crane during movement of fuel assemblies in the pool.

10 CFR 50.68(b)(7) - The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

> Specification 4.3.1.2(a) limits the enrichment of the fuel assemblies stored in the fresh fuel storage racks to 5.0 percent by weight. The new criticality analysis analyzes fuel for storage in the SFP up to 5.0 w/o. Revised Specification TS 4.3.1.1 (a) will require the maximum enrichment of fuel stored in the SFP to be 5.0 w/o.

10 CFR 50.68(b)(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).

> PBNP will update the FSAR in accordance with this requirement.

7.3 Applicable Regulatory Requirements / Guidance

The PBNP was licensed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC) to 10 CFR Part 50 and before the Standard Review Plan (SRP), NUREG-0800 was promulgated. Hence the

PBNP units were not licensed to the Appendix A GDC. The Point Beach Final Safety Analysis Report (FSAR), Section 1.3, discusses the plant-specific GDCs to which the plant was licensed. The PBNP GDCs are similar in content to the draft GDC proposed for public comment in 1967. The PBNP GDC addressing the prevention of criticality is Point Beach GDC-66, "Prevention of Fuel Storage Criticality," which states,

> Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

This is analogous to 10 CFR 50, Appendix A, GDC-62, which states:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The revised design basis for preventing criticality in the SFP is consistent with the requirements of 10 CFR 50.68(b) satisfying the requirements of GDC-62. Implementation of the proposed TS changes in the required fuel storage configurations and the associated assembly reactivity requirements as determined by the PBNP plant-specific criticality analysis will continue to satisfy the requirements of GDC-62.

Relevant NRC Guidance Documents

- NRC Information Notice (IN) 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the KENO and SCALE Codes" (Reference 16)

  The revised analysis used the newer version (Version 4.4a) of the SCALE computer code package as opposed to Version 3, to which this error pertains, and the 44-group library rather than the 16-group Hansen-Roach library.

- IN 92-21, "Spent Fuel Pool Reactivity Calculations" (Reference 21)

  The modeling problems cited by this notice concerned methodologies not detecting flaws due to the lack of analytical benchmark data sets containing strong neutron absorbers such as Boraflex®. Credit was not taken in the revised criticality analysis for the Boraflex® and therefore, the problem of selecting representative benchmarks did not arise.

- IN 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks" (Reference 22)

  The spent fuel racks at South Texas Project were originally designed with Boraflex®. The revised criticality analysis does not credit Boraflex® as a neutron absorber, therefore, this information notice is no longer applicable.

- NRC IN 2005-13, "Potential Non-Conservative Error in Modeling Geometric Regions in the KENO-V.A Criticality Code" (Reference 17)

  None of the input models involve cylindrical holes with shared boundaries in the analysis for the PBNP, so the criticality analysis was not affected by this error.

- NRC IN 2005-31, "Potential Non-Conservative Error in Preparing Problem-Dependent Cross Sections for use with the KENO-V.A or KENO-V.I Criticality Code" (Reference 18)

  SCALE version 5 has not been implemented for criticality analyses (4.4 or earlier versions were used). The pressurized and boiling water reactor fuel lattices are not of the slab geometry (e.g., plate-type fuel) where this problem was identified.

- NRC Regulatory Issue Summary 2001-12, "Nonconservatism in Pressurized Water Reactor Spent Fuel Storage Pool Reactivity Equivalencing Calculations" (Reference 23)

  The new criticality analysis does not use reactivity equivalencing, so the nonconservatisms identified in this regulatory issue summary were avoided.

NMC concludes that the proposed changes are in accordance with 10 CFR 50.36(c)(3) with regard to maintaining the necessary quality of systems and components, sustaining facility operation within safety limits, and meeting the limiting conditions for operation. These changes also continue to meet the requirements stated in the PBNP FSAR. The proposed changes thus continue to be compliant with the above regulatory requirements and guidance.

7.4 Precedents

The soluble boron credit methodology applied in the new PBNP plant-specific criticality analysis is analogous to that applied in several recent analyses for various plants that have been reviewed and approved by the NRC, including:

- R. E. Ginna Nuclear Power Plant (Reference 24).
- Diablo Canyon Nuclear Power Plant – Units 1 and 2, September 25, 2002. (Reference 25)

- Oconee Nuclear Station – Units 1, 2 and 3, April 22, 2002 (Reference 26)
- Shearon Harris Nuclear Power Plant – Unit 1, March 10, 2006 (Reference 27)

7.5 Commitments

The NMC currently holds an exemption for 10 CFR 70.24. As required by 10 CFR 50.68(b)(8) the NMC is proposing the following commitment:

> Following approval of this amendment the PBNP Unit 1 and 2, Final Safety Analysis Report will be revised no later than the next update required under 10 CFR 50.71(e) to reflect the adoption of 10 CFR 50.68(b).

The following is a list of commitments that will no longer be effective based on the new criticality analysis:

1. We will replace the REI-25 surveillance with non-destructive examination for the presence of gaps of ten representative full-length Boraflex panels using neutron attenuation measurements. The surveillance sample will include the four panels with accelerated exposures on the cells adjacent to the surveillance coupons and six others, selected at random, from those that have been exposed to the greatest number of freshly discharged fuel assemblies at the time of the surveillance. This surveillance will be repeated at five year intervals.

2. If this program is modified we will inform the NRC.

3. We will maintain a data base to track the position and movement of spent fuel assemblies in the spent fuel storage racks.

4. In-situ blackness testing of at least 10 full-length boraflex panels every five years will be continued under the current boraflex surveillance program. The next blackness test campaign is anticipated for the summer of 2001.

5. In the event that the Boraflex is determined to have deteriorated to a point beyond the bounds of the current spent fuel pool criticality analysis, we will administratively control placement of new fuel assemblies and those not meeting an established criterion to a designated area in the spent fuel pool in a checkerboard pattern. The current criterion for spent fuel is 38,400 MWD/MTU as stated in the Boraflex surveillance program safety evaluation.

6. Implement an enhanced Boraflex monitoring program prior to the period of extended operation.

7. Certain accelerated Boraflex panels will be areal density and blackness tested every two years during the period of extended operation.

   The first Boraflex areal density testing of the Boraflex panels will be performed prior to the period of extended operation.

   A new procedure to schedule and perform Boraflex areal density and blackness testing will be created.

   If silica sampling and trending indicates a boron areal density depletion trend to a value less than the acceptance criteria (i.e., maintaining the 5% subcriticality margin) prior to the next scheduled test, then an evaluation will be performed within the corrective action program and the frequency of blackness and areal density testing increased.

   Corrective actions will be taken to ensure that the 5% subcriticality margin of the spent fuel racks in the SFP is maintained during the period of extended operation. Corrective actions will be initiated if the test results find that the 5% subcriticality margin cannot be maintained because of current or projected future degradation. Corrective actions may include, but are not necessarily limited to, the following:

   * Reanalysis
   * Repair and/or Replacement

Other statements made in this submittal are provided for information purposes and are not considered to be commitments.

8.0 ENVIRONMENTAL EVALUATION

NMC has evaluated the proposed changes and determined that (i) the proposed amendment involves no significant hazards considerations, (ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) there is no significant increase in the individual or cumulative occupational exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed changes is not required.

9.0 REFERENCES

1. WCAP-14416-P-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," dated June 1995.

2. WCAP-16541-P, "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis," Revision 0, February 2006.

3. Letter from L. L. Gundrum (NRC) to R. R. Grigg (WEP), "Point Beach Nuclear Plant, Unit Nos. 1 and 2 - Issuance of Exemption from the Requirements of 10 CFR 70.24, (TAC Nos. M98973 and M98974)," dated October 6, 1997.

4. NRC Generic Letter 96-04: "Boraflex Degradation in Spent Fuel Pool Storage Racks," dated June 26, 1996.

5. Point Beach Nuclear Plant Unit 2, LER 2006-02-00, "Fuel Assemblies in Spent Fuel Pool do not Meet Technical Specification Requirements," dated August 25, 2006.

6. Westinghouse NSAL-99-003, Credit for Integral Fuel Burnable Absorbers in Spent Fuel Pool Criticality Analysis, dated February 26, 1999.

7. 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.

8. NRC NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3, June 2004.

9. Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)," dated October 25, 1996.

10. 10 CFR 50, Appendix A, General Design Criterion (GDC) 62, "Prevention of Criticality in Fuel Storage and Handling."

11. ANS 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," dated October 7, 1983.

12. NRC Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978.

13. Westinghouse NSAL-00-015. "Axial Burnup Shape Reactivity Bias," November 2, 2000.

14. Letter from S. Dembek (NRC) to H.A. Sepp (Westinghouse), "Non-conservatisms in Axial Burnup Biases for Spent Fuel Rack Criticality Analysis Methodology," dated July 27, 2001 (ML012080337).

15. "SCALE 4.4a Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC Code Package CCC-545, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2000.

16. NRC Information Notice 91-26, "Potential Nonconservative Errors in the Working Format Hansen-Roach Cross-Section Set Provided with the KENO and SCALE Codes," dated April 2, 1991.

17. NRC Information Notice 2005-13, "Potential Non-Conservative Error in Modeling Geometric Regions in the KENO-V.A Criticality Code," dated May 17, 2005.

18. NRC Information Notice 2005-31, "Potential Non-Conservative Error in Preparing Problem-Dependent Cross Sections for use with the KENO V.A or KENO-V.I Criticality Code," dated November 17, 2005.

19. M. N. Baldwin, et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel; Summary Report," BAW-1484-7, July 1979.

20. "DIT: Discrete Integral Transport Assembly Design Code," CE-CES-11, Revision 4-P, April 1994.

21. NRC Information Notice 92-21, "Spent Fuel Pool Reactivity Calculations," dated March 24, 1992.

22. NRC Information Notice 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks," dated September 8, 1995.

23. NRC Regulatory Issue Summary 2001-12, "Nonconservatism in Pressurized Water Reactor Spent Fuel Storage Pool Reactivity Equivalencing Calculations," dated May 18, 2001.

24. Letter from G. S. Vissing (NRC) to R. C. Mecredy (RGE), "R. E. Ginna Nuclear Power Plant – Amendment RE: Revision to the Storage Configuration Requirements within the Existing Storage Racks and Taking Credit for a Limited Amount of Soluble Boron (TAC No. MA8443)," dated December 7, 2000. (ML003761578)

25. Letter from B. Benney (NRC) to G. M. Rueger (PG&E), "Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis (TAC Nos. MB2982 and MB2984)," dated September 25, 2002. (ML022610080)

26. Letter from L. N. Olshan (NRC) to W. R. McCollum (Duke), "Oconee Nuclear Station, Units 1, 2 and 3, Re: Issuance of Amendments (TAC Nos. MB0894, MB0895 and MB0896)," dated April 22, 2002. (ML020930470)

27. Letter from C. P. Patel (NRC) to C. J. Gannon (CP&L) "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Regarding Soluble Boron Credit for Fuel Storage Pools (TAC No. MC8267)," dated March 10, 2006. (ML060600349)

28. Safety Evaluation Report, "Related to the License Renewal of the Point Beach Nuclear Plant Units 1 and 2," Docket Nos. 50-266 and 5-301, October 2005.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

LICENSE AMENDMENT REQUEST 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

(14 pages follow)

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Storage

LCO 3.7.12 ~~Fuel assembly storage in the spent fuel pool shall be as follows:~~

~~a. Fuel assembly initial enrichment ≤ 4.6 % w/o U-235; or~~

~~b. Fuel assembly contains Integral Fuel Burnable Absorber (IFBA) rods within the "acceptable" range of Figure 3.7.12-1.~~

<u>The combination of initial enrichment, burnup and decay time of each fuel assembly stored in the spent fuel pool shall be within the Acceptable range of Figure 3.7.12-1 or in accordance with Specification 4.3.1.1.</u>

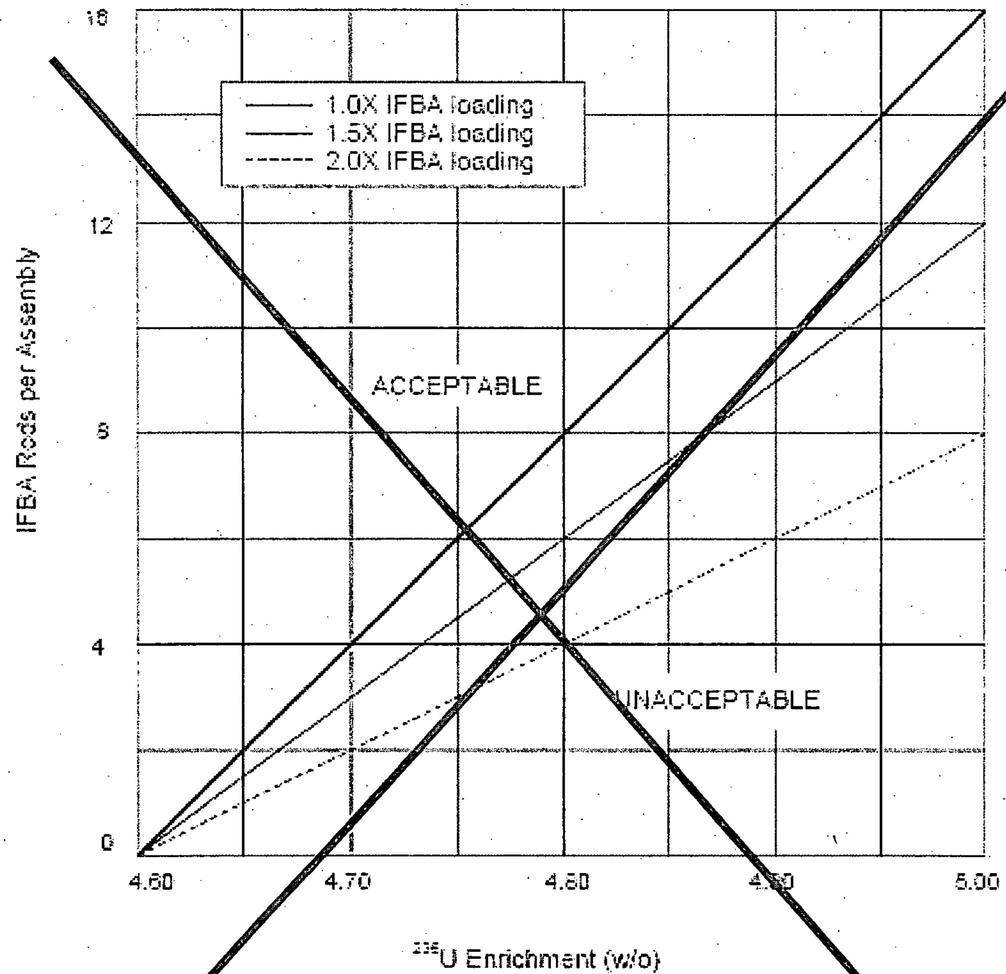
APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	-------------------NOTE---------------------- LCO 3.0.3 is not applicable. --- A.1 Restore the spent fuel pool within fuel storage limits.	 Immediately

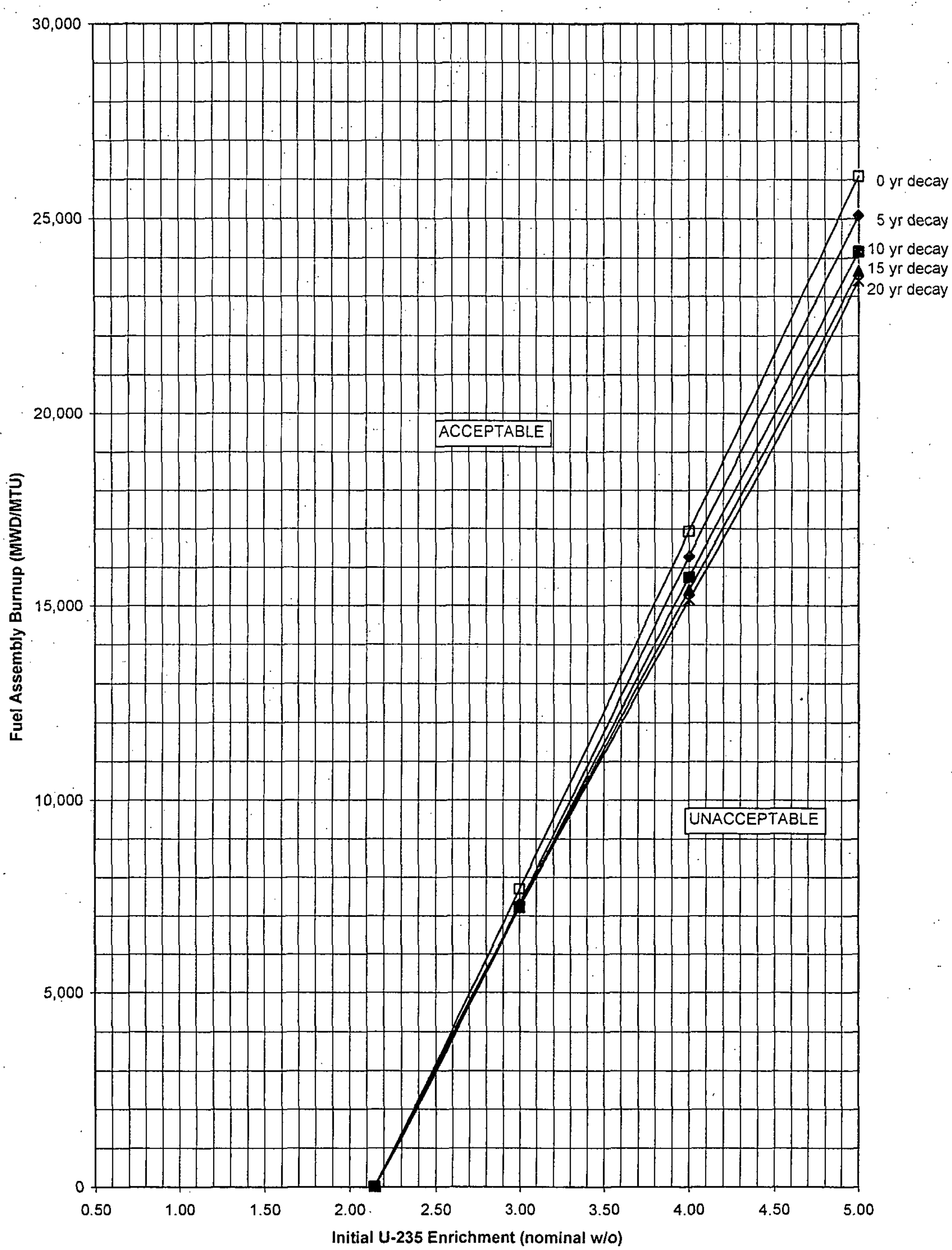
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify by administrative means each fuel assembly meets fuel storage limits.	Prior to storing the fuel assemblies in the spent fuel storage pool



Note: 1.0X, 1.5X, and 2.0X IFBA rods have normal poison material loadings of 1.67, 2.50, and 3.34 milligrams B-10 per inch, respectively.

Figure 3.7.12-1 (page 1 of 1)
Fuel Assembly IFBA Requirements



0 yr decay = $-49.58e^3 + 561.12e^2 + 7134.44e - 17417.89$
5 yr decay = $-108.67e^3 + 1212.54e^2 + 4510.52e - 14201.98$
10 yr decay = $-24.62e^3 + 247.20e^2 + 7696.65e - 17424.68$
15 yr decay = $68.81e^3 - 786.56e^2 + 11140.74e - 20978.70$
20 yr decay = $163.70e^3 - 1797.81e^2 + 14448.84e - 24359.00$

Figure 3.7.12-1
Fuel Assembly Burnup Requirement of "All-Cell" Storage Configuration

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

~~a. Fuel assemblies meeting at least one of the following storage limits may be stored in the spent fuel storage racks:~~

~~1. Fuel assemblies with an enrichment ≤4.6% weight percent U-235; or~~

~~2. Fuel assemblies which contains Integral Fuel Burnable Absorber (IFBA) pins in the "acceptable range" of Figure 3.7.12-1.~~

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

b. keff ~~≤ 0.95~~ < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~Section 9.4 of the FSAR~~ Reference 1;

c. k_{eff} ≤ 0.95 if fully flooded with water borated to 805 ppm, which includes an allowance for uncertainties as described in Reference 1;

~~c.~~ d. A nominal 9.825 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "Acceptable" range of Figure 3.7.12-1 may be allowed unrestricted storage in the fuel storage racks; and

f. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "Unacceptable" range of Figure 3.7.12-1 will be stored in compliance with Figures 4.3.1-1 through 4.3.1-8.

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

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

4.0 DESIGN FEATURES

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR;

c. $k_{eff} \leq 0.98$ under optimum moderator density conditions, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR; and

d. A nominal 20 inch center to center distance between fuel assemblies placed in the storage racks.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 40 ft 8 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1502 fuel assemblies.

REFERENCE 1. "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis," WCAP-16541-P, Westinghouse Electric Company, February 2006.

4.0 DESIGN FEATURES

L1	H1
L1	L1

H1: Fresh fuel assembly with maximum 5.0 w/o U-235.

No restriction on burnup.

L1: Spent fuel assemblies in the “Acceptable” range of Figure 4.3.1-6.

Figure 4.3.1-1
1-Out-of-4 for 5 w/o with no IFBA Storage Configuration

4.0 DESIGN FEATURES

L2	H2
L2	L2

H2: Fresh fuel assembly with maximum 4.0 w/o U-235 with no IFBA or maximum 5.0 w/o U-235 with IFBA in the "Acceptable" range of Figure 4.3.1-8.

No restriction on burnup.

L2: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-7.

Figure 4.3.1-2
1-Out-of-4 for 4 w/o with IFBA Storage Configuration

4.0 DESIGN FEATURES

1-Out-of-4 for 5 w/o Fresh

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
L1	L1	L1	L1	A	A	A
H1	L1	H1	L1	A	A	A
L1	L1	L1	L1	A	A	A
H1	L1	H1	L1	A	A	A

All-Cell

A: Fuel assembly in "Acceptable" range of Figure 3.7.12-1.

H1: Fresh fuel assembly with maximum 5.0 w/o U-235.
No restriction on burnup.

L1: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-6.

Figure 4.3.1-3
1-Out-of-4 for 5 w/o with no IFBA / "All Cell" Interface

4.0 DESIGN FEATURES

1-Out-of-4 for 4 w/o Fresh with IFBA

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
L2	L2	L2	L2	A	A	A
H2	L2	H2	L2	A	A	A
L2	L2	L2	L2	A	A	A
H2	L2	H2	L2	A	A	A

All-Cell

A: Fuel assembly in "Acceptable" range of Figure 3.7.12-1.

H2: Fresh fuel assembly with maximum 4.0 w/o U-235 with no IFBA or maximum 5.0 w/o U-235 with IFBA in the "Acceptable" range of Figure 4.3.1-8.

No restriction on burnup.

L2: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-7.

Figure 4.3.1-4
1-Out-of-4 for 4 w/o with IFBA / "All Cell" Interface

4.0 DESIGN FEATURES

1-Out-of-4 for 4 w/o Fresh with IFBA

1-Out-of-4 for 5 w/o Fresh

L1	L1	L1	L1	L1	L1	L1
L1	H1	L1	H1	L1	H1	L1
L1	L1	L1	L1	L1	L1	L1
L2	L2	L2	L2	L1	H1	L1
H2	L2	H2	L2	L1	L1	L1
L2	L2	L2	L2	L1	H1	L1
H2	L2	H2	L2	L1	L1	L1

H1: Fresh fuel assembly with maximum 5.0 w/o U-235.

No restriction on burnup.

L1: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-6.

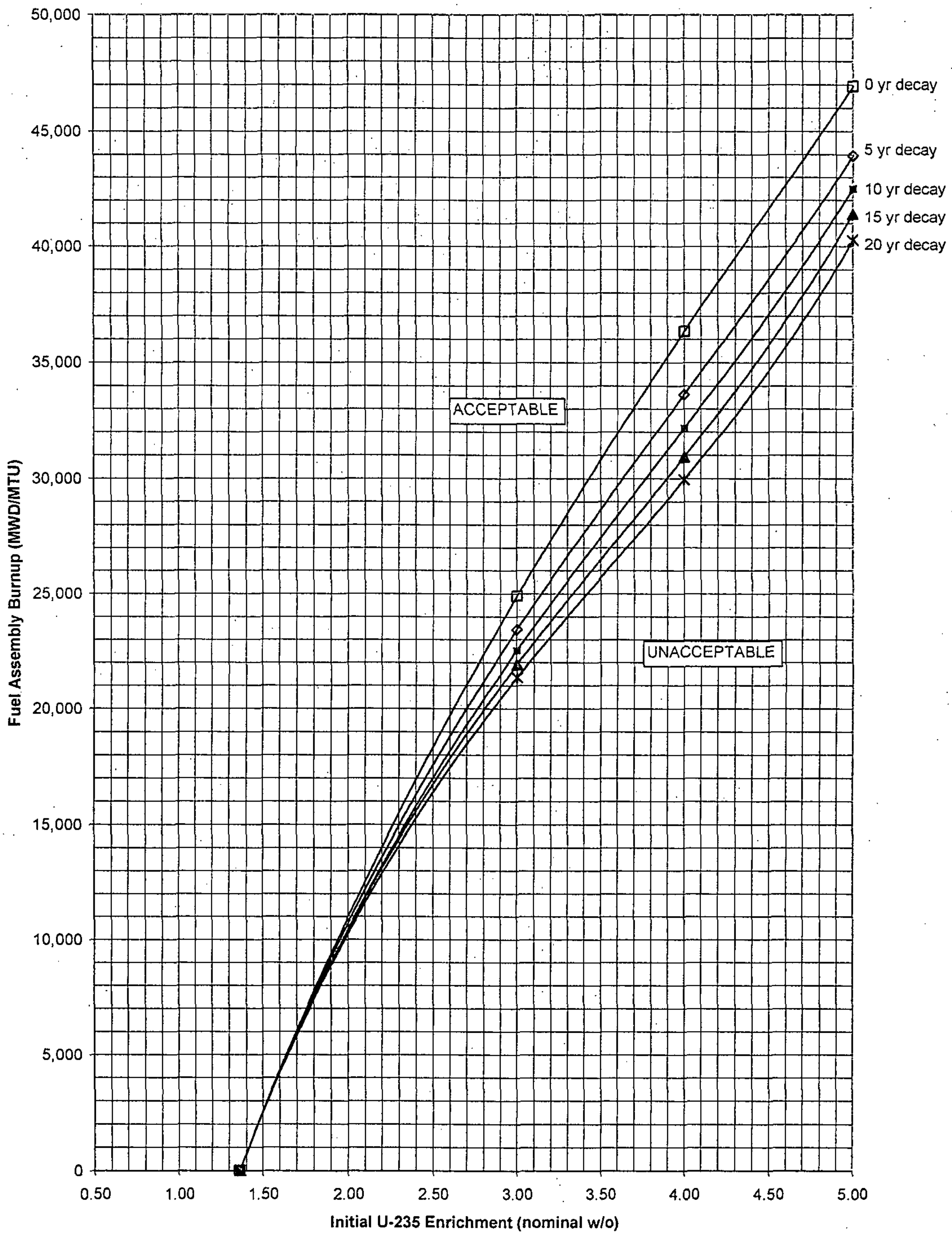
H2: Fresh fuel assembly with maximum 4.0 w/o U-235 with no IFBA or maximum 5.0 w/o U-235 with IFBA in the "Acceptable" range of Figure 4.3.1-8.

No restriction on burnup.

L2: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-7.

Figure 4.3.1-5
1-Out-of-4 for 4 w/o with IFBA / 1-Out-of-4 for 5 w/o with no IFBA

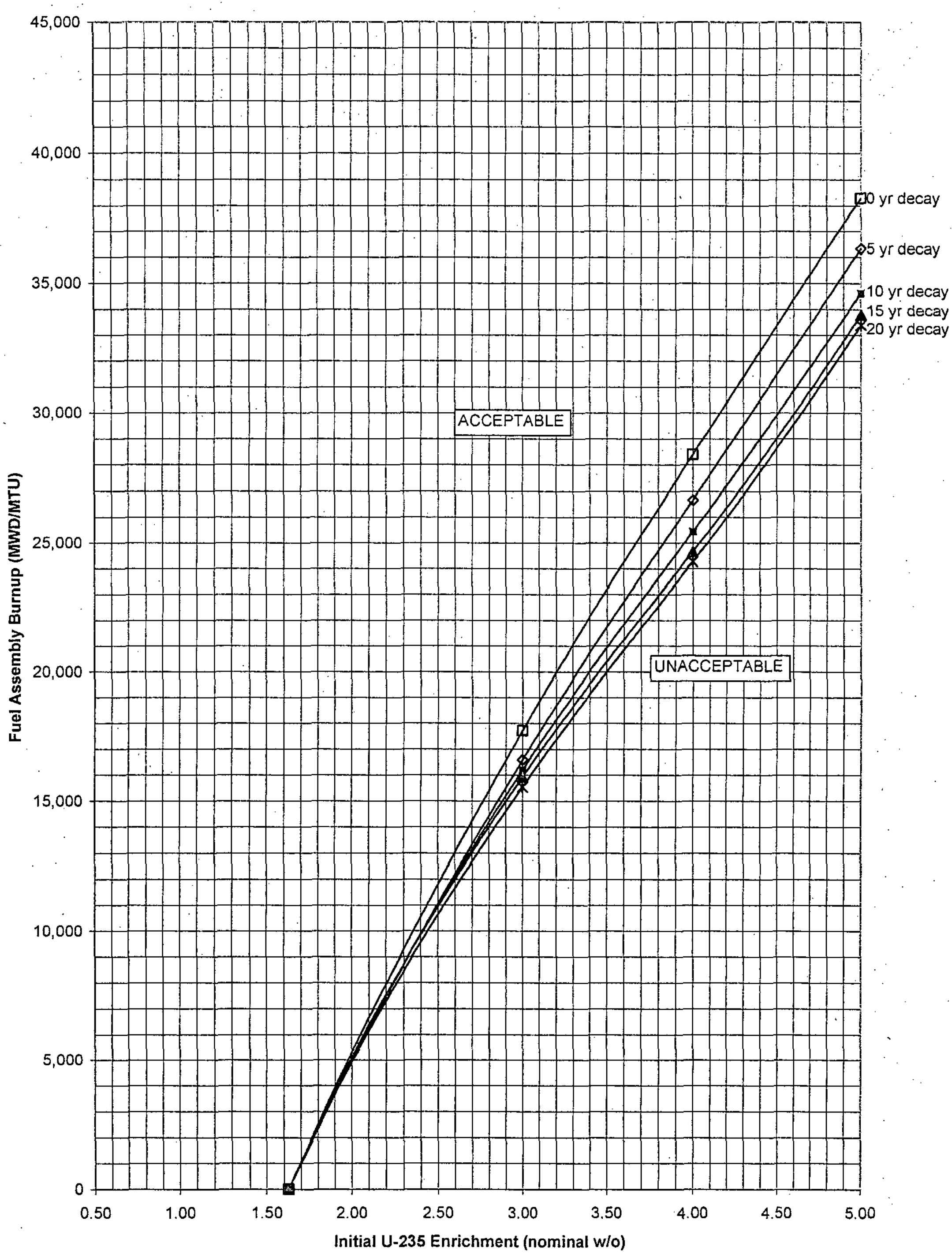
4.0 DESIGN FEATURES



0 yr decay = 276.83e³ - 3740.62e² + 27373.43e - 31037.19
5 yr decay = 452.23e³ - 5348.68e² + 30881.71e - 33275.57
10 yr decay= 524.70e³ - 5948.94e² + 31863.58e - 33682.58
15 yr decay= 651.51e³ - 7101.87e² + 34622.21e - 35621.37
20 yr decay= 692.01e³ - 7461.55e² + 35236.72e - 35893.48

Figure 4.3.1-6
Spent Fuel Assembly Burnup Requirements for 1-Out-of-4 for 5.0 w/o with no IFBA

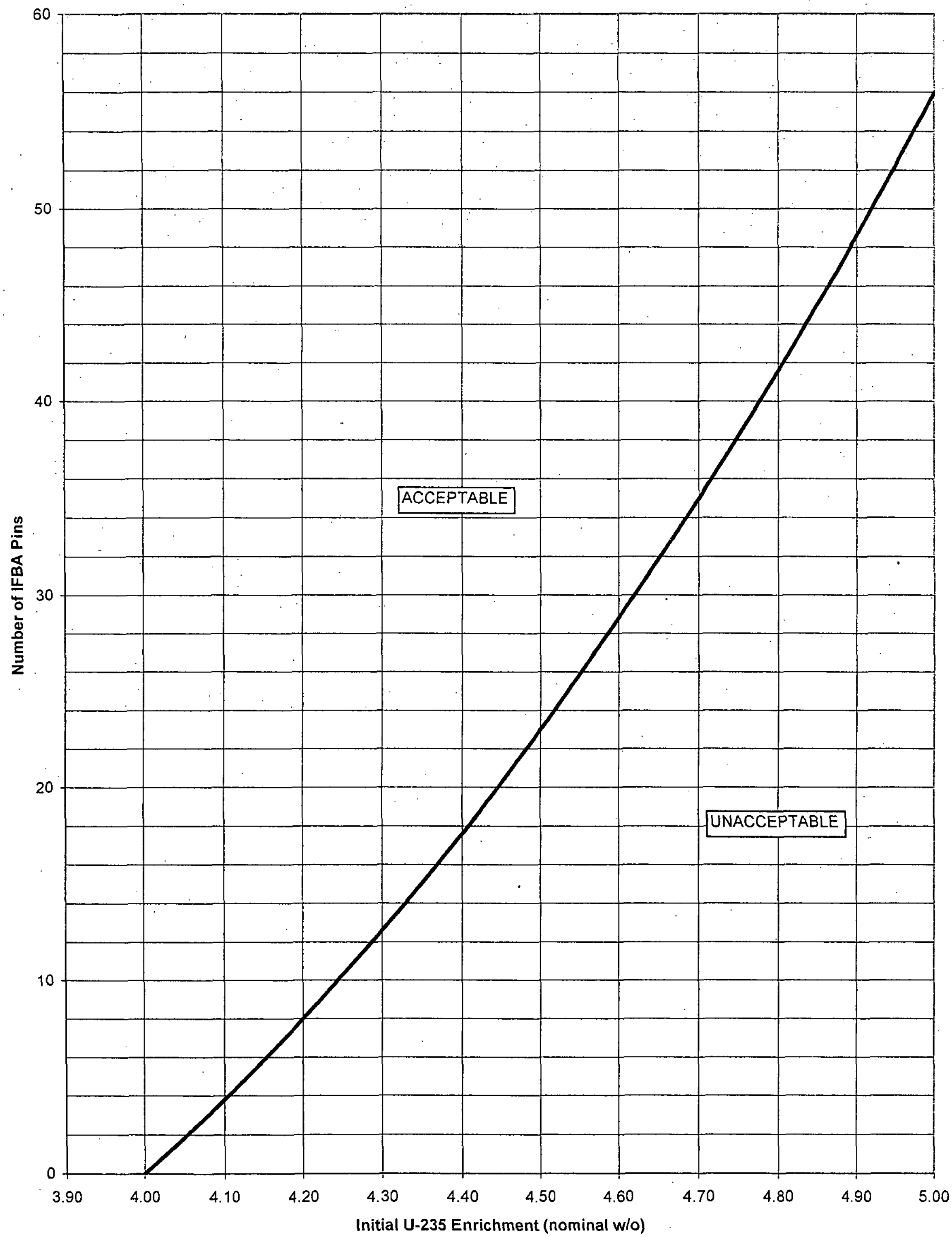
4.0 DESIGN FEATURES



0 yr decay = 170.36e³ - 2420.63e² + 21300.07e - 28991.46
5 yr decay = 195.63e³ - 2541.74e² + 20617.59e - 27668.70
10 yr decay= 318.85e³ - 3841.38e² + 24315.99e - 30776.68
15 yr decay= 423.62e³ - 4874.30e² + 27165.59e - 33130.16
20 yr decay= 389.42e³ - 4464.01e² + 25551.28e - 31442.11

Figure 4.3.1-7
Spent Fuel Assembly Burnup Requirements for 1-Out-of-4 for 4.0 w/o with IFBA

4.0 DESIGN FEATURES



Number of IFBA Pins = $20.0e^2 - 124.0e + 176.0$

Figure 4.3.1-8
Fresh Fuel IFBA Requirements

ENCLOSURE 3

REVISED TECHNICAL SPECIFICATION CHANGES

LICENSE AMENDMENT REQUEST 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

(12 pages follow)

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Storage

LCO 3.7.12 The combination of initial enrichment, burnup and decay time of each fuel assembly stored in the spent fuel pool shall be within the Acceptable range of Figure 3.7.12-1 or in accordance with Specification 4.3.1.1.

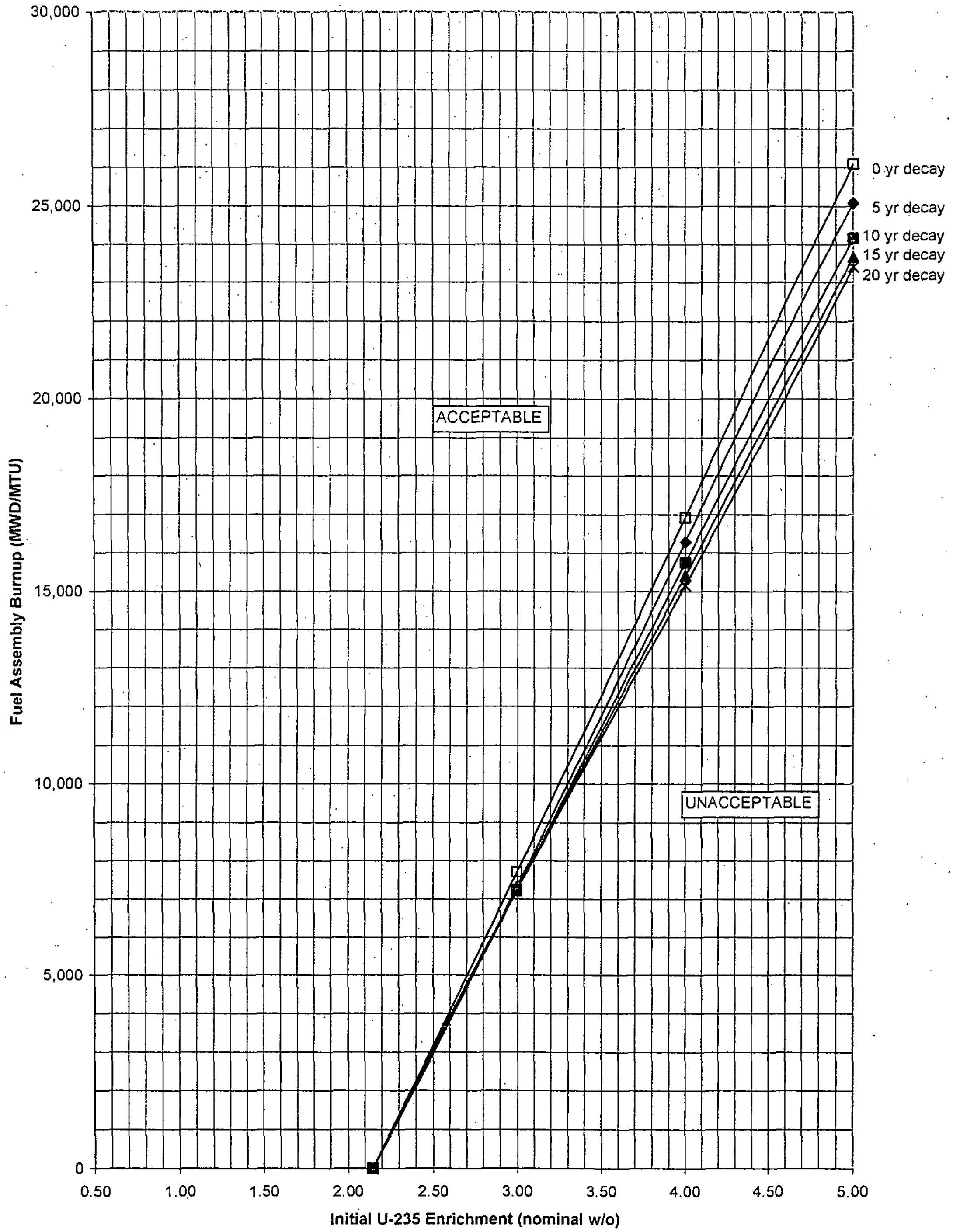
APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	------------------NOTE------------------ LCO 3.0.3 is not applicable. ---	
	A.1 Restore the spent fuel pool within fuel storage limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify by administrative means each fuel assembly meets fuel storage limits.	Prior to storing the fuel assemblies in the spent fuel storage pool



0 yr decay = $-49.58e^3 + 561.12e^2 + 7134.44e - 17417.89$
5 yr decay = $-108.67e^3 + 1212.54e^2 + 4510.52e - 14201.98$
10 yr decay = $-24.62e^3 + 247.20e^2 + 7696.65e - 17424.68$
15 yr decay = $68.81e^3 - 786.56e^2 + 11140.74e - 20978.70$
20 yr decay = $163.70e^3 - 1797.81e^2 + 14448.84e - 24359.00$

Figure 3.7.12-1
Fuel Assembly Burnup Requirement of "All-Cell" Storage Configuration

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum 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 Reference 1;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 805 ppm, which includes an allowance for uncertainties as described in Reference 1;

d. A nominal 9.825 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "Acceptable" range of Figure 3.7.12-1 may be allowed unrestricted storage in the fuel storage racks; and

f. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "Unacceptable" range of Figure 3.7.12-1 will be stored in compliance with Figures 4.3.1-1 through 4.3.1-8.

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

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

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR;

c. $k_{eff} \leq 0.98$ under optimum moderator density conditions, which includes an allowance for uncertainties as described in Section 9.4 of the FSAR; and

d. A nominal 20 inch center to center distance between fuel assemblies placed in the storage racks.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 40 ft 8 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1502 fuel assemblies.

REFERENCE 1. "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis," WCAP-16541-P, Westinghouse Electric Company, February 2006.

4.0 DESIGN FEATURES

L1	H1
L1	L1

H1: Fresh fuel assembly with maximum 5.0 w/o U-235.

No restriction on burnup.

L1: Spent fuel assemblies in the “Acceptable” range of Figure 4.3.1-6.

Figure 4.3.1-1
1-Out-of-4 for 5 w/o with no IFBA Storage Configuration

4.0 DESIGN FEATURES

L2	H2
L2	L2

H2: Fresh fuel assembly with maximum 4.0 w/o U-235 with no IFBA or maximum 5.0 w/o U-235 with IFBA in the "Acceptable" range of Figure 4.3.1-8.

No restriction on burnup.

L2: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-7.

Figure 4.3.1-2
1-Out-of-4 for 4 w/o with IFBA Storage Configuration

4.0 DESIGN FEATURES

1-Out-of-4 5 w/o Fresh

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
L1	L1	L1	L1	A	A	A
H1	L1	H1	L1	A	A	A
L1	L1	L1	L1	A	A	A
H1	L1	H1	L1	A	A	A

All-Cell

A: Fuel assembly in "Acceptable" range of Figure 3.7.12-1.

H1: Fresh fuel assembly with maximum 5.0 w/o U-235.

No restriction on burnup.

L1: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-6.

Figure 4.3.1-3
1-Out-of-4 for 5 w/o with no IFBA / "All Cell" Interface

4.0 DESIGN FEATURES

1-Out-of-4 4 w/o Fresh with IFBA

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
L2	L2	L2	L2	A	A	A
H2	L2	H2	L2	A	A	A
L2	L2	L2	L2	A	A	A
H2	L2	H2	L2	A	A	A

All-Cell

A: Fuel assembly in "Acceptable" range of Figure 3.7.12-1.

H2: Fresh fuel assembly with maximum 4.0 w/o U-235 with no IFBA or maximum 5.0 w/o U-235 with IFBA in the "Acceptable" range of Figure 4.3.1-8.

No restriction on burnup.

L2: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-7.

Figure 4.3.1-4
1-Out-of-4 for 4 w/o with IFBA / "All Cell" Interface

4.0 DESIGN FEATURES

1-Out-of-4 4 w/o Fresh with IFBA

1-Out-of-4 5 w/o Fresh

L1	L1	L1	L1	L1	L1	L1
L1	H1	L1	H1	L1	H1	L1
L1	L1	L1	L1	L1	L1	L1
L2	L2	L2	L2	L1	H1	L1
H2	L2	H2	L2	L1	L1	L1
L2	L2	L2	L2	L1	H1	L1
H2	L2	H2	L2	L1	L1	L1

H1: Fresh fuel assembly with maximum 5.0 w/o U-235.

No restriction on burnup.

L1: Spent fuel assemblies in the “Acceptable” range of Figure 4.3.1-6.

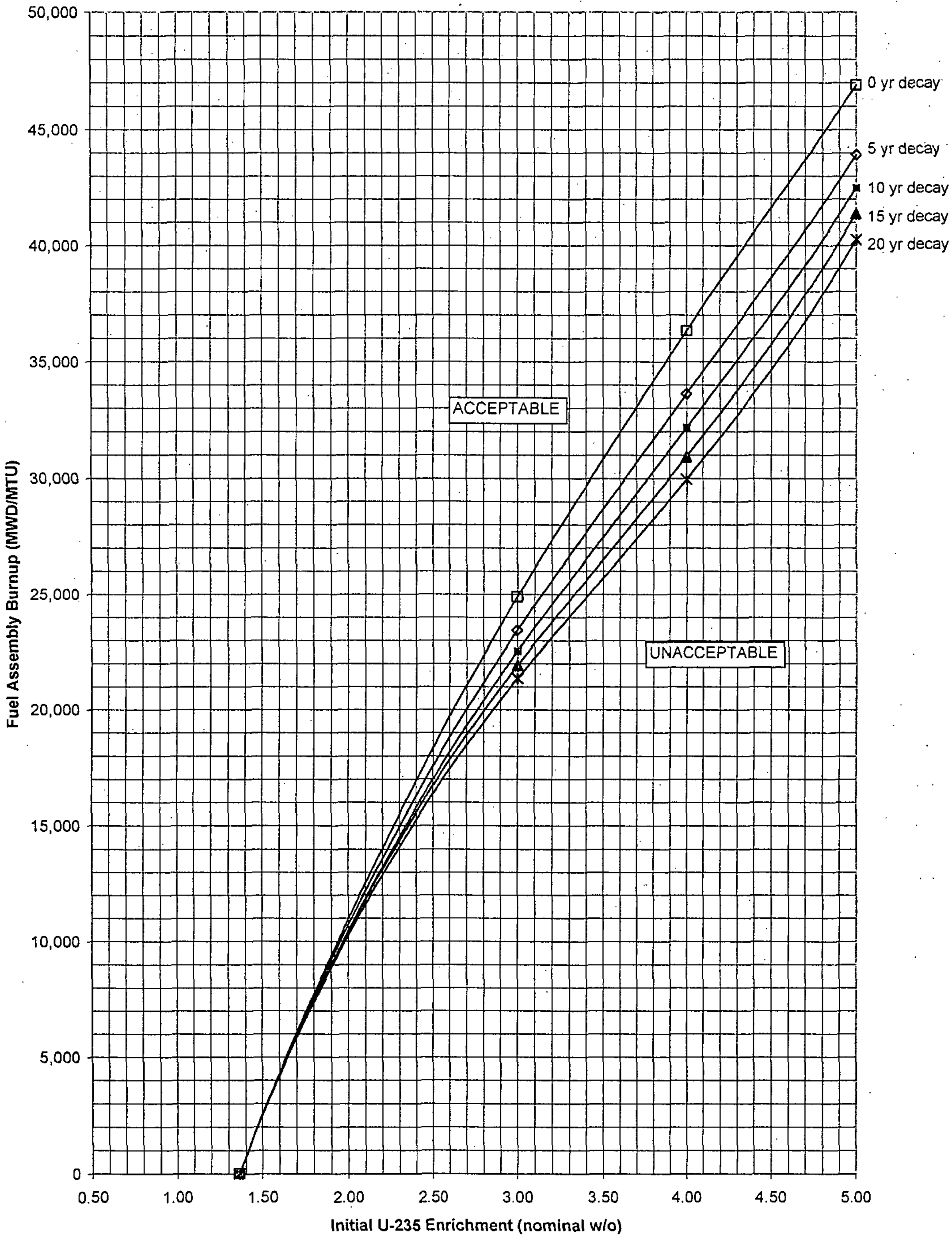
H2: Fresh fuel assembly with maximum 4.0 w/o U-235 with no IFBA or maximum 5.0 w/o U-235 with IFBA in the “Acceptable” range of Figure 4.3.1-8.

No restriction on burnup.

L2: Spent fuel assemblies in the “Acceptable” range of Figure 4.3.1-7.

Figure 4.3.1-5
1-Out-of-4 for 4 w/o with IFBA / 1-Out-of-4 for 5 w/o with no IFBA

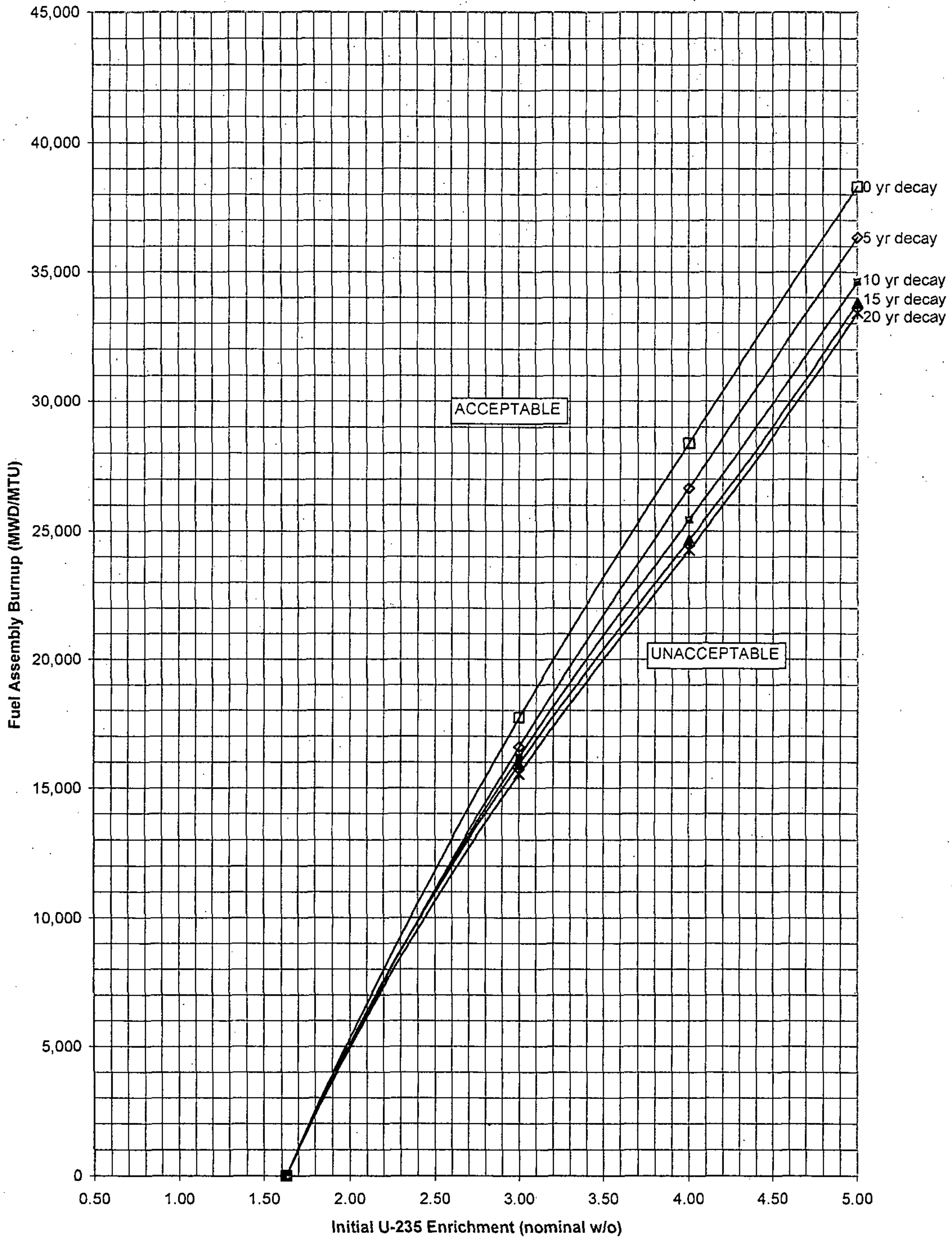
4.0 DESIGN FEATURES



0 yr decay = $276.83e^3 - 3740.62e^2 + 27373.43e - 31037.19$
5 yr decay = $452.23e^3 - 5348.68e^2 + 30881.71e - 33275.57$
10 yr decay= $524.70e^3 - 5948.94e^2 + 31863.58e - 33682.58$
15 yr decay= $651.51e^3 - 7101.87e^2 + 34622.21e - 35621.37$
20 yr decay= $692.01e^3 - 7461.55e^2 + 35236.72e - 35893.48$

Figure 4.3.1-6
Spent Fuel Assembly Burnup Requirements for 1-Out-of-4 for 5.0 w/o with no IFBA

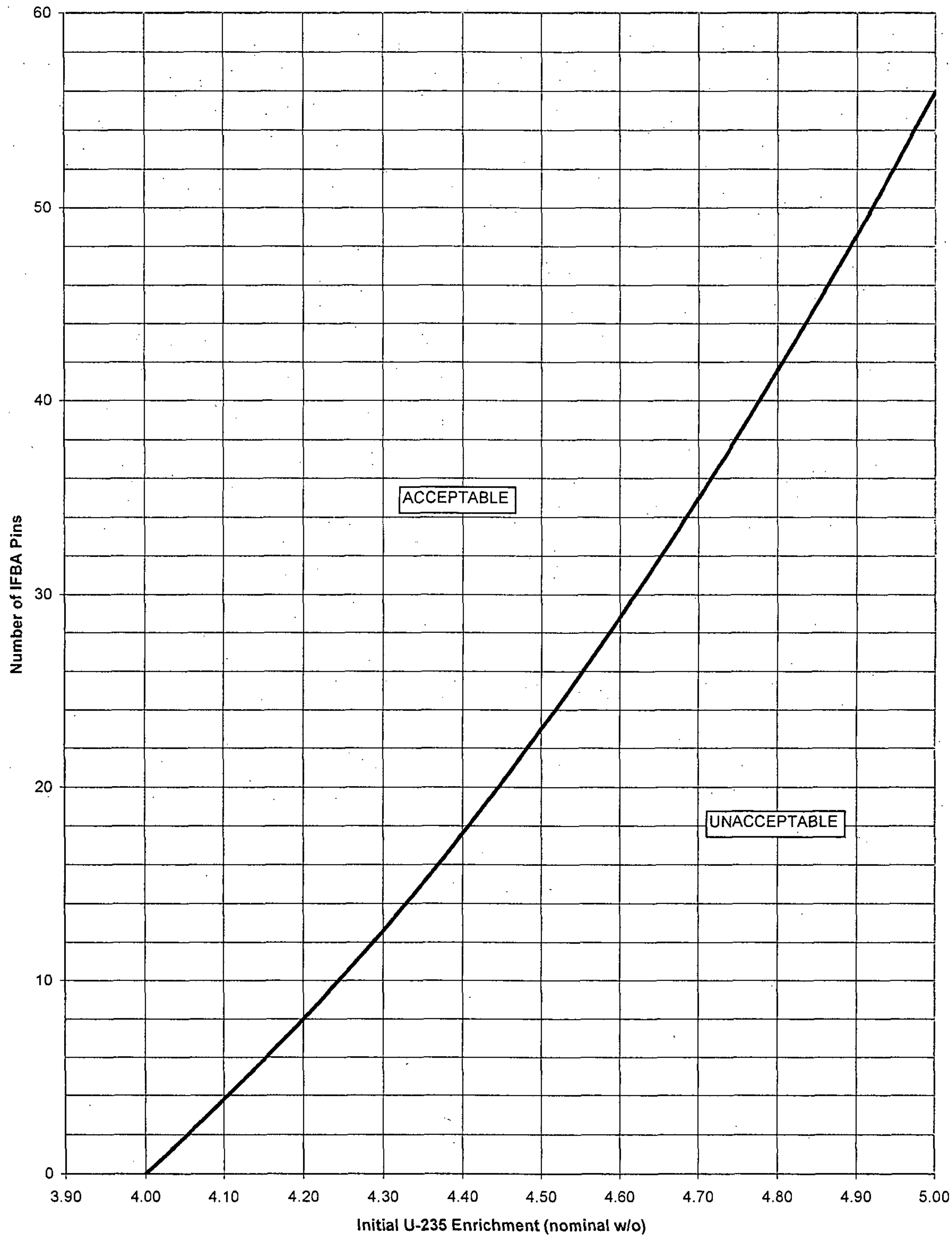
4.0 DESIGN FEATURES



0 yr decay = $170.36e^3 - 2420.63e^2 + 21300.07e - 28991.46$
5 yr decay = $195.63e^3 - 2541.74e^2 + 20617.59e - 27668.70$
10 yr decay= $318.85e^3 - 3841.38e^2 + 24315.99e - 30776.68$
15 yr decay= $423.62e^3 - 4874.30e^2 + 27165.59e - 33130.16$
20 yr decay= $389.42e^3 - 4464.01e^2 + 25551.28e - 31442.11$

Figure 4.3.1-7
Spent Fuel Assembly Burnup Requirements for 1-Out-of-4 for 4.0 w/o with IFBA

4.0 DESIGN FEATURES



Number of IFBA Pins = $20.0e^2 - 124.0e + 176.0$

Figure 4.3.1-8
Fresh Fuel IFBA Requirements

ENCLOSURE 4

DRAFT PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

(provided for information)

LICENSE AMENDMENT REQUEST 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

(6 pages follow)

B 3.7 PLANT SYSTEMS

B 3.7.11 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	In the spent fuel storage rack design, the spent fuel pool is considered a single region. The spent fuel storage pool will accommodate 1502 fuel assemblies with a maximum enrichment of 5.0 wt% U-235. The racks may contain fresh or spent fuel within the acceptable domain according to Figure 3.7.12-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure 3.7.12-1 shall be stored in accordance with paragraph 4.3.1.1 in section 4.3, Fuel Storage. The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under normal conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting keff of less than 1.0 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel storage racks is based on the use of unborated water, which maintains the spent fuel pool in a subcritical condition during normal operation with the pool full loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of a fresh fuel assembly in a spent fuel assembly location for the "All-Cell" configuration. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel rack with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.12, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.
APPLICABLE SAFETY ANALYSES	Most accident conditions do not result in an reactivity increase for the fuel stored in the spent fuel pool. Examples of these accident conditions are loss of cooling (reactivity increase with decreasing water density) and dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevent criticality. For these events, the spent fuel pool keff storage limit of 0.95 is maintained by maintaining a minimum boron concentration of 805 ppm (Ref. 2). Simultaneous occurrence of these events is not postulated. The double

BASES

APPLICABLE SAFETY ANALYSES (continued)	contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. The accident analyses are provided in the FSAR, Section 14.2.1 (Ref. 4). The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.
LCO	The fuel storage pool boron concentration is required to be ≥ 2100 ppm. The specified concentration of dissolved boron provides significant margin to the boron concentration used in the analyses of the potential critical accident scenarios as described in Reference 4. This concentration is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool and encompasses movement of fuel assemblies in the spent fuel storage pool. This LCO provides assurance that keff of the spent fuel storage pool will remain less than or equal to 0.95, even under postulated accident conditions.
ACTIONS	The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies or restoration of boron concentration is not sufficient reason to require a reactor shutdown. <u>A.1</u> When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to suspending the movement of fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. By suspending movement of fuel, inadvertent placement of a fuel assembly between a fuel storage rack module and the wall of the spent fuel pool is precluded.

BASES

ACTIONS (continued)

A.2

Immediate action must be taken to restore boron concentration in the fuel storage pool to ≥2100 ppm to assure protection from excessive fuel pool cooldown reactivity insertion events. Restoration of boron concentration could take several hours or days depending on the magnitude of change required, which may involve feed and bleed operations. Immediate initiation of action is warranted based on the importance of maintaining keff of the spent fuel pool ≤0.95. As stated in Reference 2, 805 ppm is adequate to prevent the spent fuel pool keff storage limit of 0.95 from being exceeded as a result of the most limiting accident. Accordingly, for minor deviations, significant margin exists to the analysis limit.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. FSAR. Section 9.4.

2. "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-16541-P, Westinghouse Electric Company, February, 2006.

3. 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).

4. FSAR. Section 14.2.1.

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool Storage

BASES

BACKGROUND

In the spent fuel storage rack design, the spent fuel pool is considered a single region. The spent fuel storage pool will accommodate 1502 fuel assemblies with a maximum enrichment of 5.0 wt% U-235. The racks may contain fresh or spent fuel within the acceptable domain according to Figure 3.7.12-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Figure 3.7.12-1 shall be stored in accordance with paragraph 4.3.1.1 in section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting keff of <1.0 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel storage racks is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with misloading a fresh fuel assembly in place of a spent fuel assembly in the "All-Cell" storage configuration. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the spent fuel storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (ref. 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.11, "Fuel Storage Pool Boron Concentration") prevents criticality in the spent fuel pool. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10CFR 50.36(c)(2)(ii).

BASES

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.7.12-1, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain < 1.0, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure 3.7.12-1 shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

A.1

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

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.12-1, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.12-1 or Specification 4.3.1.1.

If unable to move fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

1. This SR verifies by administrative means, that the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.12-1 in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.12-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. FSAR. Section, 9.4.
2. 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).
3. "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-16541-P, Westinghouse Electric Company, February, 2006.
4. NRC Safety Evaluation Report, USNRC to WEPCO, dated February 23, 1990.

ENCLOSURE 5

EC 9694

BORON DILUTION ANALYSIS

LICENSE AMENDMENT REQUEST 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

(53 pages follow)

Engineering Change	
EC Number : 0000009694 000	Print Date: 12/11/2006
Status/Date : CLOSED 12/11/2006	PASSPORT
Facility : PB	
Type/Sub-type: EVAL	Page: 1

EC Title: BORON DILUTION ANALYSIS TO SUPPORT LICENSE AMENDMENT REQUEST

Mod Nbr : KW1: KW2: KW3: KW4: KW5:

Master EC :	N	Work Group :		Temporary :	N
Outage :	N	Alert Group:	E-NS/RE SE	Aprd Reqd Date:	12/08/2006
WO Required :	N	Image Addr :		Exp Insvc Date:	
Adv Wk Appvd:		Alt Ref. :		Expires On :	
Auto-Advance:	N	Priority :	C	Auto-Asbuild :	N
Caveat Outst:		Department :		Discipline :	
Resp Engr :	MICHAEL	D BARTEL			
Location :					

Milestone	Date	PassPort	Name		Req By
APPROVED BY	12/07/2006	WE2052	HENNESSY	WILLIAM	APPROVED
CLOSE	12/07/2006	WE9443	BARTEL	MICHAEL	CLOSED
OTHER1	12/07/2006	PB3114	WOOD	RICHARD	
I completed a review of EC 9694 and verified all references to the EC are correct.					
PREPARED (EVL)	12/06/2006	WE9443	BARTEL	MICHAEL	H/APPR
Updated header reference on attached PDF dilution analysis with the current EC number. This was not corrected when EC 9643 was created. EVAL ECs cannot be revised but only superseded. EC 9694 was created to superseded EC 9643 and 8652. The EC header reference was corrected.					
TECHNICAL RVW	12/07/2006	WE6065	VANCE	KEVIN	

Units

Fac	Unit	Description
PB	0	COMMON

Systems

Fac	System	Description
PB	FH	Fuel Handling

Reference Documents List

Facility	Type	SubType	Document	Sheet
PB	EC		0000008652	
Title: BORON DILUTION ANALYSIS TO SUPPORT LICENSE AMENDMENT REQUEST				
PB	EC		0000009694	
Title: BORON DILUTION ANALYSIS TO SUPPORT LICENSE AMENDMENT REQUEST				

Engineering Change		
EC Number	:	0000009694 000
Status/Date	:	CLOSED 12/11/2006
Facility	:	PB
Type/Sub-type:		EVAL

Print Date: 12/11/2006

PASSPORT

Reference Documents List

Facility	Type	SubType	Document	Sheet
PB	EC		0000009643	

Title: BORON DILUTION ANALYSIS TO SUPPORT LICENSE AMENDMENT REQUEST

Engineering Change Comments

Comments Last Updated By: **WE9443** Last Updated Date: **12/06/2006**

This EC supersedes EC 8652 in its entirety.

This EC supersedes EC 9643 in its entrety. See Topic notes for more information.

Engineering Change

EC Number	:	0000009694 000	Print Date:	12/11/2006
Facility	:	PB	PASSPORT	
Type/Sub-type	:	EVAL	Page:	1

Topic	:	DESCRIPTION	**Last Updated By**	:	WE9443
From Panel	:	TIME100	**Last Updated Date**	:	12/06/2006
			Text Status	:	UNLOCKED

This engineering evaluation is performed to support a license amendment request for the spent fuel pool. The NRC has required that this type of analysis be performed for plants wishing to use a soluable boron credit in the criticality analysis for the spent fuel pool. For additional information see the file in sharepoint.

Engineering Change

EC Number : 0000009694 000
Facility : PB
Type/Sub-type: EVAL

Print Date: 12/11/2006

PASSPORT

Page: 1

Topic :	JUSTIFICATION	**Last Updated By** :	WE9443
From Panel :	TIME100	**Last Updated Date**:	12/06/2006
		Text Status :	UNLOCKED

This analysis does not support or change the design basis of any installed system, structure or component. It is used to support a license amendment request. A separate calculation EC is being performed to incorporate the resutls of the new criticality analysis into the design and licensing basis. This analysis only examines potential boron dilution sources that could occur as a result of normal plant operations through approved plant procedures. See the attached file in Sharepoint for additional information.

Engineering Change EC Number : 0000009694 000 Facility : PB Type/Sub-type: EVAL	Print Date: 12/11/2006 PASSPORT Page: 1

Topic : NOTES
From Panel : TIME100

Last Updated By : WE9443
Last Updated Date: 12/06/2006
Text Status : UNLOCKED

The conclusion of this analysis is documented in the attached Word document
in Sharepoint. The following is a synopsis of the evaluation:

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A dilution of the spent fuel pool boron concentration from 2100 ppm to 805 ppm is not a credible event. This was based on a determination that it would take 251,763 gallons of water to dilute the spent fuel pool from 2100 ppm to 805 ppm. There is no single water source on site that holds such a dilution volume. There is one source that could supply this amount during normal operation, that is the demineralized water system. However, sufficient controls are in place to prevent this source from causing a dilution event. There are also sufficient controls in place to detect a dilution event and end it.

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EC 9643 superscedes EC 8652. This was necessary to correct an editorial mistake in EC8652 on page 17. The referenced time of 231 minutes to fill the SFP to the high level alarm was not updated to 249 minutes as it was in the rest of the document.

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EC 9694 supersedes EC 9643 and 8652. This was necessary to update the header information in the attached evaluation.

Engineering Change

EC Number : 0000009694 000
Facility : PB
Type/Sub-type: EVAL

Print Date: 12/11/2006

PASSPORT

Page: 1

Topic : REVIEWER COMMENTS
From Panel : TIME100

Last Updated By :
Last Updated Date:
Text Status :

Engineering Change	
EC Number :	0000009694 000
Facility :	PB
Type/Sub-type:	EVAL

Print Date: 12/11/2006

PASSPORT

Attributes

Attribute Sub-category: EE

Attribute Name	Value	PassPort	Date
EVAL NO			12/11/2006
PORC DTE			12/11/2006
PRIORITY RANKING			12/11/2006
SCRN NO			12/11/2006

1. What is the need for performing this Engineering Evaluation?

   Reason: To identify potential events which could dilute the spent fuel pool soluble boron and quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

   Background: Point Beach is requesting a license amendment to support a new criticality analysis for the Spent Fuel Pool (SFP). Implementation of WCAP-16541-P, "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis" will result in a change to the technical specifications and uses a different methodology from that currently in use. WCAP-16541-P demonstrates that the SFP will remain subcritical with no soluble boron and no Boraflex. However, soluble boron is credited in the analysis with maintaining the subcritical margins required in the current regulations and for mitigating postulated accidents. As a result of using a soluble boron credit, the NRC requires that licensees identify potential events which could dilute the spent fuel pool soluble boron concentration and quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. This engineering evaluation will be used to document the required dilution analysis.

2. Evaluation scope:

   The purpose of this non-safety related evaluation is to review potential dilution sources for the spent fuel pool. The review examines existing dilution sources at Point Beach, the volumes and flow rates of these sources, and how these sources interface with the spent fuel pool. Only dilution scenarios that could occur while operating the plant per approved operating procedures are considered.

   This evaluation is comprised of a: (1) A report section suitable for inclusion in the license amendment request; (2) An appendix for the determination of required volumes to fill the spent fuel pool and transfer canal; (3) An appendix for the determination of the volume to dilute the spent fuel pool to its final analysis concentration; and (4) An appendix of the supporting references.

   From the determination of the volume of water required and available flow rates and source volumes, the evaluation will show that there exists adequate controls both physical and administrative so that the spent fuel pool boron concentration will not get to the minimum value determined in the new criticality analysis during a dilution event.

   The report will be used as part of the license submittal for the new criticality analysis.

3. Evaluation:

1.0 INTRODUCTION

CAA-96-146, "Criticality Analysis of the Point Beach Nuclear Plant Spent Fuel Storage Racks Considering Boraflex Gaps and Shrinkage with Credit for Integral Fuel Burnable Absorber" for the spent fuel pool demonstrates that k_{eff} will be less than 0.95 when filled with unborated water. This analysis also requires that all fuel is assumed to be fresh. WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" is a Nuclear Regulatory Commission (NRC) accepted methodology for Westinghouse Pressurized Water Reactors (PWRs.) WCAP-14416-P allows the licensee to demonstrate that the spent fuel pool will have k_{eff} less than 0.95 by crediting soluble boron and fuel assembly burnup. The new Point Beach criticality analysis, WCAP-16541-P, "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis" takes advantage of the soluble boron and burnup credits so that the Boraflex need no longer be credited. The NRC approved the Westinghouse methodology with the requirement that all licensees using this method include a boron dilution analysis. WCAP-14416-P states:

"All licensees proposing to use the new method described above for soluble boron credit should identify potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit (as defined in (1)a above) and should quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. The effects of incomplete boron mixing, such as boron stratification, should be considered. This analysis should be submitted for NRC review and should also be used to justify the surveillance interval used for verification of the technical specification minimum pool boron concentration."

This boron dilution analysis has been completed to support the license amendment request based on WCAP-16541-P. This dilution analysis includes an evaluation of the following:

- Dilution Sources
- Boration Sources
- Instrumentation
- Administrative Procedures
- Piping
- Boron Dilution Initiating Events
- Boron Dilution Volumes and Times

The boron dilution analysis has been completed to ensure that sufficient time is available to detect and mitigate the dilution before the spent fuel rack criticality analysis $k_{eff} < 0.95$ is exceeded.

2.0 SPENT FUEL POOL AND RELATED SYSTEM FEATURES

This section provides background information on the spent fuel pool and its related systems and features.

2.1 Spent Fuel Pool Structure

The design purpose of the spent fuel pool is to provide for the underwater storage of spent fuel assemblies, control rods and other inserts after their removal from the reactor. The applicable design criteria require that the spent fuel pool remain subcritical, provide for adequate decay heat removal, provide adequate radiation shielding and provide prevention against radioactive release. The water in the spent fuel pool is used to remove decay heat, provides shielding and reduces the amount of radioactive gasses released during a fuel handling accident. The fuel is maintained subcritical by fuel assembly storage cell spacing and a solid neutron absorber, Boraflex. The spent fuel pool is filled with borated water that is not credited for maintaining sub-criticality during normal operation but is credited for postulated accidents. Since boron is credited for postulated accidents in the spent fuel pool a minimum boron concentration is required in the technical specifications. Evaporation of spent fuel pool water requires periodic makeup to ensure minimum levels are maintained. Since boric acid is not lost during evaporation, an unborated water source may be used to refill the spent fuel pool.

The spent fuel pool is a reinforced concrete structure that is lined with a 3/16 inch welded stainless steel liner. Collection trenches are formed into the concrete behind the welds to detect liner leakage. The leakage in the collection trenches is routed through a series of pipes to a central collection point. The pool structure is constructed of reinforced concrete and is a Class I seismic design.

The Point Beach spent fuel pool is divided in two parts that are connected through an internal divider wall. The single spent fuel pool is shared by the two units. The fuel transfer canal is set to the east of the spent fuel pool and is common to both units. Two gates maintain spent fuel pool inventory and allow the transfer canal to be drained for maintenance of fuel handling equipment. The elevation of the bottom of the gates is above the top of the spent fuel racks. The gates employ inflatable seals supplied by Instrument Air and a redundant static seal that is seated to the door jamb by hydrostatic force. Both gates must be closed to isolate the transfer canal from the spent fuel pool. Normally, one or both gates are left open and the transfer canal is normally flooded unless maintenance is going to be performed.

The north portion of the pool contains an area reserved for the loading of the spent fuel shipping cask or dry storage cask. There is also a new fuel elevator, located on the east side of the spent fuel pool on the Unit 2 side. The new fuel elevator is used to lower new fuel assemblies into the spent fuel pool and for maintenance on spent fuel assemblies. The pit for the new fuel elevator winch is located to the east of the transfer canal and the cable passes through an opening in the floor that is approximately 1 foot 4 inches below the 66 foot elevation. This opening also passes through the transfer canal into the spent fuel pool.

The spent fuel pool is approximately 42 feet deep with the top of the structure at the 66 foot elevation in the plant. The bottom of the structure is at the 24 foot - 8 inch elevation. The 26 foot elevation is considered to be ground level.

In the event of excessive makeup flow into the pool, the water would fill the pool to the level of the opening for the new fuel elevator winch. At this point water added to the spent fuel pool would spill into the transfer canal through the elevator opening. The water would fill the transfer canal until it equalizes with the spent fuel pool. The spent fuel pool and the transfer canal would continue to fill up to the 66 foot elevation, at which point water would begin to spill into the rail area where floor drains are located. The floor drains will route the water to the PAB sump and from there ultimately to the waste holdup tank. If water flow exceeds the capacity of the drains, it would overflow onto the 66 foot elevation operating deck. Some water would flow into the new fuel vault and some of it would flow off and onto the 46 foot elevation of the auxiliary building. The water would all be routed to floor drains and ultimately go to the waste holdup tank on the 8 foot elevation of the auxiliary building.

The volume of the spent fuel pool given in FSAR 9.9, Spent Fuel Cooling & Filtration (SF) is 48,283 ft^3. The low level alarm is set at elevation 62 foot - 8 inches. A substantial amount of the water volume is displaced by objects in the pool. The maximum number of assemblies that can be loaded in the spent fuel pool is 699 assemblies in the north pool and 803 assemblies in the south pool. Assuming the rack and fuel area is solid and contains no water and not accounting for the water volume in the cask laydown area, the borated water volume determined by this analysis at the low level alarm is 236,406 gallons.

2.2 Spent Fuel Storage Racks

The spent fuel storage racks for the Point Beach Nuclear Plant are designed in accordance with Regulatory Guide 1.29, Revision 2, as seismic Category I components. The structural analysis of the racks has considered all the loads and load combinations specified in the NRC Standard Review Plan. The steel structure of the rack not only provides a smooth, all welded stainless steel box structure to preclude damage during normal and abnormal load conditions, but also provides an additional margin of safety in the form of internal structural damping created by the large areas of bearing surface between boxes in the array.

2.3 Spent Fuel Pool Cooling System

The spent fuel pool cooling system, common to Units 1 and 2, is designed to remove decay heat from fuel assemblies stored in the spent fuel pool after removal from the reactor vessel. The spent fuel pool cooling system consists of two separate cooling trains, with a common suction and return header, each having an identical heat exchanger and pump. Water from the pool is pumped through one or both heat exchangers for cooling and returned to the pool. Normal operating procedures are used to cross-connect the pumps and heat exchangers as conditions require. In the unlikely event of the cooling loop of the spent fuel pool being drained, the spent fuel storage pool itself cannot be drained and no spent fuel is uncovered since the spent fuel pool cooling suction and return connections terminate or contain a siphon breaker that would limit water drawdown to a level approximately 21 feet 11 inches above the fuel.

The spent fuel pool cooling system piping and the service water system piping supplying the spent fuel pool heat exchangers are classified Safety-Related, Seismic Class 1.

The spent fuel pool cooling pumps take suction through branch lines off a common header from beneath the surface of the north half of the spent fuel pool, pump the water through the tube side of the spent fuel pool cooling heat exchangers, and return it via a common header to the south half of the spent fuel pool. The system piping is arranged so that either pump can supply either heat exchanger.

The spent fuel pool cooling heat exchangers are cooled by service water on the shell side. Because the heat exchangers are safety related and cooled with service water, they are part of the GL 89-13 program and receive regular inspection, performance testing and eddy-current testing of the tubes for degradation.

2.4 Spent Fuel Pool Cleanup System

The clarity and purity of the spent fuel pool water are maintained by passing up to the design flow of 60 gallons per minute through a filter and demineralizer.

The purification system inlet taps off the cross-connect line between the "A" and "B" cooling trains at the discharge of the fuel pool cooling pumps. The purification system return line connects with the cooling system return header. The purification system is not safety related.

The spent fuel filter removes particulate material from the spent fuel pool water. The filter cartridge is synthetic fiber and the vessel shell is stainless steel.

The demineralizer is sized to pass approximately 60 gallons per minute to provide adequate purification of the fuel pool water for unrestricted access to the working area, and to maintain optical clarity.

2.5 Dilution Sources

2.5.1 Chemical and Volume Control System (CVCS) Holdup Tanks

There are three CVCS holdup tanks, each with an approximate volume of 61,000 gallons. The CVCS hold up tanks may be pumped to the spent fuel pool or the transfer canal via a

4 inch line. The holdup tanks are used to drain the transfer canal and store the canal water during maintenance operations and supply water back into the canal when maintenance is complete. This connection is normally isolated by a manually operated valve from the discharge of the holdup tank recirculation pump. The CVCS holdup tanks are an approved method of makeup to the spent fuel pool, though additional administrative requirements are put in place when using this source. The CVCS holdup tanks may also be used as a source during a loss of inventory event.

The holdup tanks cannot gravity-drain to the spent fuel pool because the maximum tank water level is below the minimum spent fuel pool level.

The holdup tank recirculation pump is shared between Units 1 and 2 and is used to mix the contents of a holdup tank or transfer the contents of one holdup tank to another or transfer the spent fuel pool transfer canal water to the hold up tanks or spent fuel pool. The wetted surface of this pump is constructed of austenitic stainless steel. By procedure, only one holdup tank is aligned to the transfer pump at a time. Manual valve manipulations are required to switch the pump suction to another tank. Each holdup tank has a total volume of approximately 61,000 gallons. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle. Each holdup tank has a low level alarm at 13%. The design flow from this source is 500 gpm.

2.5.2 Reactor Makeup Water (RMUW) Tank.

One reactor makeup water tank is shared between the two units and is used to store makeup water, which is primarily supplied from the water treatment plant, but can also be supplied from the monitor tanks. The tank contains a diaphragm membrane and is constructed of coated carbon steel.

Two reactor makeup water pumps, shared between Unit 1 and Unit 2, take suction from the reactor makeup water tank. These pumps are used to feed dilution water to the boric acid blender and are also used to supply makeup water for intermittent flushing of equipment and piping. Each pump is sized to match the combined maximum letdown flow from each unit. One pump serves as a standby for the other.

The volume of the RMUW tank is approximately 96,150 gallons. The tank has a low level alarm at 4%. The tank administrative low level limit is 31%.

There is no direct flow path from the RMUW tank to the spent fuel pool. By plant procedure, reactor makeup water may be used for spent fuel pool makeup by sending it through either units' boric acid blender. From the boric acid blender the reactor makeup water is put into the spent fuel pool through the purification loop piping. During a loss of inventory accident, makeup is allowed by procedure through either units' boric acid blender. It could also possibly be routed to the spent fuel pool through the demineralizer flush line, though this is not a normal alignment and not allowed by procedure. The monitor tanks and pumps are used to flush the demineralizer line, which is isolated from the spent fuel pool during demineralizer flushing. The demineralizer flush valve is normally closed and locked shut.

The reactor makeup tank cannot gravity drain to the spent fuel pool because the top of the tank is lower than the top of the spent fuel pool low level alarm.

Makeup to the spent fuel pool using reactor makeup water through the boric acid blender is not preferred because of the number of manual valve manipulations involved. In addition the inlet valve to the spent fuel pool purification loop is a normally closed and locked shut valve. The design flow rate of a reactor makeup water pump is 270 gpm. The flow is limited by a flow control valve to 120 gpm.

2.5.3 Demineralized (DI) Water System

Demineralized water is supplied from the water treatment plant. DI water may be provided directly to the spent fuel pool cooling return line through a check valve and manually operated two inch diaphragm valve.

DI water is the typical means of makeup to the spent fuel pool. Additional administrative controls are put in place when using DI water to ensure that dilution below the technical specifications value does not occur. DI water may also be used during a loss of inventory event through the same flow path.

DI water is constantly supplied by the water treatment plant. The maximum flow rate from the water treatment plant is 400 gpm, though the actual amount that can be supplied to the spent fuel pool is approximately 200 gpm due to piping losses. 400 gpm is based on maximum values allowed in the plant operating procedure for the DI Water system.

2.5.4 Service Water

Each fuel pool cooling heat exchanger uses 3/4 inch U-tubes with service water on the shell side. There is no direct piping connection between the service water system and the spent fuel pool cooling system. The normal operating pressure of the service water system is higher than the normal operating pressure of the spent fuel pool cooling system. In the event of a heat exchanger tube break, differential pressure will normally result in leakage from the service water system to the spent fuel pool cooling system. Under certain conditions, for example during refueling when higher service water flow rates to the spent fuel pool heat exchangers are required, service water pressure may fall below spent fuel pool cooling system pressure. Under these conditions, a heat exchanger tube break will result in leakage from the spent fuel pool cooling system into the service water system. A spent fuel pool heat exchanger tube rupture is considered improbable based upon the low operating pressures, the seismic installation of the heat exchanger, and the heat exchanger design specifications.

Service water is operated between 50 psig and 90 psig. The discharge pressure of the spent fuel pool water at the heat exchanger outlet is low and typically less than 10 psig.

Conservatively assuming a 90 psi differential pressure between service water and the spent fuel pool, the expected flow through a 3/4 inch opening would be less than 200 gpm. If a leak were to develop given the size of the tubes, the flow rate would be bounded by the dilution from the DI water system.

It is expected that the flow rate of any leakage of service water would be very low due to the small difference in operating pressures between the two systems. Given that such an event is considered improbable, no further consideration will be given to service water as a dilution source.

In a loss of inventory event, service water may be used to add water to the spent fuel pool, though no direct connection exists and there is presently no detailed procedure guidance to perform this action. Also, the spent fuel pool as analyzed under the new criticality analysis will remain subcritical, even if filled with unborated water. The only time when service water would be used to add inventory to the spent fuel pool would be because some more limiting event had occurred.

2.5.5 Fire Protection System

In an emergency loss of spent fuel pool inventory, fire hose stations are available. There is one hose reel in each fan room and hose reels on the PAB 46 foot elevation, central area. The design flow rate of a hose station is 100 gpm of non-borated raw water. Although an available source, the fire hose is not specifically addressed by normal operating procedures for makeup and would only be required if some more limiting event were to occur where cooling and shielding of the spent fuel is the primary concern. Even if all four fire hoses were positioned into the spent fuel pool the flow rate would be 400 gpm and this is bounded by the DI water dilution event. In an event such as this, the criticality analysis still ensures that the spent fuel pool will remain subcritical, even in the presence of unborated water.

2.5.6 Monitor Tanks

Four monitor tanks can be shared by Unit 1 and Unit 2. Each tank has a capacity of approximately 10,000 gallons. Liquid effluent in the holdup tanks can be routed to the monitor tanks via the boric acid feed demineralizers for subsequent discharge. The tanks are located on the 26 foot elevation of primary auxiliary building. Two monitor tank pumps, shared by Units 1 and 2, discharge water from the monitor tanks. The pumps are constructed of austenitic stainless steel. The monitor tanks can also be filled with water from the water treatment plant. These tanks contain a diaphragm membrane and are constructed of stainless steel. The tanks have a low level alarm at 5% and a pump trip at 10%.

The monitor tanks and pumps are used during demineralizer resin replacement to flush the resin. During these operations, only one tank and one pump are aligned at a time to the spent fuel pool demineralizer. The purification loop for spent fuel cooling is isolated during resin replacement activities. The resin flush valve is normally closed and locked shut during purification loop operation.

The monitor tanks cannot gravity drain to the spent fuel pool because the top of the tanks is below the low level alarm in the spent fuel pool.

Because the monitor tanks take waste from the CVCS holdup tanks and can also be filled with water from the water treatment plant, their boron concentration can be variable. Normally, only one monitor tank is aligned to the pump during resin replacement activities. Manual valve manipulations and intentional disregard for operating procedures would be required to switch the pump to another tank. The design flow rate of one monitor tank pump is 60 gpm.

2.5.7 Dry Cask Storage Operations

During dry fuel storage evolutions, spent fuel pool water is added to the cask prior to placement in the spent fuel pool. After fuel loading the water is pumped back to the spent fuel pool. Additional sampling requirements are put in place as required by cask technical specifications. Since the water pumped from the cask was from the SFP, this is not considered a dilution source. DI water may used to rinse the cask down as it is removed from the spent fuel pool. Additional procedure controls are in place during these evolutions and require a flow totalizer be installed and limit the total rinse volume to 500 gallons. Rinse down work is conducted with a garden hose, further limiting the rate of flow. This is not considered a credible dilution source and will not be further evaluated.

2.5.8 Dilution Source and Flow Rate Summary

Based on the evaluation of potential spent fuel pool dilution sources summarized above, the following dilution sources were determined to be capable of providing a significant amount of non-borated water to the spent fuel pool. The potential for these sources to dilute the spent fuel pool boron concentration to the design basis boron concentration will be evaluated in Section 3.0.

Source	Design Flow Rate (GPM)
CVCS	
- Holdup tank to Spent Fuel Pool	500
RMUW	
- Unit 1 or 2 Boric Acid Blender	120
DI	
- Via SF-00812B valve	400
Monitor Tanks	
- Through demineralizer flush	60
Fire Protection	
- Fire hose station	100

2.6 Boration Sources

The normal source of borated water to the spent fuel pool is the boric acid blender. The other source is the CVCS hold up tanks, which are checked for acceptable boron concentration prior to use. Another possibility would be the addition of dry boric acid directly to the spent fuel pool water. The Refueling Water Storage Tanks (RWST) is also a possible borated water source.

2.6.1 Refueling Water Storage Tank

There is one Refueling Water Storage Tank (RWST) for each unit. Each RWST contains approximately 290,000 gallons of water borated to approximately 3000 ppm (Technical Specifications require the RWST to be maintained greater than 275,000 gallons and greater than 2700 ppm.)

The refueling water circulating pump is used primarily to circulate water in a loop between the RWST and the spent fuel pool demineralizer and filter. All wetted surfaces of the pump are austenitic stainless steel. The pump is operated manually from a local station.

RWST makeup to the spent fuel pool is only used in the case of a loss of inventory event. The RWST is not a source for normal makeup to the spent fuel pool . The refueling water circulation pump is powered from a non-vital bus power supply.

2.6.2 Boric Acid Storage Tanks (BAST)

The BASTs are an approved source of makeup to the spent fuel pool through either units' boric acid blender. This source is also approved for use during a loss of inventory accident. Because of the number of manual valve manipulations, this is not the preferred method to makeup to the spent fuel pool.

There are three boric acid storage tanks. Each of the three boric acid storage tanks has a capacity of 5000 gallons. The tanks are located on the 46 foot elevation of the primary auxiliary building.

2.6.3 Direct Addition of Boric Acid

If necessary, the boron concentration of the spent fuel pool could be increased by emptying bags of dry boric acid directly into the spent fuel pool. However, boric acid dissolves very slowly at room temperature and requires that the spent fuel pool cooling pumps be available for mixing the spent fuel pool water. Furthermore, there is no procedure currently in place to provide operator guidance for this method. Therefore, this method would be used only in an emergency and will not given additional consideration.

2.6.4 CVCS Holdup Tanks

The CVCS holdup tanks may be borated depending on which tank is selected and the time in core life. For the purposes of this analysis, the CVCS holdup tanks are assumed to be dilution sources.

2.7 Spent Fuel Pool Instrumentation

Instrumentation is available to monitor spent fuel pool water level and temperature. Additional instrumentation is provided to monitor the pressure and flow of the spent fuel pool cleanup system, and pressure, flow and temperature of the spent fuel pool cooling system.

The instrumentation to monitor spent fuel pool temperature and level alarm on a common annunciator in the control room. The alarm actuates on high spent fuel pool temperature and high or low spent fuel pool level. The temperature and level alarms are powered from the vital DC power supply.

Two area radiation monitors are available in the spent fuel pool area for low range and high range area monitoring.

The spent fuel pool low level alarm is set at 62 foot - 8 inches and the high level alarm is set at 64 foot - 10 inches. The temperature alarm is set for greater than 120 degF.

2.8 Administrative Controls

The following administrative controls are in place to control the spent fuel pool boron concentration and water inventory:

1. Procedures are available to aid in the identification and termination of dilution events.

2. Procedures for loss of inventory (other than normal makeup) are ordered such that borated water sources are used first.

3. Procedure for makeup allows use of DI water, RMUW, and CVCS hold up tank water provided it meets additional requirements for sampling and/or initial boron concentration.

4. In accordance with procedures, plant personnel perform rounds at the spent fuel pool at least once every 8 hours. The personnel making rounds to the spent fuel pool are trained to be aware of the change in the status of the spent fuel pool. They record temperature and level on data loggers during these rounds (temperature from the local indicator in the area is recorded every 8 hours, level is recorded once per day.)

5. Administrative controls (locked closed valves on RMUW flow paths to the spent fuel pool cooling system; procedures requirements) are placed on the potential dilution paths.

6. When making up using DI water or RMUW, procedures require that the spent fuel pool initial boron concentration is greater than or equal to 2,500 ppm, the spent fuel cooling system is operating with a flow rate greater than or equal to 1,000 gpm and the makeup is limited to less than or equal to 12 inches of level. Prior to adding more water, the boron concentration must be re-verified greater than 2,500 ppm. CVCS holdup tanks must meet minimum chemistry requirements including boron concentration before being used as a makeup source.

7. The spent fuel pool boron concentration is administratively maintained at greater than 2,300 ppm and it is typically around 3,000 ppm. It is sampled every 7 days per technical specifications.

2.9 Piping

There are no systems identified which have piping in the vicinity of the spent fuel pool which could result in a dilution of the spent fuel pool if they were to fail.

3.0 SPENT FUEL POOL DILUTION EVALUATION

3.1 Calculation of Boron Dilution Times and Volumes

For the purposes of evaluation of spent fuel pool dilution times and volumes, the total pool volume available for dilution, as described in section 2.1, is conservatively assumed to be 236,406 gallons.

Based on the new criticality analysis, the soluble boron concentration required to maintain the spent fuel pool at $k_{eff} < 0.95$, including uncertainties with a 95% probability and 95% confidence level is 805 ppm.

The spent fuel pool boron concentration is typically maintained above the administrative value of 2,300 ppm at around 3,000 ppm (technical specification value is 2,100 ppm.) If the concentration falls below 2,100 ppm, Point Beach enters a Technical Specification Action Condition to restore the concentration to within limits immediately. For the purposes this of evaluation, the initial spent fuel pool boron concentration is assumed to be at the technical specification limit of 2,100 ppm. The evaluations are based on the spent fuel pool boron concentration being diluted from 2,100 ppm to 805 ppm. To dilute the spent fuel pool volume of 236,406 gallons from 2,100 ppm to 805 ppm would conservatively require 251,763 gallons of non-borated water. This is based on initially filling the spent fuel pool to the elevation where water spills into the transfer canal, then filling the transfer canal, filling the spent fuel pool and the transfer canal to the top of the structure and then spilling over the structure onto the floor. This sequence of events maximizes the time until the high level alarm would be actuated.

This analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool with the contents of the spent fuel pool. Based on the design flow of 1,250 gpm per spent fuel pool pump, the 236,406 gallon system volume is turned over approximately every 3 hours with one pump running, which is the normal alignment. It is unlikely with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing was not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the criticality analysis which shows that the spent fuel rack k_{eff} will be less than 1.0 with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, k_{eff} would not exceed 1.0 anywhere in the pool.

3.2 Evaluation of Boron Dilution Events

The time to dilute the spent fuel pool depends on the initial volume of the pool and the postulated rate of dilution. The potential spent fuel pool dilution events that could occur are evaluated below.

3.2.1 Dilution from CVCS Holdup Tanks

Dilution water from a CVCS holdup tank can be transferred via the recirculation pump to the spent fuel pool directly. The flow path to the spent fuel pool is isolated through a normally closed valve. The tanks are also kept isolated as a source to the pump through normally closed valves until the water is needed to be moved. This connection is a designated source of makeup water in a loss of spent fuel pool inventory event. This is also a designated source of normal makeup to the spent fuel pool. Each of the three CVCS holdup tanks has a total volume of approximately 61,000 gallons. The water in the tanks has a variable boron concentration which could be as low as 0 ppm. Any amount of boron in the CVCS holdup tank water would increase the required dilution volume from transfer of CVCS holdup tank water to the spent fuel pool. To dilute the spent fuel pool volume from 2,100 ppm to 805 ppm requires 251,763 gallons of unborated water. The combined contents of three CVCS holdup tanks (approximately 183,000 gallons) is less than the total required dilution volume. The recirculation pump is rated to flow at 500 gpm. If transfer of the CVCS holdup tanks were initiated and left unattended, it would take approximately 199 minutes to increase the spent fuel pool level from the low to high alarm setpoint and 8 hours to provide the 251,763 gallons required to dilute the pool from 2,100 ppm to 805 ppm, assuming 0 ppm boron in the tanks and an unlimited supply in the tanks. Note that the low level alarm for the Primary Auxiliary Building operator is 13% level in the tanks. In addition, the B holdup tank is administratively maintained with sufficient boron at 3.5 weight percent to support a plant cooldown. The boron in the this tank would further reduce the total volume of dilution water that can be supplied to the spent fuel pool.

The CVCS recirculation pump can take suction from either of three CVCS holdup tanks. Administrative procedures specify that the pump is aligned to one holdup tank at a time. Manual valve manipulations are required to switch the pump suction to another tank. Thus, it is assumed for the purposes of this evaluation that only the contents of one CVCS holdup tank are available for a spent fuel pool dilution event. The 61,000 gallons of water contained in one CVCS holdup tank is less than the 251,763 gallons necessary to dilute the spent fuel pool from 2,100 ppm to 805 ppm. There is no automatic makeup to the CVCS holdup tanks.

3.2.2 Dilution From Reactor Makeup Water Storage Tank

The contents of the Reactor Makeup Water Storage (RMUW) tank cannot be transferred via the RMUW pumps directly to the spent fuel but it can be indirectly transferred via either units' boric acid blender. It could also be transferred to the spent fuel pool via the purification loop through the demineralizer flush line, though this is not in a plant approved procedure and would require the mis-positioning of manual valves.

The RMUW system consists of a single water storage tank and two pumps for both operating units. RMUW can be supplied to the spent fuel pool cooling system from the tank and pumps through either units' boric acid blender. This is an approved makeup method for both normal pool makeup and loss of inventory events. The RMUW tank contains approximately 96,150 gallons of de-ionized water. Because 251,763 gallons of water is required to dilute the spent fuel pool from 2,100 ppm to 805 ppm, the tank does not have sufficient inventory to dilute the spent fuel pool.

In addition, only one of the two RMUW pumps is kept available at any time. The other pump is maintained in a pullout condition. The RMUW pumps are not in constant operation, but start only on command from the control room.

The design flow rate of a reactor makeup water pump is 270 gpm. The flow is limited by a flow control valve to 120 gpm. If makeup to the spent fuel pool were started and then left unattended (and the tank had an unlimited supply), the pool would rise from the low level alarm to the high level alarm in 829 minutes. For the given flow rate it would take 35 hours to supply the required 251,763 gallons. There is no automatic makeup to the RMUW tank.

3.2.3 Dilution from the Demineralized (DI) Water System

DI water is supplied and administratively controlled from the water treatment plant in the turbine building of the plant. The DI system can makeup directly to the spent fuel pool through a 2 inch connection to the purification loop return line. The demineralized water system is rated to supply water at 400 gpm, though actual available supply rate is about 200 gpm. Use of DI water to makeup to the spent fuel pool is controlled by procedure. At the specified flow rate, it would take 249 minutes to increase the spent fuel pool to the high level alarm and 10 hours to add the required dilution volume of 251,763 gallons.

3.2.4 Dilution from Fire Protection System

The fire protection system draws raw water directly from the lake. In order to have firewater makeup to the spent fuel pool, a hose station would need to be unrolled and a nozzle positioned to the spent fuel pool and constantly attended. The nearest fire hoses are the Unit 1 and 2 fan rooms or from the 46 foot of the PAB. The fire protection system is estimated to be capable of supplying about 100 gpm at the nozzle. The only reason fire water would be used is in the case of an emergency. At the given flow rate it would take 994 minutes to raise the water level to the high alarm and 42 hours to provide the necessary 251,763 gallons of dilution.

3.2.5 Dilution from the Monitor Tanks

The monitor tanks consist of four tanks, each 10,000 gallons. The tanks may contain unborated water or borated water awaiting discharge. These tanks and their associated pumps are the source of water used during demineralizer flushing of the spent fuel pool. Demineralizer flushing and recharge are administratively controlled so that the demineralizer is isolated from the spent fuel pool during recharge. In addition, only one monitor tank is allowed to be used at a time. Even if the all four tanks were aligned, the dilution source is less than that required to dilute the spent fuel pool.

The rated flow of the monitor tank pump is 60 gpm. At this flowrate (assuming an unlimited supply) it would take 1,657 minutes to raise the level from the low level alarm to the high level alarm. It would take 70 hours to supply the total dilution volume.

3.2.6 Review of Operating Experience

1. LER 369-94005, McGuire, 7/10/1994, The spent fuel pool boron concentration was diluted below the technical specification limit. This when the transfer canal was being pumped to the spent fuel pool in preparation for maintenance. A DI misting system was placed in service during the draindown to limit airborne activity. The DI water mixing with the transfer canal water diluted the water as it was pumped to the spent fuel pool. The spent fuel pool was 50 ppm below technical specification limit. Boron was added to restore the spent fuel pool to above the technical specification limit.

Point Beach does not have a similar misting system installed. If decontamination of the transfer canal walls is necessary prior to maintenance, a flow totalizer is used to track how much water is added to the transfer canal. Prior to pumping water from the holdup tank back to the spent fuel pool, the hold up tank must be analyzed to ensure the boron concentration is not below the technical specification limit. When draining the canal water is first pumped to the spent fuel pool and then to the holdup tanks by procedure.

2. LER 289-980204, Three Mile Island, 2/4/1998, Operators failed to notify chemistry to sample the spent fuel pool after adding makeup water. This was a repeat occurrence. The spent fuel pool boron concentration was not diluted below the technical specification minimum.

By procedure the spent fuel pool boron concentration must be verified to be greater than 2,500 ppm prior to filling. The total fill volume is limited to 12 inches of level and must be re-verified prior to additional filling using DI water or RMUW.

3.3 Summary of Dilution Events

The five available water sources for spent fuel pool dilution are RMUW, DI water, CVCS holdup tanks, monitor tanks and fire protection. Fire protection is the least likely source since it would only be used as a measure of last resort in a loss of inventory accident and because the makeup hose is not located in the vicinity of the spent fuel pool. The monitor tanks are the next least likely source since they are only used for demineralizer flushing and not for normal or emergency makeup. The RMUW tank is the next least likely source since there is not a direct makeup path to the spent fuel pool. It may be used but is not the preferred source because of the required valve lineup. The CVCS holdup tank source is the second most likely, but they are normally borated to some degree. The volume of all three tanks is less than that required to dilute the spent fuel pool from 2,100 ppm to 805 ppm. The DI water source is the most likely source because it has a direct connection and an unlimited supply from the water treatment plant. It is also the preferred makeup source for the spent fuel pool.

Flow rates from the DI water system supply pump vary depending on plant mode and other plant demands. The maximum flow the DI water system can supply is 400 gpm with two pumps running. Typically only one pump and demineralizer are in service, limiting the maximum design output to 200 gpm. The actual flow rate would be less given the length of the piping run and pipe size. Even at the maximum flow rate, it takes 249 minutes to fill the spent fuel pool to the high level alarm assuming the pool level was initially at the low level alarm. If the transfer canal were full, as is the normal case, the high level alarm would alert the control room much sooner. Assuming that the high level alarm were to fail, the pool would overflow, spilling onto the refueling floor, resulting in water filling the PAB sump. If the flow exceeds the capacity of the drains, it would flow out over the refueling deck and into other parts of the building. All water would eventually end up in the waste holdup tank. The waste holdup tank volume is 23,960 gallons with a high alarm at 63% (variable) of tank level and a high-high alarm at 85% tank level. Thus, the waste holdup tank would act as a secondary backup to the spent fuel pool high level alarm. By procedure, the operator must inform the water treatment operator prior to filling the spent fuel pool. Continued makeup to the spent fuel pool should be noticed by the water treatment operator. In addition, it would take 10 hours to reach the required dilution volume and routine operator rounds of the area would identify the overflow of the spent fuel pool.

Furthermore, for any dilution scenario to successfully add 251,763 gallons of water to the spent fuel pool, plant operators would have to fail to question or investigate the continuous makeup of water to the spent fuel pool for the required time period, and fail to recognize that the need for extended water supply to the spent fuel pool was unusual.

4.0 CONCLUSIONS

A boron dilution analysis has been completed for the spent fuel pool. As a result of this spent fuel pool boron dilution analysis, it is concluded that an unplanned or inadvertent event which would result in the dilution of the spent fuel pool boron concentration from 2,100 ppm to 805 ppm is not a credible event. This conclusion is based on the following:

The preferred method of normal makeup to the spent fuel pool is DI water from the water treatment plant. Use of this source requires verification of the spent fuel pool concentration prior to filling and limits the amount that may be filled. Additional filling requires re-verification of the spent fuel pool boron concentration.

If an inadvertent dilution were to be initiated, administrative procedures are in place to address a high level alarm in the spent fuel pool. Borated water from the RWST is available via the refueling water circulation. Borated water is also available from the BAST via the boric acid blender of either unit to the purification loop of the spent fuel cooling system.

In order to dilute the spent fuel pool to k_{eff} 0.95, a substantial amount of water (251,763 gallons) is needed. To provide this volume, an operator would have to initiate the dilution flow, then abandon monitoring of the pool level, and ignore administrative procedures, and a high level alarm for a period of at least 10 hours. The required dilution volume of 251,763 gallon exceeds the volume of all unborated water sources in the plant used for normal makeup with the exception of the DI water system.

Since such a large water volume turnover is required, a spent fuel pool dilution event would be readily detected by plant personnel via alarms, flooding in the primary auxiliary building, or eventually by operator rounds through the spent fuel pool area.

It should be noted that this boron dilution evaluation was conducted by determining the time and water volumes required to dilute the spent fuel pool from 2,100 ppm to 805 ppm. The 805 ppm endpoint was used to ensure that k_{eff} for the spent fuel racks would remain less than or equal to 0.95. As part of the criticality analysis for the spent fuel racks, a calculation has been performed to show that the spent fuel rack k_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the spent fuel pool were diluted to zero ppm, which would take significantly more than the volume determined above, the spent fuel pool would remain subcritical and the health and safety of the public would be assured.

Appendix A - Volumes

PURPOSE

The purpose of Appendix A is to determine (1) the volume of water in the spent fuel pool at the low level alarm; (2) the volume of water to raise the spent fuel pool to the new fuel elevator spill over elevation; (3) the volume of water to raise the transfer canal to the new fuel elevator spill over elevation; (4) the volume of water to raise the spent fuel pool and transfer canal to the high level alarm; and (5) the volume of water to raise the spent fuel pool and transfer canal to the top of the structure.

BACKGROUND
The new spent fuel pool criticality analysis, WCAP-16541-P, Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis (Ref. 11), uses the Westinghouse methodology in WCAP-14416-P. WCAP-14416-P credits soluble boron to mitigate potential accidents. In its approval of WCAP-14416-P, the NRC requires that licensees identify potential dilution events.

Part of this analysis includes an evaluation of the times to identify the dilution events. Flow rates of the dilution sources are available from plant information. In order to determine the time, the volume of the spent fuel pool and transfer canal needs to be known.

The volumes will be broken up into the following parts to facilitate the information's use in the analysis:

(a) The volume to fill the spent fuel pool from the low level alarm to the spill over point.
(b) The volume to fill the transfer canal from empty to the spill over point.
(c) The volume to fill both the spent fuel pool and the transfer canal from the spill over point to the high level alarm.
(d) The volume to fill the spent fuel pool and transfer canal from the high level alarm to the top of the structure.

ASSUMPTIONS

Validated Assumptions

1. It is assumed that the doors to the transfer canal are closed and the transfer canal drained.
Basis: It is conservative to assume the doors are closed and the transfer canal is drained. This decreases the volume of water available to be diluted and increases the time until the high level alarm is reached (because the high level alarm elevation is greater than the spill over point elevation.)

2. It is assumed that the water level in the SFP is at the low level alarm.
Basis: It is conservative to assume the water is at the low level alarm. This decreases the volume of water to be diluted and thus decreases the required dilution volume. This is also the condition when plant operators would initiate normal makeup to raise the spent fuel pool level.

3. It is assumed that the pool fills with unborated water from the low level to the spill over point through the new fuel elevator cable hole cut in the divider wall. The transfer canal will then fill to the spill over point. Both the transfer canal and spent fuel pool will then fill together until the top of structure is reached, when spilling over to the floor occurs.
Basis: This is the most likely scenario that does not involve some type of malicious action on the part of a plant employee.

4. It is assumed the high level alarm does not annunciate until transfer canal is filled and the spent fuel pool and transfer canal fill to the high level alarm.
Basis: Plant operating experience shows that during filling of the spent fuel pool from the CVCS hold up tanks, with a sufficiently high flow rate the high level alarm will temporarily annunciate even though water is spilling over to the transfer canal. By assuming the high level alarm is not reached, a greater dilution volume will occur before the control room is notified.

5. It is assumed that the transfer canal volume can be represented by the following: (a) volume for the expanded portion by the transfer tube (b) volume for the middle section that is the same elevation as the transfer tube ends (c) volume for the region from elevation for (a) and (b) to the elevation where the transfer canal is no longer angled up, and (d) elevation in the straight vertical region to spill over point, and (e) volume from spill over point to top of transfer canal.
Basis: This is an accurate representation of the transfer canal and does not account for equipment such as the fuel transfer system or the transfer tube. See Figure 1 for a graphical description.

6. It is assumed that the volume of spaced in the spent fuel pool occupied by the fuel contains no water.
   Basis: This is a conservative because the amount of water available to be diluted is decreased.

Un-validated Assumptions

None

REFERENCES (verified current as of 6/30/2006).

1. N-93-048, "Time to SFP Boiling Following a Loss of SFP Cooling", Rev 3.
2. Setpoint Document, STPT 8.1, LC-634 SFP Level, Rev 7.
3. Drawing BECH C-203, PAB SFP Liner Plate, Rev 8.
4. Drawing BECH C-160, PAB SFP Plans & Sections, Rev 9.
5. Technical Specification 3.7.11, Fuel Storage Pool Boron Concentration.
6. NF-NMC-06-5, Point Beach Spent Fuel Pool Criticality Analysis, 1/27/06.
7. Drawing BECH C-206, PAB SFP Elevator Winch, Rev. 2
8. Drawing BECH C-204, Cont/PAB Refueling Canal and SFP Liner Plate, Rev. 12
9. Drawing BECH C-202, Cont/PAB Refueling Canal and SFP Liner Plate, Rev. 6
10. Drawing BECH C-205, PAB SFP Liner Plate, Rev 7
11. WCAP-16541-P, Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis, 2/1/2006.

INPUTS

1.	Gross volume of SFP water from low level alarm.	V_{Gross}	50,160 ft^3	Ref. 1
2.	Volume of divider wall	V_{DW}	1,877 ft^3	Ref. 1
3.	Volume of cask area	V_{CA}	1,213 ft^3	Ref. 1
4.	Volume of fuel zone	V_{FZ}	15,467 ft^3	Ref. 1
5.	Low level alarm	H_0	62 ft - 8 in	Ref. 2
6.	Distance to the spill over point	D_1	1 ft - 4 5/16 in	Ref. 7
7.	Upper elevation of SFP structure	E_2	66 ft - 0 in	Ref. 3
8.	Length of SFP	L_0	72 ft - 0 in	Ref. 4
9.	Width of SFP	W_0	18 ft - 4 in	Ref. 4
10.	Width of divider wall (west end)	W_8	4 ft	Ref. 4
11.	Length of divider wall (west end)	L_8	8 ft	Ref. 4
12.	Upper elevation of divider wall (west end)	E_0	63 ft - 8 in	Ref. 4
13.	Minimum SFP boron	C_0	2100 ppm	Ref. 5
14.	Required SFP boron	C_f	805 ppm	Ref. 6
15.	Length of transfer canal (bottom) (w/o ends) (per unit)	L_4	48 ft - 4 in	Ref. 3
16.	Length of transfer canal (bottom) (ends)	L_1	5 ft - 1 5/8 in	Ref. 3
17.	Elevation of transfer canal (bottom)	H_2	24 ft - 8 in	Ref. 3
18.	Width of transfer canal (mid section)	W_5	3 ft	Ref. 3
19.	Width of transfer canal (ends)	W_3	5 ft	Ref. 4
20.	Elevation of transfer canal (ends, top)	H_1	31 ft - 6 in	Ref. 9
21.	Length of transfer canal (top) (per unit)	L_6	37 ft - 1 1/2 in	Ref. 10
22.	Length of vertical transfer canal section	D_2	3 ft - 4 in	Ref. 8
23.	Width of transfer canal divider	L_7	4 ft	Ref. 3
24.	Length of transfer canal divider (vertical)	D_3	35 ft - 4 in	Ref. 3
25.	Length of the end transition region	L_2	1 ft	Ref. 3
26.	High level alarm	E_6	64 ft - 10 in	Ref. 2

ATTACHMENTS

None.

ACCEPTANCE CRITERIA

None. The output of this numerical manipulation is a determination of the volume of water necessary to fill the Spent Fuel Pool (SFP) from the low level alarm to the top of the spent fuel pool structure. This volume will be used as part of dilution analysis for the spent fuel pool.

METHODOLOGY AND CALCULATION

Determine SFP water volume at low level alarm:

The SFP contains water, fuel racks, fuel, an internal divider wall and a cask area. This information is taken as inputs from calculation N-93-048 (Reference 1). The volume of water in the SFP at the low level alarm can be expressed as gross volume [input 1] minus the divider wall volume [input 2] minus the cask area volume [input 3] minus the fuel zone volume [input 4]:

$V_1 = V_{Gross} - V_{DW} - V_{CA} - V_{FZ}$

V_1 = Volume of water in SFP at low level alarm, ft^3
V_{Gross} = Gross volume, ft^3
V_{DW} = Divider wall volume, ft^3
V_{CA} = Cask area volume, ft^3
V_{FZ} = Fuel zone volume, ft^3

This treatment does not consider the volume of water in the cask area, the volume of water in the fuel zone, or in areas around the spent fuel racks or inspection area (see Assumption 6 of this Appendix.)

$V_1 = 50{,}160\ ft^3 - 1{,}877\ ft^3 - 1{,}213\ ft^3 - 15{,}467\ ft^3 = 31{,}603\ ft^3$

Put the volume in terms of gallons:

$V_1 = 31{,}603\ ft^3 * 7.4805\ gal/\ ft^3 = \underline{236{,}406\ gal}$

(a) Determine the SFP water volume from low level alarm to spill over elevation:

During the initial dilution, the SFP will fill from the low level alarm to the transfer canal spill over elevation through the new fuel elevator opening.

The elevation of this opening needs to be determined. It is found by taking the elevation of the top of the spent fuel pool [input 7] and subtracting the distance to the spill over point [input 6].

$E_1 = E_2 - D_1$

E_1 = Elevation of the spill over point, ft
E_2 = Elevation of the top of the spent fuel point (reference point), ft
D_1 = Distance to the spill over point from the top of the spent fuel pool, ft

$E_1 = (66\ ft + 0\ in) - (1\ ft + 4/12\ ft + 5/16/12\ ft) = 64\ ft - 7\ 11/16\ inch$

The volume of the spent fuel pool from the low level alarm to the spill over point is given by the volume of the divider wall subtracted from the gross volume to fill the spent fuel pool to the spill over point.

$V_2 = V_3 - V_4$

V_2 = Volume of water to fill SFP from low level alarm to spill over point, ft^3
V_3 = Gross volume of SFP from low level alarm to spill over point, ft^3
V_4 = Volume of divider wall from low level alarm to top its of structure, ft^3

The gross volume of the SFP from the low level alarm to the spill over point is the length of the SFP [input 8] multiplied by the width [input 9] by the elevation of the spill over point [E_1] minus the elevation of the low level alarm [input 5]:

$V_3 = L_0 * W_0 * (E_1 - H_0)$

V_3 = Gross volume of SFP from low level alarm to spill over point, ft^3
L_0 = Length of the spent fuel pool, ft
W_0 = Width of spent fuel pool, ft
E_1 = Elevation of the spill over point, ft
H_0 = Elevation of the low level alarm, ft

V_3 = (72 ft) (18 ft + 4/12 ft) ((64 ft + 7/12 ft + 11/16/12 ft) - (62 ft + 8/12 ft))

$V_3 = 2{,}606\ ft^3$

The volume of the divider wall from the low level alarm to the top of its structure is given by the length of the divider wall [input 11] multiplied by the width of the divider wall [input 10] multiplied by the upper elevation of the divider wall [input 12] minus the low level alarm [input 5]:

$V_4 = L_8 * W_8 * (E_0 - H_0)$

V_4 = Volume of divider wall from low level alarm to top its of structure, ft^3
L_8 = Length of the spent fuel pool divider wall, ft
W_8 = Width of spent fuel pool divider wall, ft
E_0 = Elevation of top of spent fuel pool divider wall, ft
H_0 = Elevation of the low level alarm, ft

V_4 = (8 ft) (4 ft) [(63 ft + 8/12 ft) - (62 ft + 8/12 ft)] = 32 ft^3

The volume of water to fill the SFP from the low level alarm to the spill over point is then given by:

$V_2 = 2{,}606\ ft^3 - 32\ ft^3 = 2{,}574\ ft^3$

Put the volume in terms of gallons:

$V_2 = 2{,}574\ ft^3 * 7.4805\ gal/\ ft^3 = \underline{19{,}252\ gal}$

(b) Determine the volume of the transfer canal to the spill over point

The volume of the transfer canal can be approximated by dividing it up into four sections. (a) The volume of the ends where the transfer tubes are located. (b) The volume of the middle section between the ends. (c) The transfer canal volume with angled walls. (d) The volume of the transfer canal with vertical walls. Also, the volume of the transfer canal divider wall will be subtracted.

(a) Volume of the ends

The volume of the ends is given by the length of the ends [input 16] plus the length of the transition region [input 25] multiplied by the width of the ends [input 19] multiplied by the elevation of the bottom of the transfer canal [input 17] subtracted from the elevation of the top of the ends [input 20]. This is multiplied by two for both units.

$V_5 = (L_1 + L_2) * W_3 * (H_1 - H_2) * 2$

V_5 = Volume in the ends of the transfer canal, ft^3
L_1 = Length of the ends, per unit, ft
L_2 = Length of transition region, ft
W_3 = Width of ends, ft
H_1 = Upper elevation of ends, ft
H_2 = Elevation of transfer canal bottom, ft

$V_5 = [(5\ ft + 1/12\ ft + 5/8/12\ ft) + (1\ ft)] * (5\ ft) * [(31\ ft + 6/12\ ft) - (24\ ft + 8/12\ ft)] * 2$

$V_5 = 419\ ft^3$

(b) Volume of the transfer canal between the ends

The volume of water between the ends in the transfer canal is the length of the transfer canal at the bottom [input 15] multiplied by the width of the transfer canal [input 18] multiplied by the elevation of the bottom of the transfer canal [input 17] subtracted from the elevation at the top of the ends [input 20] multiplied by two for both units.

$V_6 = L_4 * W_5 * (H_1 - H_2) * 2$

V_6 = Volume in the bottom of the transfer canal between the ends with same elevation, ft^3

L_4 = Length of the section between the ends, per unit, ft
W_5 = Width of the transfer canal, ft
H_1 = Upper elevation of ends, ft
H_2 = Elevation of transfer canal bottom, ft

V_6 = (48 ft + 4/12 ft) * (3 ft) * [(31 ft + 6/12 ft) - (24 ft + 8/12 ft)] *2

V_6 = 1,982 ft^3

<u>(c) Volume of the transfer canal in section with angled walls</u>

First the elevation where the angled walls end must be determined (upper elevation.) This is found by subtracting the height of the straight section [input 22] from the top of spent fuel pool elevation [input 7].

$E_3 = E_2 - D_2$

E_3 = Elevation where the angled wall starts, ft
E_2 = Elevation of the top of the SFP, ft
D_2= Length of the upper vertical section of the transfer canal, ft

E_3 = 66 ft - (3 ft - 4/12 ft)

E_3 = 62 ft - 8 in

The volume of water in the portion of the transfer canal with angled walls is further divided into the center section and triangle section.

The center section can be given by multiplying the width of the transfer canal [input 18] by the difference in elevation from the top of the ends elevation [input 20] to where the vertical section starts [E_3] by the length of the transfer canal per unit [input 21] multiplied by 2. This is the volume in the center section.

$V_7 = W_5 * (E_3 - H_1) * L_6 * 2$

V_7 = Volume in-between the angled wall sections , ft^3
W_5 = Width of the transfer canal, ft
E_3 = Elevation where the angled wall starts, ft
H_1 = Upper elevation of ends, ft
L_6 = Length of the transfer canal at the top elevation, per unit, ft

V_7 = 3 ft * [(62 ft + 8/12 ft) - (31 ft + 6/12 ft)] * (37 ft + 1/12 ft + 1/2/12 ft) * 2

V_7 = 6,942 ft^3

There is one triangle section for each unit. This volume can be given by multiplying the width of the transfer canal [input 18] by the elevation from the top of the ends [input 20] subtracted from the elevation where the vertical section starts [E_3] by length of the transfer canal [input 21] subtracted from the length of the transfer canal at the bottom [input 15] multiplied by one half (for a triangle) multiplied by 2 each unit.

$V_8 = W_5 * (E_3 - H_1) * (L_4 - L_6) * 0.5 * 2$

V_8 = Volume of the triangles formed by the angled section of wall, ft^3
W_5 = Width of the transfer canal, ft
E_3 = Elevation where the angled wall starts, ft
H_1 = Upper elevation of ends, ft
L_4 = Length of the section between the ends, per unit, ft
L_6 = Length of the transfer canal at the top elevation, per unit, ft

V_8 = 3 ft * [(62 ft + 8/12 ft) - (31 ft + 6/12 ft)] * [(48 ft + 4/12 ft) - (37 ft + 1/12 ft + 1/2/12 ft)] * 0.5 * 2

V_8 = 1,048 ft^3

<u>(d) Volume of the transfer canal in vertical section</u>

The volume of the transfer canal in the vertical section from elevation of the vertical section to the spill over point is determined by multiplying the width of the transfer canal [input 18] by the length of the transfer canal (times two) [input 21] by the elevation at the start of the vertical section [E_3] subtracted from the elevation of the spill over [E_1].

$V_9 = W_5 * L_6 * 2 * (E_1 - E_3)$

V_9 = Volume of the water to the spill elevation in the vertical region of transfer canal at top elevation, ft^3
W_5 = Width of the transfer canal, ft
L_6 = Length of the transfer canal at the top elevation, per unit, ft
E_1 = Elevation of the spill over point, ft
E_3 = Elevation where the angled wall starts, ft

V_9 = 3 ft * (37 ft + 1/12 ft + 1/2/12 ft) * 2 * [(64 ft + 7/12 ft + 11/16/12 ft) - (62 ft + 8/12 ft)]

V_9 = 440 ft^3

<u>(e) Volume of the transfer canal divider wall</u>

To determine the transfer canal divider wall volume to the spill point, the elevation at the bottom of the divider wall must be known. This is determined by subtracting the height of the divider wall [input 24] from the elevation of the top of the spent fuel pool [input 7]

$E_5 = E_2 - D_3$

E_5 = Elevation of bottom of transfer canal divider wall, ft
E_2 = Elevation of the top of the SFP, ft
D_3= Length of divider wall, ft

$E_5 = 66 \text{ ft} - (35 \text{ ft} + 4/12 \text{ ft})$

$E_5 = 30 \text{ ft} - 8 \text{ in}$

The volume of the transfer canal divider wall can be determined by multiplying the width of the transfer canal [input 18] by the width of the divider [input 23] by elevation of transfer canal divider wall at the bottom [E_5] subtracted from the elevation of the spill over point [E_1].

$V_{10} = W_5 * L_7 * (E_1 - E_5)$

V_{10} = Volume of the transfer canal divider wall, ft^3
W_5 = Width of the transfer canal, ft
L_7 = Width of the transfer canal divider wall, ft
E_1 = elevation of the spill over point, ft
E_5 = Elevation of bottom of transfer canal divider wall, ft

$V_{10} = 3 \text{ ft} * 4 \text{ ft} * [(64 \text{ ft} + 7/12 \text{ ft} + 11/16/12 \text{ ft}) - (30 \text{ ft} + 8/12 \text{ ft})]$

$V_{10} = 408 \text{ ft}^3$

<u>(f) Volume of transfer canal from bottom to spill</u>

The volume of the transfer canal from the bottom to the spill over elevation is given by the sum of the previously determined volumes as:

$V_{11} = V_5 + V_6 + V_7 + V_8 + V_9 - V_{10}$

V_{11} = Volume to fill the transfer canal from bottom to spill, ft^3
V_5 = Volume in the ends of the transfer canal, ft^3
V_6 = Volume in the bottom of the transfer canal between the ends with same elevation, ft^3
V_7 = Volume in-between the angled wall sections , ft^3
V_8 = Volume of the triangles formed by the angled section of wall, ft^3

V_9 = Volume of the water to the spill elevation in the vertical region of transfer canal at top elevation, ft^3
V_{10} = Volume of the transfer canal divider wall, ft^3

$V_{11} = 419\ ft^3 + 1{,}982\ ft^3 + 6{,}942\ ft^3 + 1{,}048\ ft^3 + 440\ ft^3 - 408\ ft^3$

$V_{11} = 10{,}423\ ft^3$

Putting the volume in terms of gallons of water

$V_{11} = 10{,}423\ ft^3 * 7.4805\ gal/ft^3 = \underline{77{,}971\ gal}$

(c) Determine volume of water to fill the spent fuel pool and transfer canal from the spill over to the high level alarm

Next need to determine the volume of water that is required to fill the transfer canal and spent fuel pool from the spill over elevation to the high level alarm.

First need to calculate the volume of the transfer canal divider wall in that elevation. This is determined multiplying the width of the divider wall [input 23] by the width of the transfer canal [input 18] by the elevation of the spill over point [E_1] subtracted from the elevation of the high level alarm [input 26].

$V_{12} = W_5 * L_7 * (E_6 - E_1)$

V_{12} = Volume of the transfer canal divider wall from the spill over to high level elevations, ft^3
W_5 = Width of the transfer canal, ft
L_7 = Width of the transfer canal divider wall, ft
E_6 = Elevation of the high level alarm, ft
E_1 = Elevation of the spill over point, ft

$V_{12} = 3\ ft * 4\ ft * [(64\ ft + 10/12\ ft) - (64\ ft + 7/12\ ft + 11/16/12\ ft)]$

$V_{12} = 2\ ft^3$

The volume to fill the transfer canal can be determined by multiplying the length of the transfer canal [input 21] by the width of the transfer canal [input 18] by two for each unit by the elevation of the spill over point [E_1] subtracted from the elevation of the high level alarm [input 26]. From this the volume of the divider wall is subtracted [V_{12}]

$V_{13} = W_5 * L_6 * 2 * (E_6 - E_1) - V_{12}$

V_{13} = Volume of the water to fill the transfer canal from the spill elevation to the high level alarm, ft^3
W_5 = Width of the transfer canal, ft
L_6 = Length of the transfer canal at the top elevation, per unit, ft
E_6 = Elevation of the high level alarm, ft
E_1 = Elevation of the spill over point, ft
V_{12} = Volume of the transfer canal divider wall from the spill over to high level elevations, ft^3

$V_{13} = 3\ ft * (37\ ft + 1/12\ ft + 1/2/12\ ft) * 2 * [(64\ ft + 10/12\ ft) - (64\ ft + 7/12\ ft + 11/16/12\ ft)] - 2\ ft^3$

$V_{13} = 41\ ft^3$

The volume of the spent fuel pool to raise the level from the spill elevation to the high level alarm can be given by multiplying the length of the spent fuel pool [input 8] by the width of the spent fuel pool [input 9] by the subtracting the elevation of the spill point [E_1] from the high level alarm [input 26].

$V_{14} = L_0 * W_0 * (E_6 - E_1)$

V_{14} = Volume of the spent fuel pool to raise from spill elevation to the high level alarm, ft^3
L_0 = Length of the spent fuel pool, ft
W_0 = Width of spent fuel pool, ft
E_6 = Elevation of the high level alarm, ft
E_1 = Elevation of the spill over point, ft

$V_{14} = 72\ ft * (18\ ft + 4/12\ ft) * [(64\ ft + 10/12\ ft) - (64\ ft + 7/12\ ft + 11/16/12\ ft)]$

$V_{14} = 254\ ft^3$

The total volume to raise the level from the spill over point to the high level alarm is the sum of the volume to raise the transfer canal [V_{13}] plus the volume to raise the spent fuel pool [V_{14}].

$V_{15} = V_{13} + V_{14}$

V_{15} = Volume to raise the transfer canal and spent fuel pool to the high level alarm, ft^3
V_{13} = Volume of the water to fill the transfer canal from the spill elevation to the high level alarm, ft^3
V_{14} = Volume of the spent fuel pool to raise from spill elevation to the high level alarm, ft^3

$V_{15} = 41 \text{ ft}^3 + 254 \text{ ft}^3$

$V_{15} = 295 \text{ ft}^3$

Put this volume in terms of gallons:

$V_{15} = 295 \text{ ft}^3 * 7.4805 \text{ gal/ ft}^3 = \underline{2,207 \text{ gal}}$

<u>(d) Determine the volume to raise the transfer canal and spent fuel pool from the high level alarm to the top of structure elevation</u>

Need to determine the volume of water that is required to fill the transfer canal and spent fuel pool from the high level alarm to the top of the structure.

First need to calculated the volume of the transfer canal divider wall in that elevation. This is determined multiplying the width of the divider wall [input 23] by the width of the transfer canal [input 18] by the elevation of the high level alarm [input 26] subtracted from the elevation of the top of structure [input 7].

$V_{16} = W_5 * L_7 * (E_2 - E_6)$

V_{16} = Volume of the transfer canal divider wall from the high level alarm to the top of SFP structure, ft^3
W_5 = Width of the transfer canal, ft
L_7 = Width of the transfer canal divider wall, ft
E_2 = Elevation of the top of the spent fuel point (reference point), ft
E_6 = Elevation of the high level alarm, ft

$V_{16} = 3 \text{ ft} * 4 \text{ ft} * [66 \text{ ft} - (64 \text{ ft} + 10/12 \text{ ft})]$

$V_{16} = 14 \text{ ft}^3$

The volume to fill the transfer canal can be determined by multiplying the length of the transfer canal [input 21] by the width of the transfer canal [input 18] by two for each unit by the elevation of the high level alarm [input 26] subtracted from the elevation of the top of structure [input 7]. From this the volume of the divider wall is subtracted [V_{16}]

$V_{17} = W_5 * L_6 * 2 * (E_2 - E_6) - V_{16}$

V_{17} = Volume of the water to fill the transfer canal from the high level alarm to the top of SFP structure, ft^3
W_5 = Width of the transfer canal, ft

L_6 = Length of the transfer canal at the top elevation, per unit, ft
E_2 = Elevation of the top of the spent fuel point (reference point), ft
E_6 = Elevation of the high level alarm, ft
V_{16} = Volume of the transfer canal divider wall from the high level alarm to the top of SFP structure, ft^3

V_{17} = 3 ft * (37 ft + 1/12 ft + 1/2/12 ft) * 2 * [66 ft - (64 ft + 10/12 ft)] -14 ft^3

$V_{17} = 246\ ft^3$

The volume of the spent fuel pool to raise the level from the the high level alarm to the top of structure can be given by multiplying the length of the spent fuel pool [input 8] by the width of the spent fuel pool [input 9] by the subtracting the high level alarm [input 26] from the elevation of top of structure [input 7].

$V_{18} = L_0 * W_0 * (E_2 - E_6)$

V_{18} = Volume to raise spent fuel pool from high level alarm to spill over top of structure, ft^3
L_0 = Length of the spent fuel pool, ft
W_0 = Width of spent fuel pool, ft
E_2 = Elevation of the top of the spent fuel point (reference point), ft
E_6 = Elevation of the high level alarm, ft

V_{18} = 72 ft * (18 ft + 4/12 ft) * [66 ft - (64 ft + 10/12 ft)]

$V_{18} = 1540\ ft^3$

The volume to raise the transfer canal and spent fuel pool from the high level alarm to the top of structure is the sum of the volume to raise the transfer canal [V_{17}] plus the volume to raise the spent fuel pool [V_{18}].

$V_{19} = V_{17} + V_{18}$

V_{19} = Volume to raise the spent fuel pool and transfer canal to the top of structure, ft^3
V_{17} = Volume of the water to fill the transfer canal from the high level alarm to the top of SFP structure, ft^3
V_{18} = volume to raise spent fuel pool from high level alarm to spill over top of structure, ft^3

$V_{19} = 246\ ft^3 + 1{,}540\ ft^3$

$V_{19} = 1{,}786\ ft^3$

Put this volume in terms of gallons:

$$V_{19} = 1,786 \ ft^3 * 7.4805 \ gal/ft^3 = \underline{13,359 \ gal}$$

CONCLUSIONS

1.	Initial volume of water is SFP	236,406 gal
2.	Volume to raise SFP to low level to spill	19,252 gal
3.	Volume to fill transfer canal to spill	77,971 gal
4.	Volume to fill SFP and transfer canal from spill to high level alarm	2,207 gal
5.	Volume to fill SFP and transfer canal from high level to top of structure	13,359 gal

DOCUMENT UPDATES

None

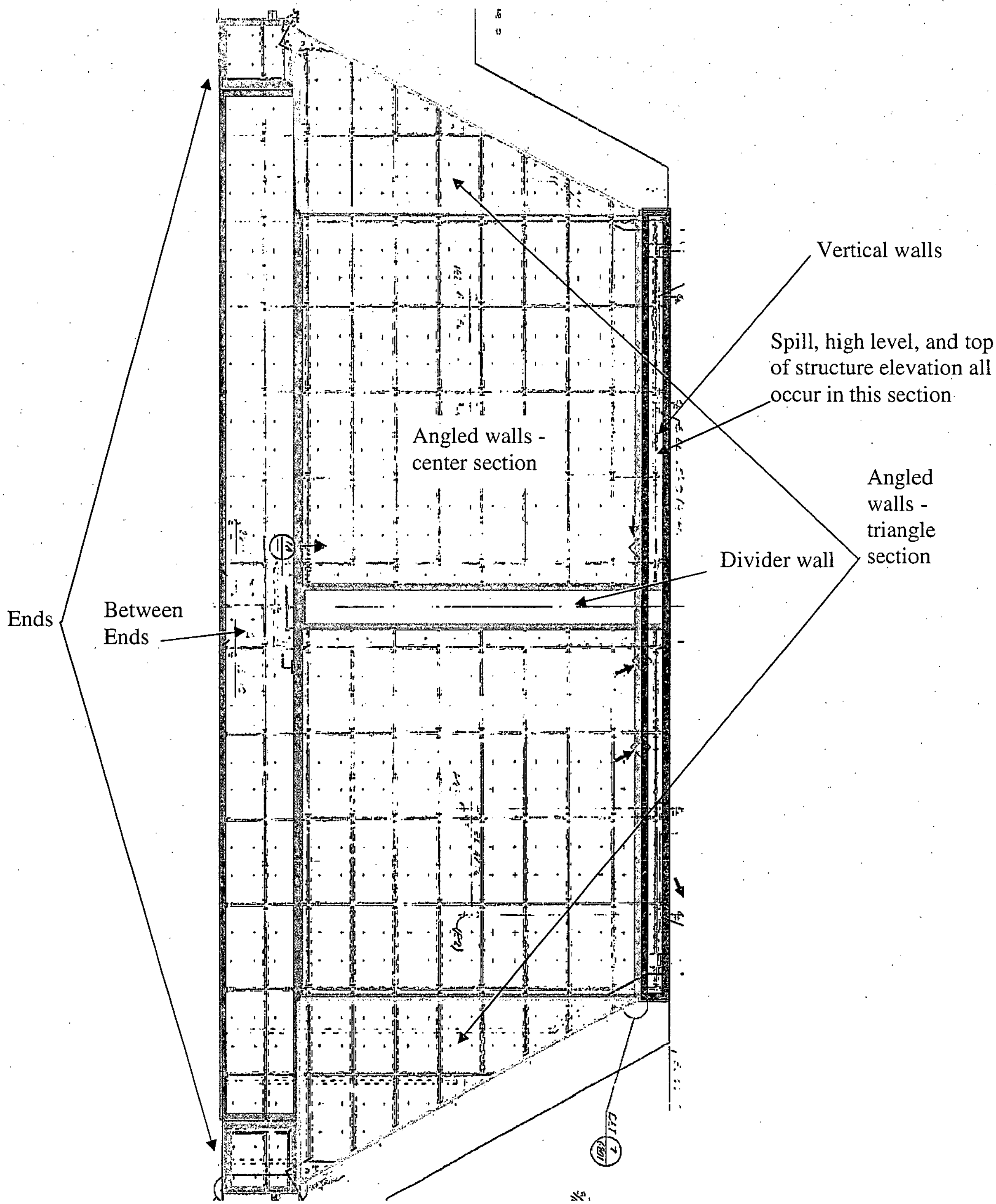

Figure 1 - Transfer Canal Volume Sections

Appendix B - Dilution Determination

PURPOSE

The purpose of Appendix B is to determine the volume of water required to dilute the spent fuel pool to the analysis limit.

BACKGROUND

The new spent fuel pool criticality analysis WCAP-16541-P, Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis, uses the Westinghouse methodology in WCAP-14416-P. WCAP-14416-P credits soluble boron to mitigate potential accidents. In its approval of WCAP-14416-P, the NRC requires that licensees identify potential dilution events. The dilution flow rates are available from plant data. The volume required to dilute the spent fuel pool from the technical specification limit to the limit determined in the criticality analysis must be determined. From this volume and plant data, the dilution time may be calculated.

ASSUMPTIONS

Validated Assumptions

1. It is assumed that the doors to the transfer canal are closed.
Basis: It is conservative to assume the doors are closed. This decreases the volume of water available to be diluted and thus decreases the required dilution volume.

2. It is assumed that the transfer canal is drained.
Basis: The transfer canal is drained for maintenance prior to each refueling outage. The high level alarm is set so that it is above the spill over point from the spent fuel pool to the transfer canal. This will increase the amount of time from the beginning of the dilution event to the time the high-level alarm is reached.

3. It is assumed that the water level in the SFP is at the low level alarm.
Basis: It is conservative to assume the water is at the low level alarm. This decreases the volume of water to be diluted and thus decreases the required dilution volume. This is also the condition when plant operations would initiate normal makeup to raise the spent fuel pool level.

3. It is assumed that the pool fills with unborated water from the low level to the spill over point through the new fuel elevator cable hole cut in the divider wall. The transfer canal will then fill to the spill over point. Both the transfer canal and spent fuel pool will then fill together until the top of structure is reached, when spilling over to the floor occurs.
Basis: This is the most likely scenario that does not involve some type of malicious action on the part of a plant employee.

4. It is assumed that the spent fuel pool cooling system is in operation.

Basis: Normal plant operation has the cooling system in operation at all times. Consideration of two independent accidents is not required.

5. It is assumed that there is uniform mixing in the SFP.
Basis: It was previously assumed that the SFP cooling system is in operation, which promotes mixing. In addition the decay heat from the fuel promotes natural convection cooling and water movement aiding in mixing.

6. It is assumed the high level alarm does not annunciate until transfer canal is filled.
Basis: Plant operating experience shows that during filling of the spent fuel pool from the CVCS hold up tanks, with a sufficiently high flow rate the high level alarm will temporarily annunciate even though water is spilling over to the transfer canal. By assuming the high level alarm is not reached, a greater dilution volume will occur before the control room is notified by the alarm.

7. It is assumed that when the spent fuel pool spills over into the transfer canal it is similar to a bleed-and-feed operation and there is constant volume in the spent fuel pool. The same is true when the spent fuel pool and transfer canal flood over the structure.
Basis: As unborated water is pushed in through the dilution flow path, it will push other water out of the spent fuel pool. Because we are assuming uniform mixing, as water is introduced into the spent fuel pool, it is uniformly mixed and then flows out. This model most closely mimics the dilution that would be occurring in the spent fuel pool.

8. It is assumed that there is no mixing between the spent fuel pool and the transfer canal when both are filling to the top of structure.
Basis: The level will rise in the spent fuel pool and the transfer canal. Water introduced into the spent fuel pool will mix with the borated water. Some of this mixture enters the transfer canal causing the total level to rise (spent fuel pool and transfer canal.) The volume of water added is the total amount to fill both the spent fuel pool and the transfer canal, but only the spent fuel pool is considered to be diluted. This is conservative because some mixing with the transfer canal would occur.

9. It is assumed that the density of the spent fuel pool water and the additional unborated water is at the same temperature and density.
Basis: Water is considered in incompressible fluid and the density changes due to temperature differences would be small.

<u>Un-validated Assumptions</u>

None

REFERENCES (verified current as of 6/30/2006).

1. Technical Specification 3.7.11, Fuel Storage Pool Boron Concentration.
2. NF-NMC-06-5, Point Beach Spent Fuel Pool Criticality Analysis, 1/27/06.
3. Appendix A, Dilution Volumes, EC9694, 12/6/2006.
4. WCAP-16541-P, Point Beach Units 1 and 2 Spent Fuel Pool Criticality Analysis, 2/1/2006.
5. PBNP Calculation 97-0108, Constants Used to Compile the Unit 1 and Unit 2 Blender Tables for 3.75% Boric Acid, 4/18/2000.

INPUTS

1.	Initial spent fuel pool boron concentration	C_0	2100 ppm	Ref. 1
2.	End point boron concentration	C_f	805 ppm	Ref. 2
3.	Initial spent fuel pool volume	V_1	236,406 gal	Ref. 3
4.	Volume to fill spent fuel pool to spill point	V_2	19,252 gal	Ref. 3
5.	Volume to fill transfer canal to spill point	V_{11}	77,971 gal	Ref. 3
6.	Volume to fill transfer canal and spent fuel pool from spill point to high alarm	V_{15}	2,207 gal	Ref. 3
7.	Volume to fill transfer canal and spent fuel pool from high alarm to spill over	V_{19}	13,359 gal	Ref. 3
8.	Volume to fill spent fuel pool from spill point to high alarm	V_{14}	254 ft^3	Ref. 3
9.	Volume to fill spent fuel pool from high alarm to top of structure	V_{18}	1,540 ft^3	Ref. 3
10.	Concentration reduction for feed and bleed.		Equation 1	Ref. 5

ATTACHMENTS

None.

ACCEPTANCE CRITERIA

None. The output of this numerical manipulation is a determination of the volume of water necessary to dilute the SFP from the tech spec value to the limit in the criticality analysis. This volume will be used as part of dilution analysis for the spent fuel pool.

METHODOLOGY AND CALCULATION

In order to determine the final volume of water that must be added to dilute the spent fuel pool from the technical specification limit of 2100 ppm to the analysis limit of 805 ppm, the analysis must be done in several steps. The first step is to calculate how much the concentration decreases when water is added to fill the spent fuel pool to the spill over point. Next the concentration decrease from a feed-and-bleed addition to fill the transfer canal is calculated. Then the concentration decrease from filling the spent fuel pool and the transfer canal to the point of overflow is calculated. Finally, the volume to decrease the boron concentration to the analysis limit is determined.

<u>Determine the change in boron concentration from low level alarm to spill over point:</u>

The initial addition of water that raises the spent fuel pool volume to the spill over point will dilute the spent fuel pool. The boron concentration at the end of this operation must be calculated to determine the initial boron concentration of the spent fuel pool for the transfer canal fill. The general equation for determining the change in concentration when mixing liquids is:

$$C_{final} = (C_{fluid1} * V_{fluid1} + C_{fluid2} * V_{fluid2}) / (V_{fluid1} + V_{fluid2})$$

where C is the concentration and V is the volume. For this evaluation, fluid 1 is the spent fuel pool water and fluid 2 is the unborated water that is added to the spent fuel pool. The concentration of fluid 1 is the concentration of the spent fuel pool before the dilution starts. Since fluid 2 is unborated water, it has zero concentration. The equation becomes:

$$C_{final} = C_{initial} * V_{SFP} / (V_{SFP} + V_{added})$$

The change in concentration is found by multiplying the initial concentration [input 1] by the initial spent fuel pool volume at the low level alarm [input 3] and dividing by the sum of the initial spent fuel pool volume [input 3] and the water added to raise the level to the spill point [input 4]. The change in concentration can be determined as follows:

$$C_1 = C_o * V_1 / (V_1 + V_2)$$

C_1 = The concentration of the spent fuel pool after the fill to spill over point, ppm
C_o = Initial spent fuel pool boron concentration, ppm
V_1 = Volume of water in SFP at low level alarm, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal

$$C_1 = 2100 \text{ ppm} * 236{,}406 \text{ gal} / (236{,}406 \text{ gal} + 19{,}252 \text{ gal})$$

$$C_1 = 1{,}942 \text{ ppm}$$

<u>Determine the concentration during spill over into the transfer canal</u>

In order to determine the change in boron concentration during a bleed-and-feed operation during the filling of the transfer canal, the following equation from Reference 5 is used:

$(C_{B\ final} - C_{B\ RMU/Boric\ Acid}) / (C_{B\ initial} - C_{B\ RMU/Boric\ Acid}) = \exp [(-Vol_{RMU/Boric\ Acid}\ \rho_{RMU/Boric\ Acid}) / (Vol_{Prim}\ \rho_{Prim} + Vol_{CVCS}\ \rho_{CVCS})]$ (Equation 1)

Because it is assumed the spent fuel pool is filled with unborated water, the term $C_{B\ RMU/Boric\ Acid}$ is zero. It is also assumed that the density of the spent fuel pool and any water added is the same, removing the density terms from the equation. In the above equation, the volume of the primary system and the CVCS were the initial volume. The volume of RMU/Boric acid was for the water added to the primary system. The only concentration being considered is boric acid and the subscript B term may be dropped as well. Thus the equation above reduces to:

$C_{final} / C_{initial} = \exp (-V_{added}/V_{initial})$

or

$C_{final} = C_{initial} * \exp (-V_{added}/V_{initial})$

This can also be re-written in terms of the volume of water added as:

$V_{added} = V_{initial} * \ln (C_{initial}/C_{final})$

For this case, the final concentration of the spent fuel pool after bleed-and-feed to the spent fuel pool is determined by raising the log of the starting boron concentration [C_1] subtracted from the volume of water added [input 5] divided by the starting water volume [input 3 plus input 4] .

$C_2 = C_1 * \exp\text{^}[- V_{11} / (V_1 + V_2)]$

C_2 = The concentration of the spent fuel pool after filling the transfer canal, ppm
V_{11} = Volume to fill the transfer canal from bottom to spill, gal
V_1 = Volume of water in SFP at low level alarm, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal
C_1 = The concentration of the spent fuel pool after the fill to spill over point, ppm

C_2 = 1,942 ppm * exp[- 77,971 gal / (236,406 gal + 19,252 gal)]

C_2 = 1,431 ppm

<u>Determine the change in boron concentration during fill from spill over to high level alarm</u>

After the transfer canal is filled, water will continue to be added and the spent fuel pool level will rise to the high level alarm. This will continue to dilute the spent fuel pool (see assumption 8.)

The change in concentration is found by multiplying the concentration after filling the transfer canal [C_2] by the spent fuel pool volume at the spill over point [input 3 plus input 4] and dividing by the sum of the spent fuel pool volume at the spill over point [input 3 plus input 4] and the water added to raise the level to the high level alarm [input 6]. The change in concentration can be determined as follows:

$$C_3 = C_2 * (V_1 + V_2) / (V_1 + V_2 + V_{15})$$

C_3 = The concentration of the spent fuel pool after filling spent fuel pool and transfer canal to high level alarm, ppm
C_2 = The concentration of the spent fuel pool after filling the transfer canal, ppm
V_1 = Volume of water in SFP at low level alarm, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal
V_{15} = Volume to raise the transfer canal and spent fuel pool to the high level alarm, gal

C_3 = 1,431 ppm * (236,406 gal + 19,252 gal) / ((236,406 gal + 19,252 gal + 2,207 gal)

C_3 = 1,419 ppm

<u>Determine the change in boron concentration during fill from high level alarm to overflow over</u>

After the transfer canal and spent fuel pool are filled to the high level alarm, water will continue to be added and the spent fuel pool level will rise to the top of the structure. This will continue to dilute the spent fuel pool (see assumption 8.) The change in concentration is found by multiplying the concentration after filling the transfer canal and spent fuel pool to the high level alarm [C_3] by the spent fuel pool volume at the high level alarm [input 3 plus input 4 plus input 6] and dividing by the sum of the spent fuel pool volume at the high level alarm [input 3 plus input 4 plus input 6] and the water added to raise the level to the top of the structure [input 7]. The change in concentration can be determined as follows:

$$C_4 = C_3 * (V_1 + V_2 + V_{15}) / (V_1 + V_2 + V_{15} + V_{19})$$

C_4 = The concentration of the spent fuel pool after filling to top of structure, ppm
C_3 = The concentration of the spent fuel pool after filling spent fuel pool and transfer canal to high level alarm, ppm
V_1 = Volume of water in SFP at low level alarm, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal
V_{15} = Volume to raise the transfer canal and spent fuel pool to the high level alarm, gal
V_{19} = Volume to raise the spent fuel pool and transfer canal to the top of structure, gal

C_4 = 1,419 ppm * (236,406 gal + 19,252 gal + 2,207 gal) / ((236,406 gal + 19,252 gal + 2,207 gal + 13,359 gal)

C_4 = 1,349 ppm

Determine the volume to dilute the spent fuel pool to the limit

Now the final dilution volume can be calculated using the concentration of the spent fuel pool at the top of structure to the analysis limit. This is done by using the previously determined approach for a bleed-and-feed dilution of the spent fuel pool. This can be found by taking the volume to raise the spent fuel pool to the stop of structure [input 3 plus input 4 plus input 8 plus input 9] (transfer canal volume is excluded because only the spent fuel pool is bleed-and-feed) multiplied by the log of the spent fuel pool concentration at the top structure divided be the final spent fuel pool boron concentration.

$$V_{20} = (V_1 + V_2 + V_{14} + V_{18}) * \ln (C_4 / C_f)$$

V_{20} = Volume to dilute the spent fuel pool to analysis limit, gal
C_f = End point boron concentration, ppm
C_4 = The concentration of the spent fuel pool after filling to top of structure, ppm
V_1 = Volume of water in SFP at low level alarm, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal
V_{14} = Volume to raise the spent fuel pool to the high level alarm, ft^3
V_{18} = Volume to raise the spent fuel pool to the top of structure, ft^3

V_{20} = (236,406 gal + 19,252 gal + 254 ft^3 * 7.4805 gal/ ft^3 + 1,540 ft^3 * 7.4805 gal/ ft^3) * ln (1,349 ppm / 805 ppm)

V_{20} = 138,973 gal

Determine the total dilution volume added:

Finally, the total dilution volume is the sum of the volumes to raise the spent fuel pool to the spill over point [input 4] plus the volume to fill the transfer canal to the spill point [input 5] plus the volume to raise the spent fuel pool and transfer canal to the high level alarm [input 6] plus the volume to raise the transfer canal and spent fuel pool to the top of structure [input 7] plus the volume to dilute the spent fuel pool to its final concentration [V_{20}]

$$V_{21} = V_2 + V_{11} + V_{15} + V_{19} + V_{20}$$

V_{21} = Total dilution volume for spent fuel pool, gal
V_{11} = Volume to fill the transfer canal from bottom to spill, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal

V_{15} = Volume to raise the transfer canal and spent fuel pool to the high level alarm, gal
V_{19} = Volume to raise the spent fuel pool and transfer canal to the top of structure, gal
V_{20} = Volume to dilute the spent fuel pool to analysis limit, gal

V_{total} = 19,252 gal + 77,971 gal + 2,207 gal + 13,359 gal + 122,471 gal = 251,763 gal

The volume to raise the SFP to the high level alarm is given by adding the volume to raise the SFP to the spill over elevation [input 4] to the volume to fill the transfer canal [input 5] to the volume to raise the SFP and transfer canal to the high level alarm [input 6].

$$V_{22} = V_2 + V_{11} + V_{15}$$

V_{22} = Total volume to raise SFP to high level alarm, gal
V_2 = Volume of water to fill SFP from low level alarm to spill over point, gal
V_{11} = Volume to fill the transfer canal from bottom to spill, gal
V_{15} = Volume to raise the transfer canal and spent fuel pool to the high level alarm, gal

V_{22} = 19,252 gal + 77,971 gal + 2,207 gal = 99,430 gal

Determine the time required to dilute the SFP:

The time to dilute the SFP is then dependent on the source flowrate. This can be assessed in the rest of the work. However, a short table is provided for self-checking.

Rate (gpm)	Time to High Level (minutes)	Time to Min Boron (hours)
60	1,657	70
100	994	42
120	829	35
400	249	10
500	199	8

4. CONCLUSION

The total volume of unborated water required to dilute the SFP from the minimum technical specification value to the minimum value required by the criticality analysis is 251,763 gallons.

5. DOCUMENT UPDATES

No documents updates required.

Appendix C - Hidden Text References

This section contains the references to the hidden text comments in the evaluation portion. It is meant to provide an easy means of reference for information included in the report section.

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36. WSC D96G0901, Rev. Osmosis Water Treatment Sys P&ID, Rev. 1034
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ENCLOSURE 6

WCAP-16541-NP (NON-PROPRIETARY)

POINT BEACH UNITS 1 AND 2
SPENT FUEL POOL CRITICALITY ANALYSIS
DATED FEBRUARY 2006

LICENSE AMENDMENT REQUEST 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

(114 pages follow)

WCAP-16541-NP

Point Beach Units 1 and 2
Spent Fuel Pool Criticality Analysis

February 2006

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Business Development

"Official Record Electronically Approved in EDMS"

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1.0 Introduction

1.1 Objective

This report presents the results of criticality analyses for the Point Beach spent fuel pool racks with credit for burnup, integral fuel burnable absorber (IFBA), ^{241}Pu decay and soluble boron, where applicable. The primary objectives of this calculation are as follows:

1. To determine the fuel assembly burnup versus initial enrichment limits required for safe storage of fuel assemblies in the "All-Cell," "1-out-of-4 5.0 w/o Fresh with no IFBA," and "1-out-of-4 4.0 w/o Fresh with IFBA" storage configurations with credit for 5, 10, 15, and 20 years of ^{241}Pu decay.
2. To determine the number of IFBA pins versus initial enrichment limits required for safe storage of fuel assemblies in the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration. [ ]^{a,c}
3. To determine the assembly loading requirements at the interface between storage configurations.
4. To determine the amount of soluble boron required to maintain k_{eff} less than or equal to 0.95 in the spent fuel pools, including all biases and uncertainties, assuming the most limiting plausible reactivity accident.

The methodology used in this analysis for soluble boron credit is analogous to that of Reference 1 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 2. Reference 1 was reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC). [ ]^{a,c}

1.2 Design Criteria

The design criteria are consistent with General Design Criterion (GDC) 62, Reference 4, and NRC guidance given in Reference 5. Section 1.3 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria follows.

1. Determine the fresh and spent fuel storage configurations using no soluble boron conditions such that the 95/95 upper tolerance limit value of k_{eff}, including applicable biases and uncertainties, is less than 0.995. [ ]^{a,c} Note that the actual NRC k_{eff} limit for this condition is unity. Therefore, an additional margin of 0.005 Δk_{eff} units is included in the analysis results.
2. Determine the amount (ppm) of soluble boron necessary to reduce the k_{eff} value of all storage configurations by at least 0.05 Δk_{eff} units. [ ]^{a,c} As an example, storage configurations which contain depleted fuel assemblies (and represented by depleted isotopics) are less reactivity-sensitive to changes in soluble boron concentration than a fuel assembly represented by zero burnup and relatively low initial fuel enrichment.

3. Determine the amount of soluble boron necessary to compensate for 5% of the maximum burnup credited in any storage configuration. In addition, determine the amount of soluble boron necessary to account for a reactivity depletion uncertainty of 1.0% Δk_{eff} per 30,000 MWD/MTU of credited fuel burnup. [

]a, c

4. Determine the largest increase in reactivity caused by postulated accidents and the corresponding amount of soluble boron needed to offset this reactivity increase.

An alternative form of expressing the soluble boron requirements is given in Reference 2. The final soluble boron credit (SBC) requirement is determined from the following summation.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

Where,

SBC_{TOTAL} = total soluble boron credit requirement (ppm)

$SBC_{95/95}$ = soluble boron requirement for 95/95 k_{eff} less than or equal to 0.95 (ppm)

SBC_{RE} = soluble boron required to account for burnup and reactivity uncertainties (ppm)

SBC_{PA} = soluble boron required to offset accident conditions (ppm).

For purposes of the analyses, minimum burnup limits established for fuel assemblies to be stored in the storage configurations racks include burnup credit established in a manner that takes into account approximations to the operating history of the fuel assemblies. [

]a,c

1.3 Design Approach

The soluble boron credit methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and to eliminate the need to credit any of the degraded Boraflex.

The square storage cell pitch modeled for fuel assembly storage configurations is 9.938 inches. No credit is taken for Boraflex in any of the storage configurations.

The fuel assembly types used for all the analyses are the Westinghouse 14x14 Standard and OFA designs. The most reactive spent fuel pool temperature (with full moderator density of 1 g/cc) is used for each fuel assembly storage configuration such that the analysis results are valid over the nominal spent fuel temperature range (50° to 180°F).

The reactivity characteristics of the storage racks were evaluated using infinite lattice analyses; this environment was used in the evaluation of the burnup limits versus initial enrichment as well as the evaluation of physical tolerances and uncertainties. [

]a, c

1.4 Methodology

This section describes the methodology used to assure the criticality safety of the Point Beach spent fuel pool and to define limits placed on fresh and depleted fuel assembly storage configurations. The analysis methodology employs: (1) SCALE-PC, a personal computer version of the SCALE-4.4a code system with the updated SCALE-4.4a version of the 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross section library, and (2) the two-dimensional Discrete Integral Transport (DIT) code (Reference 8) with an Evaluated Nuclear Data File, Version 6 (ENDF/B-VI) neutron cross section library.

SCALE-PC was used for calculations involving infinite arrays for all the storage configurations in the spent fuel pool. [

]$^{a, c}$

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. All references to KENO in this Calculation Note refer to the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

1.4.1 SCALE-PC

The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on personal computers.

1.4.2 Validation of SCALE-PC

Validation of SCALE-PC for purposes of fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs: the Babcock & Wilcox (B&W) experiments in support of Close Proximity Storage of Power Reactor Fuel (Reference 9) and the Pacific Northwest Laboratory (PNL) Program in support of the design of Fuel Shipping and Storage Configurations. References 10 and 11, as well as several of the relevant thermal experiment evaluations in Reference 12 were found to be useful in updating pertinent experimental data for the PNL experiments.

The validation of SCALE-PC was limited to the 44-group library provided with the SCALE-PC version 4.4a package. The 238-group library, which is utilized for the off-nominal temperature cases, was further validated by comparing the results from identical cases performed with the 44-group library and confirming that the results agreed within the statistical uncertainty.

Nineteen experimental configurations were selected from the B&W experimental program; these consisted of the following experimental cores: Core X, the seven measured configurations of Core XI, Cores XII through XXI, and Core XIIIA. These analyses used measured critical data, rather than the extrapolated configurations to a fixed critical water height reported in Reference 9, to avoid introducing possible biases or added uncertainties associated with the extrapolation techniques. In addition to the active fuel region of the core, the full environment of the latter region, including the dry fuel above the critical water height, was represented explicitly in the analyses.

The B&W group of experimental configurations used variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or Borated Aluminum (B/Al) plates of different boron contents in the water channels between rod clusters were measured. Table 1-1 summarizes the results of these analyses performed with both the 44-group and 238-group libraries.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blackness. As in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 1-2 summarizes the results of these analyses performed with both the 44-group and 238-group libraries.

The approach used for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 14. For a given KENO-calculated value of k_{eff} and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed by the equation below. By this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

$$k_{95/95} = k_{keno} + \Delta k_{bias} + M_{95/95}\left(\sigma_m^2 + \sigma_{KENO}^2\right)^{1/2}$$

Where,

k_{keno} is the KENO-calculated multiplication factor

Δk_{bias} is the mean calculational method bias

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses, and is obtained from the tables of Reference 15

σ_m^2 is the mean calculational method variance deduced from the validation analyses

σ_{KENO}^2 is the square of the KENO standard deviation

The equation for the mean calculational methods bias is as follows:

$$\Delta k_{bias} = \frac{1}{n}\sum_{i=1}^{n}(1 - k_i)$$

Where,

k_i is the i^{th} value of the multiplication factor for the validation lattices of interest

The equation for the mean calculational variance of the relevant validating multiplication factors is as follows:

$$\sigma_m^2 = \frac{n\sum_{1}^{n}\frac{(k_i - k_{ave})^2}{\sigma_i^2}}{(n-1)\sum_{1}^{n}\frac{1}{\sigma_i^2}} - \sigma_{ave}^2$$

where k_{ave} is given by the following equation:

$$k_{ave} = \frac{\sum_{1}^{n} \frac{k_i}{\sigma_i^2}}{\sum_{1}^{n} \frac{1}{\sigma_i^2}},$$

σ_{ave}^2 is given by the following equation:

$$\sigma_{ave}^2 = \frac{\sum_{1}^{n} \sigma_i^2 G_i}{\sum_{1}^{n} G_i},$$

G_i is the number of generations.

For purposes of this bias evaluation, the data points of Table 1-1 and Table 1-2 are pooled into a single group from the 44-group library calculations. With this approach, the mean calculational methods bias, Δk_{bias}, and the mean calculational variance, σ_m^2, calculated by the equations given above, were determined to be [ ]$^{a, c}$ respectively. The magnitude of $M_{95/95}$ is obtained from Reference 15 for the total number of pooled data points, 30.

The magnitude of $k_{95/95}$ is given by the following equation for SCALE 4.4a KENO analyses employing the 44-group ENDF/B-V neutron cross section library and for analyses where these experiments are a suitable basis for assessing the methods bias and calculational variance:

[ ]^{a,c}

Based on the above analyses, the mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier for the 44-group library were deduced as [ ]$^{a, c}$ respectively.

1.4.3 Application to Fuel Storage Pool Calculations

As noted above, the CSAS25 control module was used to execute the functional modules within SCALE-PC. The CSAS25 control module was used to analyze either infinite arrays of single or multiple storage cells [ ]^{a,c}. Standard material compositions were used in the SCALE-PC analyses consistent with the design input given in Section 2.0; these data are listed in. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS25 module using input consistent with the data in

Table 1-3. For burnt fuel representations, the fuel isotopics were derived from the DIT code as described below.

1.4.4 The DIT Code

The DIT code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells through the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the ENDF/B-VI. Cross sections have been collapsed into an 89-group structure that is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually 4 broad groups), for use in the assembly flux calculation. [

]$^{a, c}$

The DIT code and its cross section library are used in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For the purpose of spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each complete set of fuel isotopics is reduced to a smaller set of burnt fuel isotopics at specified time points after discharge. [

]$^{a, c}$

[

]$^{a, c}$

1.5 Assumptions

- The Westinghouse 14x14 Standard fuel was modeled as the design basis fuel assembly to conservatively represent all the depleted fuel assemblies and the Westinghouse 14x14 OFA fuel was modeled as the design basis fuel assembly to conservatively represent all fresh fuel assemblies residing in the storage configurations. Although Westinghouse V422+ fuel is the present fuel design for Point Beach, the Standard fuel is bounding. Standard fuel assembly is 0.75 inches longer than V422+ and uses Zirc-4 as the cladding material, which is less absorbent than Zirlo used by V422+.

- Fresh fuel assemblies were conservatively modeled with a UO_2 density of 10.686 g/cm^3 (97.5% of theoretical density). This translates into a pellet density equal 98.6% of theoretical density with a 1.1% dishing (void) fraction.

- All fuel assemblies, fresh and depleted, were conservatively modeled as containing solid right cylindrical pellets and uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs that incorporate lower enrichment blanket or annular pellets.

- All of the Boraflex poison material residing in the storage racks was conservatively omitted for this analysis and replaced by water. The stainless steel material encasing the Boraflex, however, was modeled.

- [

  ]^{a,c}

- The design basis limit for k_{eff} at the zero soluble boron condition was conservatively reduced from 1.0 to 0.995 for this analysis.

Table 1-1
Calculational Results for Cores X Through XXI of the B&W Close Proximity Experiments

a,b,c

[1] Entry indicates metal separating unit assemblies.
[2] Entry indicates spacing between unit assemblies in units of fuel rod pitch.

Table 1-2
Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping and Storage Configurations

a,b,c

Table 1-3
Standard Material Compositions Used in Criticality Analysis of the Point Beach Spent Fuel Storage Racks

Material	Element	Weight Fraction
Zircaloy[3], Density = 6.578 g/cm^3 @ 293.15 K	Zr	0.9824
	Sn	0.0145
	Fe	0.0021
	Cr	0.0010
Water	SCALE Standard Composition Library Density = 1.0 g/cm^3 @ 293.15 K	
Stainless Steel	SCALE Standard Composition Library Density = 7.94 g/cm^3 @ 293.15 K	
Fresh UO_2	Fraction of Theoretical Density = 0.975 Enrichment up to 5.0 w/o ^{235}U @ 293.15 K	
Regular Concrete	SCALE Standard Composition Library Density = 2.3 g/cm^3 @ 293.15 K	
	Element or Isotope	**Isotopics (atoms/barn/cm^2)**
[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
	[ ]^{a,c}	[ ]^{a,c}
	[ ]^{a,c}	[ ]^{a,c}

[3] Point Beach also uses Zirlo cladding; however, the fuel rod, guide tube, and instrumentation tube claddings are modeled with Zircaloy in this analysis. This is conservative with respect to the Westinghouse ZIRLO product, which is a zirconium alloy containing additional elements including niobium. Niobium has a small absorption cross section, which causes more neutron capture in the cladding regions resulting in a lower reactivity. Therefore, this analysis is conservative with respect to fuel assemblies containing ZIRLO cladding in fuel rods, guide tubes, and the instrumentation tube.

2.0 Design Input

This section provides a brief description of the Point Beach spent fuel storage racks with the objective of establishing a basis for the analytical models used in the criticality analyses described in Section 3.0.

2.1 Design Input from Point Beach

Design data related to the Point Beach were required to develop the KENO models.

2.2 Spent Fuel Pool Storage Configuration Description

Point Beach has a single pool divided into north and south halves which are connected through a divider wall. Each pool has an inside dimension of 220.0 inches in the west to east direction and 408.0 inches in the north to south direction. Either seven or eight rack modules, each with 90 to 110 cell locations, occupy the south and north pools, respectively. A cask area (114.61 inches x 117.33 inches) is located in the southwest corner and an elevator area (44.76 inches x 22.84 inches) is located in the southeast corner of the north pool. In the north pool, rack modules are located 4.0 inches from the north wall and 14.82 inches from the south wall, 7.56 inches from the west wall, and 2.88 inches from the east wall. In the south pool, rack modules are located 6.82 inches from the north wall and 12.00 inches from the south wall, 6.19 inches from the west wall, and 4.25 inches from the east wall. Figure 2-1 shows the spent fuel pool and the storage rack modules.

Table 2-1 summarizes the overall geometry data for the Point Beach spent fuel pool.

2.3 Individual Storage Cell Descriptions

Point Beach spent fuel pool storage cells are centered on a pitch of 9.938 + 0.093 / -0.01 inches. Each storage cell consists of an inner stainless steel canister, which has a nominal inside dimension of 8.25 + 0.083 / -0.0 inches wall thickness 0.093 ± 0.003-inches. Each Boraflex poison panel is held in place in an L-shaped shell inside the canister. The dimensions of the Boraflex poison panel are 8.0 + 0.0 / -0.3 inches in width by 0.11 + 0.000 / -0.03 inches in thickness. The sheathing panels are included as 0.021 ± 0.005 inch in thickness and are located at the outside surface of the nominal Boraflex poison panel position. Note that no credit is taken for the presence of the neutron absorbing, Boraflex material in the analysis. Table 2-2 and Figure 2-2 summarizes the storage cell dimensions used for the Point Beach analyses.

2.4 Failed Fuel Rod Storage Basket Description

Figure 2-3 shows a sketch of the failed fuel rod storage basket (FFRSB). The FFRSB is designed to accommodate individual spent and/or fresh fuel rods in a fixed array in the spent fuel pool. Forty-nine tubes are stored in the FFRSB in a 7x7 array. Nominal dimensions are not available for this design. The conservative and bounding modeling approach is discussed in Section 3.5.6.

Table 2-1
Spent Fuel Pool Dimensions
(All dimensions in inches)

Parameter	Value
Pool Length	408.0
Pool Width	220.0
Wall Thickness	24
Reflector	24

Table 2-2
Storage Cell Description
(All dimensions in inches)

Parameter	Dimension
Cell Pitch	9.938 +0.093/-0.01
Cell ID	8.250 +0.083 /-0.0
Cell Wall Thickness	0.093 ± 0.003
Cell Wall Material	SS-304
Absorber[4] Width	8.0 +0.0 /-0.03
Absorber Thickness[4]	0.11 +0.00/-0.03

[4] Boraflex is replaced with water

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a,b,c

Figure 2-1 Point Beach Spent Fuel Pool Showing Storage Rack Modules

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a,b,c

Figure 2-2 Point Beach Storage Cell

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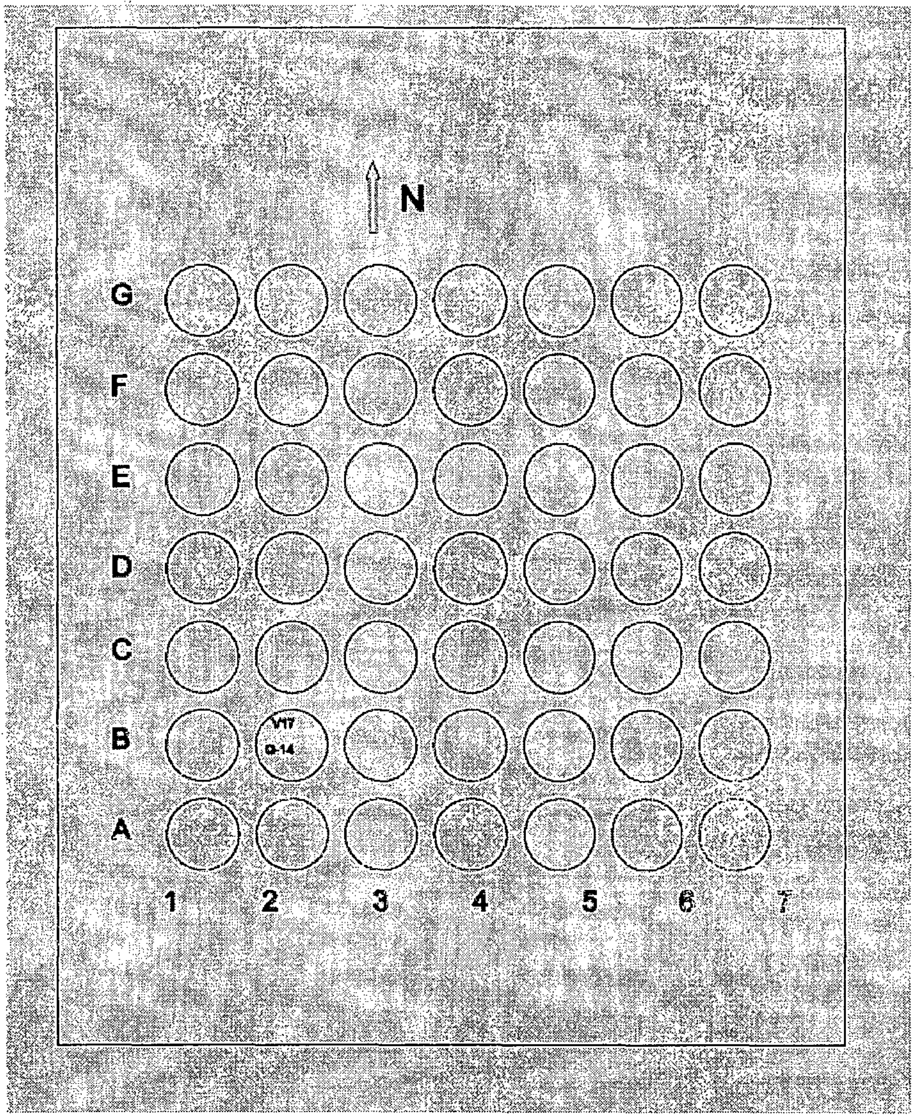

Figure 2-3 Failed Fuel Rod Storage Basket

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3.0 Analysis

3.1 KENO Models for the Spent Fuel Pool Storage Configurations

The Point Beach spent fuel storage racks employ three different fuel assembly storage configurations: "All-Cell," "1-out-of-4 5.0 w/o Fresh with no IFBA," and "1-out-of-4 4.0 w/o Fresh with IFBA." KENO models of these storage configurations are provided in the following sections. [
]^{a,c}

The fuel assemblies modeled by KENO represent the Westinghouse 14x14 Standard and OFA designs. Note that the enrichment of fresh fuel pellets is up to 5.0 w/o ^{235}U and the UO_2 density is 97.5% of theoretical density. The fuel pellets in a fuel rod are modeled as a fully enriched right solid cylinder that is 144 inches tall. This assumption conservatively bounds fuel rod designs that incorporate annular and or lower enrichment fuel pellets such as those used for axial blankets.

Each of the storage cell locations is modeled in KENO as a square cell with a pitch of 9.938 inches. The stainless steel canister, which controls the fuel assembly position within the array, is modeled with an inside dimension of 8.25 inches and is 0.093-inches thick. (Dimensions are taken from Table 2-2.) The Boraflex poison absorbers are modeled inside the stainless steel canisters with a dimension of 8.0 inch in width by 0.11 inch in thickness. The sheathing panels are included as 0.021 inch in thickness. The active fuel, storage rack box and sheathing heights are modeled in KENO as 144 inches tall. The geometry of the Boraflex poison is represented as water in the KENO model, thus no credit is taken for the presence of the neutron absorbing, Boraflex material.

Reflective boundary conditions are applied to the X and Y surfaces of 2x2 cell cells, thus simulating an infinitely repeating array. A 2-foot water reflector is modeled above and below the storage cell geometry. The pool water is simulated to be full density (1 g/cm^3) at room temperature (68°F). The top and bottom surfaces of the water reflector have reflected boundary conditions.

3.1.1 KENO Model for the "All-Cell" Storage Configuration

An "All-Cell" storage configuration is modeled in KENO as a repeating 2x2 array of storage cells that contain depleted standard fuel assemblies as shown below.

Depleted Fuel	Depleted Fuel
Depleted Fuel	Depleted Fuel

Note that the depleted fuel assemblies in the "All-Cell" storage configuration will be utilized to store fuel pins in the guide tubes. A KENO-produced plot of an "All-Cell" storage configuration is shown in Figure 3-1.

3.1.2 KENO Model for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

The "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration is modeled in KENO as a repeating 2x2 array with a fresh 5.0 w/o ^{235}U OFA fuel assembly occupying a storage cell location and depleted standard fuel assemblies occupying the remaining locations.

5.0 w/o Fresh OFA	Depleted Fuel
Depleted Fuel	Depleted Fuel

A KENO-produced plot of a single "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration is shown in Figure 3-2.

3.1.3 KENO Model for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

The "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration is modeled in KENO as a repeating 2x2 array with a fresh 4.0 w/o ^{235}U OFA fuel assembly occupying a storage cell location and depleted standard fuel assemblies occupying the remaining locations. Note that the fresh OFA fuel assembly with enrichments greater than 4.0 w/o contains IFBA rods.

4.0 w/o Fresh OFA	Depleted Fuel
Depleted Fuel	Depleted Fuel

A KENO-produced plot of a single "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration is shown in Figure 3-3. IFBA rods were modeled for this configuration using the layouts from

Figure 3-4. [

]$^{a, c}$

3.1.4 KENO Model for Entire Spent Fuel Pool

Point Beach spent fuel pool (for this analysis only the north pool has been considered) is modeled in KENO as a rectangular water cell that is 408.0 inches long and 220.0 inches wide. Seven rack modules, each with 90 to 110 cell locations, along with an empty cask area surrounded by 2-feet thick concrete walls compromise the pool model. The floor and walls of the spent fuel pool are modeled by surrounding the rectangular water cell with two feet of concrete on the bottom and sides. The pool dimensions are shown in

Table 2-1. The pool water was modeled at room temperature conditions, 68°F, and full density (1.0 g/cm^3). Figure 3-5 shows a KENO-produced plot of the spent fuel pool.

3.1.5 KENO Model for the Failed Fuel Rod Storage Basket

As mentioned in Section 2.4, the design dimensions for the Failed Fuel Rod Storage Basket (FFRSB) are not available. Therefore, a conservative and bounding approach is used for modeling the FFRSB. No credit is taken for any stainless steel structural material (tubes, grids, plates, etc) for the basket, so fresh 5.0 w/o ^{235}U fuel pins are placed on a uniform 7x7 array for simulation. This array is inserted in the All-Cell storage configuration by replacing one of the assemblies in that configuration. Figure 3-6 shows an FFRSB in the "All-Cell" configuration. Periodic boundary conditions are applied to the X and Y surfaces of the 2x2 array, thus simulating an infinitely repeating array. A 2-foot water reflector is modeled above and below the storage cell geometry. The pool water is simulated to be full density (1 g/cm^3) at room temperature (68°F). The top and bottom surfaces of the water reflector have reflected boundary conditions.

The fuel rods in FFRSB are modeled by KENO as the Westinghouse 14x14 OFA design with no burnable absorber. The UO_2 density is 97.5% of theoretical density for the fresh fuel at 5.0 w/o ^{235}U enrichment. Note that the fuel pellets in the fuel rods are modeled as a solid cylinder that is 144 inches tall. This assumption conservatively bounds fuel rod designs that incorporate annular and or lower enrichment fuel pellets such as those used for axial blankets.

3.2 Design Basis Fuel Assembly

Figure 3-7 shows the Westinghouse 14x14 fuel assembly with the standard assembly and OFA parameters given in Table 3-1. The Westinghouse standard fuel assembly design was modeled as the design basis fuel assembly to represent fresh and depleted fuel assemblies residing in all of the fuel assembly storage configurations.

The design basis fuel assemblies are modeled with the maximum enrichment over the active fuel length. The fresh fuel pellets in a fuel rod are modeled as solid right cylinder with a UO_2 density equal to 10.686 g/cm^3 (97.5 % of theoretical density). The depleted fuel pellets in a fuel rod are modeled as solid right cylinder with a UO_2 density equal to 10.412 g/cm^3 (95.0 % of theoretical density). Note that sensitivity calculations have shown that the effect of UO_2 density (in the range of 10.412 g/cc to 10.686 g/cc) on the resulting Assembly Burnup versus Initial Enrichment storage requirements is very small. Therefore, the resulting Assembly Burnup versus Initial Enrichment storage requirements generated in this analysis are valid for UO_2 densities up to 10.686 g/cm^3 (97.5 % of theoretical density). Lastly, it is noted that the above mentioned UO_2 density (10.686 g/cm^3) can only be achieved with pellets that have a dishing fraction equal to 1.1 % and a pellet UO_2 density equal to 98.6 % of theoretical density. In addition, no credit is taken for any natural or reduced enrichment pellets, even for the blanketed assemblies. This assumption results in conservative calculations of reactivity for all fuel assemblies stored in the racks. No credit is taken for any spacer grids or sleeves.

Figure 3-4 shows the IFBA patterns for [ ]^{a,b,c} IFBA rods in the Westinghouse 14x14 OFA used in this analysis. [

]^{a,c}

3.3 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology used in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the depleted fuel assemblies. This effect is important in the analysis of the spent fuel pool characteristics since the majority of spent fuel assemblies stored in the pool have a discharge burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative. Therefore, it is necessary to represent the burnt fuel assembly with a representative axial burnup profile.

[ ]^{a,c}

[

]^{a,c}

Input to this analysis is based on the limiting axial burnup profile data provided in the Department of Energy (DOE) Topical Report, as documented in Reference 18. The burnup profile in the DOE Topical Report is based on a database of 3,169 axial-burnup profiles for Pressurized Water Reactor (PWR) fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data. [

]^{a,c}

The DIT code was used to generate the isotopic concentrations for each segment of the axial burnup profile. Table 3-2 lists the fuel and moderator temperatures used in the spectral calculations for each node of the [ ]^{a,c} axial burnup models. The fuel temperatures for each axial zone are calculated based on a representative fuel temperature correlation while the moderator temperatures are based on a linear relationship with axial position. These node dependent moderator and fuel temperature data and power profile data were used in DIT to deplete the fuel to the desired burnup for each initial enrichment and each axial zone.

The values of assembly average burnups versus feed enrichment for which depleted fuel assemblies were simulated are presented in Table 3-3.

[

]^{a,c} The k_∞ and the isotopic number densities were then extracted for the KENO model development at these assembly conditions.

3.4 Tolerance / Uncertainty Calculations

Using the input described above, analytical models were developed to perform the quantitative evaluations necessary to demonstrate that the effective multiplication factor for the spent fuel pool is less than 0.995 with zero soluble boron present in the pool water. Applicable biases factored into this evaluation are: 1) the methodology bias deduced from the validation analyses of pertinent critical experiments, and 2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 50° to 180°F. Note that cases for nominal conditions were run with a full moderator density (1 g/cc), which actually corresponds to 40°F, which is less than the normal operating range and more conservative.

A second allowance is based on a 95/95 confidence level assessment of tolerances and uncertainties. The following are included in the summation of variances:

a. The 95/95 confidence level methods variance
b. The 95/95 confidence level calculational uncertainty
c. Fuel rod manufacturing tolerance
d. Storage rack fabrication tolerances
e. Tolerance due to positioning the fuel assembly in the storage cell
f. Burnup and IFBA manufacturing uncertainty

Items a. and b. are based on the calculational methods validation analyses described in subsection 1.4.2. For item c., the fuel rod manufacturing tolerance for the reference design fuel assembly is assumed to consist of an increase in fuel enrichment of 0.05w/o ^{235}U. An increase in UO_2 density is not assumed since all calculations are performed using 97.5% of theoretical density, which is the highest credible density for PWR fuel. The individual contributions of each change are combined by taking the square root of the sum of the squares of each component.

For item d., the following uncertainty components were evaluated. The inner stainless steel canister ID was increased from 8.25 inches to 8.333 inches and the thickness of the canister was decreased from 0.093 inches to 0.090 inches. The storage cell pitch was decreased from 9.938 inches to 9.928 inches.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells (item e.), all nominal calculations were carried out with fuel assemblies conservatively centered in the storage cells. Cases were run to investigate the effect of off-center position of the fuel assemblies for each of the fuel assembly storage configurations. These cases positioned the assemblies as close as possible in four adjacent storage cells.

For item f., [

]$^{a, c}$

Table 3-4 through Table 3-6 provide a summary of the KENO results used in the calculation of biases and uncertainties for the fuel assembly storage configurations.

3.5 No Soluble Boron 95/95 k_{eff} Calculational Results

The following subsections present the KENO-calculated multiplication factors for the Point Beach spent fuel pool storage configurations.

The KENO calculations reported in this section were performed at 68°F, with maximum water density of 1.0 g/cm^3, to maximize the array reactivity, and with an axially distributed burnup profile. The relative axial burnup profile used for these calculations is discussed in Section 3.3. The resulting k_{eff} data were then used to determine the burnup versus initial enrichment limits for a target k_{eff} value at zero soluble boron. The target value of k_{eff} was selected to be less than 0.995 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses.

The fuel assemblies modeled in these analyses are the Westinghouse 14x14 Standard and OFA fuel assembly designs.

3.5.1 "All-Cell" Storage Configuration

As described in subsection 3.1.1, the "All-Cell" storage configuration consists of a repeating 2x2 array of storage cells that contain depleted fuel assemblies.

The k_{eff} values were calculated for an infinite array of "All-Cell" storage configurations over a range of initial enrichment values up to 5.0 w/o ^{235}U and assembly average burnups up to 45,000 MWD/MTU. From Table 3-4, the sum of the biases and uncertainties is 0.02492 Δk_{eff} units. Therefore, the target k_{eff} value for the "All-Cell" storage configuration is 0.97008 (0.995-0.02492).

Table 3-7 lists the k_{eff} values for the "All-Cell" storage configuration versus initial enrichment and average burnups. The first entry in Table 3-7 lists the initial enrichment for no burnup. Based on the target k_{eff} value, the fresh enrichment for no burnup is 2.147 w/o ^{235}U. The derived burnup limits, for enrichments greater than 2.147 w/o ^{235}U, are based on the k_{eff} values for 3.0, 4.0, and 5.0 w/o ^{235}U. For each of these three enrichments, KENO calculations were performed at three assembly average burnup values with an axially distributed burnup profile. A second degree fit of the burnup versus k_{eff} data was then used to determine the burnup required to meet the target k_{eff} value of 0.97008.

The resulting burnup versus initial enrichment storage limits for 0, 5, 10, 15, and 20 years of decay time are provided in Table 3-8. The limiting burnups as a function of initial enrichment were fitted to a third degree polynomial for each of the decay period. These polynomials are given below Table 3-8 and will be used to determine the burnup as a function of initial enrichment for the "All-Cell" storage configuration. The data contained in Table 3-8 are plotted in Figure 4-7.

3.5.1.1 Storage of Fuel Pins in the All-Cell Configuration Guide Tubes

For storage of fuel pins in the guide tubes, the All-Cell storage configuration is considered. Table 3-9 lists the k_{eff} values for the All-Cell storage configuration with increasing number of depleted fuel pins occupying the guide tubes. Due to moderator displacement, the resulting k_{eff} values become less than the nominal k_{eff} value with increasing number of depleted fuel pins in the guide tubes. Table 3-10 shows that burnup versus initial enrichment storage limit for the All-Cell configuration with zero year decay time and no pins in the guide tubes is bounding for the fuel pins in the guide tubes. Therefore, it is concluded that any fuel pin stored in the guide tubes of an assembly in the All-Cell configuration shall meet the burnup requirements of the All-Cell configuration with zero year decay time and without any fuel pins in the guide tubes.

Fuel pins that do not meet the requirements of the All-Cell storage configuration can still be stored in the guide tubes given that pins that exceedingly meet the burnup requirement are also loaded in the guide tubes to offset the excess reactivity. To illustrate this point, calculations have been performed for the All-Cell configuration with eight of the guide tubes loaded with burned fuel that is 10,000 MWD/MTU lower than the average burnup in the configuration and the remaining eight of tubes loaded with burned fuel that is 10,000 MWD/MTU higher. Table 3-11 shows that the k_{eff} results are very similar to the case with uniform burnup and smaller than the nominal All-Cell case. This analysis demonstrates that *X* number of fuel pins that do not meet the burnup requirement and have a maximum of *Y* MWD/MTU less than the required amount can be stored in the guide tubes only with an equal or greater number of fuel pins (*X or greater*) with burnups that exceed the burnup requirement by *Y* MWD/MTU or higher.

3.5.2 "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

As described in subsection 3.1.2, the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration consists of a repeating 2x2 array, with a fresh OFA fuel assembly at 5.0 w/o enrichment in a storage cell location and depleted fuel assemblies in the remaining locations.

The k_{eff} values were calculated for an infinite array of "1-out-of-4 5.0 Fresh" storage configurations over a range of initial enrichment values up to 5.0 w/o ^{235}U and average burnups up to 55,000 MWD/MTU. From Table 3-5, the sum of the biases and uncertainties is 0.01909. Therefore, the target k_{eff} value for the "1-out-of-4 5.0 w/o at 15,000 MWD/MTU" storage configuration is 0.97591 (0.995-0.01909).

Table 3-12 lists the k_{eff} values for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration versus initial enrichment and average burnups with an axially distributed burnup profile. The first entry in Table 3-12 lists the initial enrichment for no burnup. Based on the target k_{eff} value, the interpolated enrichment for no burnup is 1.361 w/o ^{235}U. The derived burnup limits, for enrichments greater than 1.361 w/o ^{235}U, are based on the k_{eff} values for 3.0, 4.0, and 5.0 w/o ^{235}UFor each of these three enrichments, KENO calculations were performed at three assembly average burnup values for an axially distributed burnup profile. A second degree fit of the burnup versus k_{eff} data was then used to determine the burnup required to meet the target k_{eff} value of 0.97591. The resulting burnup versus initial enrichment storage limits for 0, 5, 10, 15, and 20 years of decay time are provided in Table 3-13. The limiting burnups as a function of initial enrichment were fitted to a third degree polynomial. These polynomials are given below Table 3-13 and will be used to determine the burnup as a function of initial enrichment for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration. The data in Table 3-13 are plotted in Figure 4-8.

3.5.3 "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

As described in subsection 3.1.3, the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration consists of a repeating 2x2 array, with a 4.0 w/o ^{235}U Fresh OFA fuel assembly in a storage cell location and depleted fuel assemblies in the remaining locations. For the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration, burnup limits have been evaluated for the depleted fuel assemblies and IFBA requirements have been determined for the fresh OFA fuel assembly with enrichments greater than 4.0 w/o ^{235}U.

3.5.3.1 Burnup Requirements of the Depleted Fuel Assemblies

The k_{eff} values were calculated for an infinite array of "1-out-of-4 4.0 w/o Fresh with IFBA" storage configurations over a range of initial enrichment values up to 5.0 w/o ^{235}U and average burnups up to 55,000 MWD/MTU. From Table 3-6, the sum of the biases and uncertainties is 0.02079. Therefore, the target k_{eff} value for the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration is 0.97421 (0.995-0.02079).

Table 3-14 lists the k_{eff} values for the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration versus initial enrichment and average burnups with an axially distributed burnup profile. The first entry in Table 3-14 lists the initial enrichment for no burnup. Based on the target k_{eff} value, the interpolated enrichment for no burnup is 1.627 w/o ^{235}U. The derived burnup limits, for enrichments greater than 1.627 w/o ^{235}U, are based on the k_{eff} values for 3.0, 4.0, and 5.0 w/o ^{235}U. For each of these three enrichments, KENO calculations were performed at three assembly average burnup values for an axially distributed burnup profile. A second degree fit of the burnup versus k_{eff} data was then used to determine the burnup required to meet the target k_{eff} value of 0.97421. The resulting burnup versus initial enrichment storage limits for 0, 5, 10, 15, and 20 years of decay time are provided in Table 3-15. The limiting burnups as a function of initial enrichment were fitted to a third degree polynomial. These polynomials are given below Table 3-15 and will be used to determine the burnup as a function of initial enrichment for the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration. The data in Table 3-15 are plotted in Figure 4-9.

3.5.3.2 IFBA Requirements for the Fresh Fuel Assembly

Table 3-16 and Table 3-17 list the k_{eff} values versus the number of IFBA pins contained in the fresh fuel assembly with 4.5 w/o and 5.0 w/o ^{235}U enrichments, respectively. For each fresh fuel enrichment and number of IFBA pins, k_{eff} was evaluated for different burnups of the depleted fuel assemblies with an initial enrichment of 5.0 w/o ^{235}U. [

]$^{a, c}$

From these tables, fuel assembly burnup versus k_{eff} data was fitted to a second degree polynomial using the target k_{eff} value of 0.97421. Note that this was the target k_{eff} value used to determine the burnup requirements for the depleted fuel assemblies. The resulting polynomials were then used to determine the required number of IFBA pins to meet the fuel assembly burnup requirement of 38,288 MWD/MTU with 5.0 w/o initial enrichment. [

]$^{a, c}$

Table 3-18 contains the required number of IFBA pins versus initial enrichment for the fresh fuel assemblies with enrichments greater than 4.0 w/o. The required number of IFBA pins as a function of initial enrichment was fitted to a second degree polynomial. This polynomial is given below Table 3-18 and will be used to determine the number of IFBA pins as a function of initial enrichment for the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration. The data in Table 3-18 are plotted in Figure 4-10. [

]$^{a, c}$

3.5.4 Interface Requirements

Table 3-19 shows the entire spent fuel pool k_{eff} results for the interface configurations in the point Beach storage racks. These interface configurations result in KENO-calculated multiplication factors that are less than the maximum of the infinite array multiplication factors for the involved storage configurations. As an example, the first analyzed interface involves the "1-out-of-4 5.0 w/o Fresh with no IFBA" configuration surrounded by the "All-Cell" storage configuration. From Table 3-7, the infinite array k_{eff} value for the "All-Cell" storage configuration is 0.96950 and from Table 3-12, the infinite array k_{eff} value for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration is 0.97558 . The maximum of these two values is 0.97558 . From Table 3-19, the multiplication factor for the interface configuration was then compared to this maximum value to verify that the interface meets the storage requirements.

The KENO models constructed to analyze the interface effects follow the description of the entire spent fuel pool from subsection 3.1.4. The assembly loading requirements at the interface between different storage configurations are provided in Table 3-20. As seen from this table and the Table 3-19 results, it is required that for storage configurations involving high and low reactivity assemblies (i.e., 1-out-of 4 configurations), the assemblies with lower reactivity must be placed at the interface. These interface requirements are depicted in Figure 4-4 to Figure 4-6. Note that it is acceptable to leave a storage cell empty.

3.5.5 Burnup Requirements for Intermediate Decay Time Points

For all the storage configurations in the Point Beach Spent Fuel pool crediting ^{241}Pu decay, burnup requirements for intermediate decay time points should be determined using at least a second order polynomial.

3.5.6 Failed Fuel Rod Storage Basket with 5.0 w/o ^{235}U Fuel

As described in subsection 3.1.5, FFRSB provides storage for a fixed 7x7 array of fuel rods. Calculations were performed for different fuel rod pitch values in the basket to determine the most reactive and bounding case. The pitch values ranged from *Pitch* = $O.D._{fuel}$ (fuel pins touching) to *Pitch* = *2.99357* inches (fuel pins uniformly spaced over the entire storage cell, maximizing the moderator to fuel ratio). Table 3-21 lists the k_{eff} values for the All-Cell storage configuration with one of the depleted fuel assemblies replaced with an FFRSB containing fresh 5.0 w/o ^{235}U OFA fuel rods with increasing pitch values. The calculations were performed at 68°F, with maximum water density of 1.0 g/cm^3 to maximize the array reactivity.

As seen from Table 3-21, the resulting k_{eff} values were less than the nominal k_{eff} value of the All-Cell storage configuration even with the largest pitch value for the pins inside the basket. Therefore, FRSBs filled with fresh fuel rods with a maximum enrichment of 5.0 w/o ^{235}U and no burnable absorbers can be stored in the All-Cell storage configuration.

3.5.7 Empty Cells

For all configurations at Point Beach, an empty cell is permitted in any location of the spent fuel pool to replace an assembly since the water cell will decouple the neutronic interaction between the spent fuel assemblies in the pool. Non-fissile material and debris canisters may be stored in empty cells of All-Cell storage configuration provided that the canister does not contain fissile materials.

3.5.8 Non-Fissile Equipment

Non fissile equipment, such as UT cleaning equipment is permitted on top of the fuel storage racks, as these equipments will not cause any increase in reactivity in the spent fuel pool.

3.6 Soluble Boron

The NRC Safety Evaluation Report (SER) for Westinghouse report WCAP-14416-P is given in Reference 2. Page 9 of the enclosure to Reference 2 defines the total soluble boron requirement as the sum of three quantities:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where,

SBC_{TOTAL} is the total soluble boron credit requirement (ppm),

$SBC_{95/95}$ is the soluble boron requirement for 95/95 k_{eff} less than or equal to 0.95 (ppm),

SBC_{RE} is the soluble boron required to account for burnup and reactivity uncertainties (ppm),

SBC_{PA} is the soluble boron required to offset accident conditions (ppm).

Each of these terms is discussed in the following subsections.

3.6.1 Soluble Boron Requirement to Maintain k_{eff} Less Than or Equal to 0.95

Table 3-22 contains the KENO-calculated k_{eff} values for the spent fuel pool from 0 to 600 ppm of soluble boron, in increments of 200 ppm. These KENO models assume that the pool is filled with the "All-Cell" storage configuration containing depleted fuel at 45,000 MWD/MTU with 5.0 w/o ^{235}U initial enrichment. The initial enrichment and burnup chosen to represent the storage configuration was based on minimizing the soluble boron worth. The soluble boron worth decreases as burnup increases. The reactivity worth, Δk_{eff}, of the soluble boron was determined by subtracting the k_{eff} value, for a given soluble boron concentration, from the k_{eff} value for zero soluble boron. The soluble boron concentration and reactivity worth data was then fitted to a third degree polynomial, which is shown on the bottom of Table 3-22. This polynomial was then used to determine the amount of soluble boron required to reduce k_{eff} by 0.05 Δk_{eff} units, which is 270.6 ppm.

3.6.2 Soluble Boron Requirement for Reactivity Uncertainties

The soluble boron credit, in units of ppm, required for reactivity uncertainties was determined by converting the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel burnup values to a soluble boron concentration, in units of ppm, necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly reactivity, is calculated by employing a depletion reactivity uncertainty of 0.010 Δk_{eff} units per 30,000 MWD/MTU of burnup (obtained from Reference 2) and multiplying by the maximum amount of burnup credited in a storage configuration. For this analysis, the maximum amount of burnup credited is 47,000 MWD/MTU for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration. Therefore, the depletion reactivity uncertainty is 0.015667 Δk_{eff}.

The uncertainty in absolute fuel burnup values is conservatively calculated as 5% of the maximum fuel burnup credited in a storage configuration analysis. The maximum fuel burnup credited in the various storage configurations, the 5% uncertainty in these burnup values, and the corresponding reactivity values are given in Table 3-23.

The maximum reactivity change associated with a 5% change in burnup is 0.007143 Δk_{eff} units and occurs for the "All-Cell" storage configuration.

The total of the uncertainties in fuel assembly reactivity and burnup effects is 0.022810 Δk_{eff}. By applying the polynomial at the bottom of Table 3-22, the soluble boron concentration (ppm) necessary to compensate for this reactivity is found to be of 118.9 ppm.

3.6.3 Soluble Boron Required to Mitigate Accidents

The soluble boron concentration, in units of ppm, to mitigate accidents is determined by first surveying all possible events that increase the k_{eff} value of the spent fuel pool. The accident event which produced the largest increase in spent fuel pool k_{eff} value is used to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered includes:

- Dropped fresh fuel assembly on top of the storage racks,
- Misloaded fresh fuel assembly into an incorrect storage rack location, or outside the racks,
- Spent fuel pool temperature greater than 180°F.

Several fuel mishandling events were simulated using the KENO model to assess the possible increase in the k_{eff} value of the spent fuel pool. The fuel mishandling events all assumed that a fresh Westinghouse 14x14 OFA fuel assembly enriched to 5.0 w/o ^{235}U (and no burnable poisons) was misloaded into a storage rack or in the cask area between the racks. These cases were simulated with the KENO model [.]^{a,c}

It is possible to drop a fresh fuel assembly on top of the spent fuel pool storage racks. In this case the physical separation between the fuel assemblies in the spent fuel pool storage racks and the assembly lying on top of the racks is sufficient to neutronically decouple the accident. In other words, dropping the fresh fuel assembly on top of the storage racks does not produce a positive reactivity increase. Note that the design of the spent fuel racks and fuel handling equipment is such that it precludes the insertion of a fuel assembly between the rack modules.

Calculations have shown that cool-down events produced less positive reactivity change compared to heat-up events. This is due to the fact that for cool-down events, only the temperature of the moderator is lowered since the moderator density is already at the maximum for nominal cases and the temperature effect alone is minimal or less compared to heat-up events. Therefore, results from heat-up events are reported here.

For the accident of a misloaded fresh fuel assembly, two scenarios were analyzed:

- A depleted fuel assembly was replaced with a fresh fuel assembly in a storage configuration;
- A fresh fuel assembly was placed in the cask area between the racks, face adjacent to either a depleted fuel or fresh fuel assembly of a storage configuration.

The above postulated accident scenarios involve the double contingency principle. This states that the analysis need not consider two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accident conditions, the presence of soluble boron in the spent fuel pool can be assumed as a realistic initial condition, since not assuming its presence would be a second unlikely event.

The k_{eff} values for the accident scenarios described above are summarized in Table 3-24. Note that the nominal cases were developed by filling up the pool with one of the storage configurations and then the accident scenarios, as described above, were applied. This process was repeated for all the storage configurations. Note also that both the nominal cases and the accident scenarios were simulated at zero ppm boron and using depleted fuel isotopics in the pool. As seen in Table 3-24, the accident event that produced the largest increase in the spent fuel pool k_{eff} value is the misloaded fresh 5.0 w/o ^{235}U enrichment fuel assembly in an incorrect storage rack location of the "All-Cell" configuration containing depleted fuel assemblies with 5.0 w/o ^{235}U initial enrichment at 45,000 MWD/MTU. As seen in Table 3-25, the required soluble boron concentration necessary to mitigate this and all less severe accident events was then calculated as 402.9 ppm using the Table 3-22 equation.

3.6.4 Total Soluble Boron Requirement

Soluble boron in the spent fuel pool coolant is used in this criticality safety analysis to offset the reactivity allowances for calculational uncertainties in modeling, storage rack fabrication tolerances, fuel assembly design tolerances, and postulated accidents.

The magnitude of each soluble boron requirement is as follows:

$SBC_{95/95}$ = 270.6 ppm

SBC_{RE} = 118.9 ppm

SBC_{PA} = 402.9 ppm

SBC_{TOTAL} = 792.4 ppm

Therefore, without considering an accident, the soluble boron (with 19.9% ^{10}B abundance) necessary to maintain k_{eff} less than or equal to 0.95 (including all biases and uncertainties) is:

$SBC_{95/95} + SBC_{RE}$ = 270.6 ppm + 118.9 ppm = 389.5 ppm.

The soluble boron concentration required for a ^{10}B atom percent equal to 19.6 (expected lowest pool value crediting ^{10}B depletion) is 395.5 ppm.

A total of 792.4 ppm of soluble boron (with 19.9% ^{10}B abundance) is required to maintain k_{eff} less than or equal to 0.95 (including all biases and uncertainties) and assuming the most limiting single accident. The soluble boron concentration required for a ^{10}B atom percent equal to 19.6 (expected lowest pool value crediting ^{10}B depletion) is 804.5 ppm. The recommended minimum boron level is 804.5 ppm and is sufficient to accommodate all the design requirements.

Table 3-1
Fuel Assembly Data Used in Criticality Analysis of the Point Beach Spent Fuel Storage Racks

[ ] a,b,c

Table 3-2

Relative Power and Fuel/ Moderator Temperatures for the [ ]^{a,c} Model

a,b,c

Table 3-3
Burnup and Initial Enrichment Combinations Used to Determine the Isotopic Number Densities

3.0 w/o ^{235}U (MWD/MTU)	4.0 w/o ^{235}U (MWD/MTU)	5.0 w/o ^{235}U (MWD/MTU)
0	0	0
5,000	15,000	25,000
15,000	25,000	35,000
25,000	35,000	45,000
35,000	45,000	55,000

Table 3-4
K_{eff} Values for the Tolerance/Uncertainty Cases for the "All-Cell" Storage Configuration

a,b,c

[ ]

a,b,c

[ ]

Table 3-5
K_{eff} Values for the Tolerance/Uncertainty Cases for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

a,b,c

a,b,c

Table 3-6
K_{eff} Values for the Tolerance/Uncertainty Cases for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

a,b,c

a,b,c

Table 3-7
K_{eff} Values versus Initial Enrichment and Assembly Burnup for the "All-Cell" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/ MTU)	k_{eff} Value				
		0 year decay	5 years decay	10 years decay	15 years decay	20 years decay
2.147	0	0.96950 ± 0.00035	0.96950 ± 0.00035	0.96950 ± 0.00035	0.96950 ± 0.00035	0.96950 ± 0.00035
3.000	5,000	0.99690 ± 0.00035	0.99477 ± 0.00034	0.99527 ± 0.00035	0.99451 ± 0.00037	0.99516 ± 0.00035
3.000	15,000	0.91069 ± 0.00034	0.90372 ± 0.00035	0.90071 ± 0.00034	0.89964 ± 0.00036	0.89858 ± 0.00033
3.000	25,000	0.84624 ± 0.00033	0.83651 ± 0.00033	0.83250 ± 0.00032	0.82798 ± 0.00032	0.82630 ± 0.00033
4.000	15,000	0.98417 ± 0.00035	0.97966 ± 0.00037	0.97595 ± 0.00033	0.97339 ± 0.00033	0.97128 ± 0.00035
4.000	25,000	0.91903 ± 0.00035	0.91256 ± 0.00036	0.90608 ± 0.00033	0.90142 ± 0.00036	0.89898 ± 0.00037
4.000	35,000	0.86703 ± 0.00031	0.85718 ± 0.00036	0.84939 ± 0.00037	0.84314 ± 0.00034	0.83875 ± 0.00035
5.000	25,000	0.97603 ± 0.00035	0.97054 ± 0.00035	0.96519 ± 0.00036	0.96205 ± 0.00036	0.95960 ± 0.00035
5.000	35,000	0.92513 ± 0.00037	0.91731 ± 0.00037	0.91098 ± 0.00036	0.90624 ± 0.00034	0.90143 ± 0.00036
5.000	45,000	0.88094 ± 0.00034	0.87047 ± 0.00036	0.86262 ± 0.00035	0.85623 ± 0.00034	0.85202 ± 0.00034

Table 3-8
Fuel Assembly Burnup versus Initial Enrichment for the "All-Cell" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)				
	0 yr decay	5 yr decay	10 yr decay	15 yr decay	20 yr decay
2.147	0	0	0	0	0
3.000	7,697	7,308	7,225	7,222	7,227
4.000	16,925	16,286	15,741	15,403	15,148
5.000	26,085	25,080	24,161	23,662	23,402

Note that the assembly burnups as a function of initial enrichment for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $- 49.58\, e^3 + 561.12\, e^2 + 7134.44\, e - 17417.89$

Assembly Burnup (5 yr decay) = $- 108.67\, e^3 + 1212.54\, e^2 + 4510.52\, e - 14201.98$

Assembly Burnup (10 yr decay) = $-24.62\, e^3 + 247.20\, e^2 + 7696.65\, e - 17424.68$

Assembly Burnup (15 yr decay) = $68.81\, e^3 - 786.56\, e^2 + 11140.74\, e - 20978.70$

Assembly Burnup (20 yr decay) = $163.70\, e^3 - 1797.81\, e^2 + 14448.84\, e - 24359.00$

Table 3-9
K_{eff} Values for the All-Cell Storage Configuration with Fuel Pins in the Guide Tubes

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/ MTU)	k_{eff} Value			
		0 pins in the tubes	4 pins in the tubes	9 pins in the tubes	16 pins in the tubes
3.000	5,000	0.99690 ± 0.00035	0.99425 ± 0.00035	0.99042 ± 0.00035	0.98434 ± 0.00035
3.000	15,000	0.91069 ± 0.00034	0.90815 ± 0.00034	0.90348 ± 0.00034	0.89792 ± 0.00034
3.000	25,000	0.84624 ± 0.00033	0.84458 ± 0.00033	0.84018 ± 0.00033	0.83485 ± 0.00033
4.000	15,000	0.98417 ± 0.00035	0.98186 ± 0.00035	0.97681 ± 0.00035	0.96942 ± 0.00035
4.000	25,000	0.91903 ± 0.00035	0.91749 ± 0.00035	0.91253 ± 0.00035	0.90572 ± 0.00035
4.000	35,000	0.86703 ± 0.00031	0.86621 ± 0.00031	0.86095 ± 0.00031	0.85465 ± 0.00031
5.000	25,000	0.97603 ± 0.00035	0.97363 ± 0.00035	0.96745 ± 0.00035	0.96062 ± 0.00035
5.000	35,000	0.92513 ± 0.00037	0.92242 ± 0.00037	0.91737 ± 0.00037	0.91049 ± 0.00037
5.000	45,000	0.88094 ± 0.00034	0.87848 ± 0.00034	0.87367 ± 0.00034	0.86712 ± 0.00034

Table 3-10
Fuel Assembly Burnup versus Initial Enrichment for the "All-Cell" Storage Configuration with Fuel Pins in the Guide Tubes

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)			
	0 pins in the tubes[26]	4 pins in the tubes	9 pins the in tubes	16 pins in the tubes
3.000	7,697	7,395	6,952	6,355
4.000	16,925	16,617	15,919	14,911
5.000	26,085	25,636	24,518	23,300

[26] 0 year decay column from Table 3-8.

Table 3-11
K_{eff} Values for the All-Cell Storage Configuration with Fuel Pins in the Guide Tubes: Effect of Split Burnup

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/ MTU)	k_{eff} Value		
		0 pins the tubes	16 pins with uniform burnup	16 pins with split burnup[27]
3.000	15,000	0.91069 ± 0.00034	0.89792 ± 0.00034	0.89914 ± 0.00033
3.000	25,000	0.84624 ± 0.00033	0.83485 ± 0.00033	0.83540 ± 0.00035
4.000	25,000	0.91903 ± 0.00035	0.90572 ± 0.00035	0.90499 ± 0.00034
4.000	35,000	0.86703 ± 0.00031	0.85465 ± 0.00031	0.85522 ± 0.00033
5.000	35,000	0.92513 ± 0.00037	0.91049 ± 0.00037	0.91086 ± 0.00035
5.000	45,000	0.88094 ± 0.00034	0.86712 ± 0.00034	0.86620 ± 0.00034

[27] Eight of the pins have 10,000 MWD/MTU higher burnup than the average and eight have 10,000 MWD/MTU lower.

Table 3-12
K_{eff} Values versus Initial Enrichment and Assembly Burnup for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/ MTU)	k_{eff} Value				
		0 year decay	5 years decay	10 years decay	15 years decay	20 years decay
1.361	0	0.97558 ± 0.00045	0.97558 ± 0.00045	0.97558 ± 0.00045	0.97558 ± 0.00045	0.97558 ± 0.00045
3.000	15,000	1.00793 ± 0.00042	1.00408 ± 0.00040	1.00336 ± 0.00044	1.00213 ± 0.00045	1.00182 ± 0.00040
3.000	25,000	0.97564 ± 0.00045	0.97167 ± 0.00046	0.96883 ± 0.00044	0.96668 ± 0.00046	0.96465 ± 0.00041
3.000	35,000	0.95382 ± 0.00046	0.94816 ± 0.00044	0.94471 ± 0.00052	0.94174 ± 0.00047	0.93975 ± 0.00041
4.000	25,000	1.00877 ± 0.00042	1.00353 ± 0.00041	0.99935 ± 0.00043	0.99715 ± 0.00043	0.99494 ± 0.00038
4.000	35,000	0.97902 ± 0.00040	0.97241 ± 0.00047	0.96866 ± 0.00044	0.96498 ± 0.00043	0.96217 ± 0.00046
4.000	45,000	0.95796 ± 0.00049	0.95081 ± 0.00047	0.94733 ± 0.00050	0.94348 ± 0.00045	0.94131 ± 0.00046
5.000	35,000	1.00658 ± 0.00039	1.00123 ± 0.00045	0.99693 ± 0.00043	0.99313 ± 0.00042	0.99156 ± 0.00042
5.000	45,000	0.98022 ± 0.00043	0.97353 ± 0.00044	0.96986 ± 0.00047	0.96704 ± 0.00044	0.96374 ± 0.00045
5.000	55,000	0.95941 ± 0.00048	0.95408 ± 0.00044	0.94858 ± 0.00045	0.94491 ± 0.00046	0.94114 ± 0.00044

Table 3-13
Fuel Assembly Burnup versus Initial Enrichment for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)				
	0 yr decay	5 yr decay	10 yr decay	15 yr decay	20 yr decay
1.361	0	0	0	0	0
3.000	24,892	23,442	21,919	22,356	21,347
4.000	36,323	33,615	32,170	30,934	29,957
5.000	46,918	43,944	42,500	41,382	40,311

Note that the assembly burnups as a function of initial enrichment for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $276.83\ e^3 - 3740.62\ e^2 + 27373.43\ e - 31037.19$

Assembly Burnup (5 yr decay) = $452.23\ e^3 - 5348.68\ e^2 + 30881.71\ e - 33275.57$

Assembly Burnup (10 yr decay) = $524.70\ e^3 - 5948.94\ e^2 + 31863.58\ e - 33682.58$

Assembly Burnup (15 yr decay) = $651.51\ e^3 - 7101.87\ e^2 + 34622.21\ e - 35621.37$

Assembly Burnup (20 yr decay) = $700.10\ e^3 - 7529.25\ e^2 + 35411.07\ e - 36025.79$

Table 3-14
K_{eff} Values versus Initial Enrichment and Assembly Burnup for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration (for 0 to 20 Years Decay)

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)	k_{eff} Value				
		0 year decay	5 years decay	10 years decay	15 years decay	20 years decay
1.627	0	0.97410 ± 0.00040	0.97410 ± 0.00040	0.97410 ± 0.00040	0.97410 ± 0.00040	0.97410 ± 0.00040
3.000	15,000	0.98486 ± 0.00044	0.98111 ± 0.00038	0.97968 ± 0.00040	0.97898 ± 0.00039	0.97699 ± 0.00039
3.000	25,000	0.95025 ± 0.00050	0.94454 ± 0.00043	0.94239 ± 0.00043	0.93978 ± 0.00044	0.93809 ± 0.00042
3.000	35,000	0.92350 ± 0.00053	0.91719 ± 0.00043	0.91493 ± 0.00047	0.91263 ± 0.00047	0.91070 ± 0.00048
4.000	25,000	0.98613 ± 0.00049	0.98118 ± 0.00037	0.97625 ± 0.00041	0.97277 ± 0.00038	0.97059 ± 0.00039
4.000	35,000	0.95420 ± 0.00048	0.94715 ± 0.00041	0.94195 ± 0.00043	0.93933 ± 0.00043	0.93525 ± 0.00042
4.000	45,000	0.92901 ± 0.00050	0.92305 ± 0.00045	0.91886 ± 0.00043	0.91433 ± 0.00044	0.91057 ± 0.00047
5.000	35,000	0.98435 ± 0.00039	0.97819 ± 0.00040	0.97296 ± 0.00041	0.97039 ± 0.00042	0.96866 ± 0.00043
5.000	45,000	0.95566 ± 0.00043	0.95012 ± 0.00042	0.94412 ± 0.00044	0.94121 ± 0.00044	0.93868 ± 0.00041
5.000	55,000	0.93168 ± 0.00048	0.92537 ± 0.00042	0.92082 ± 0.00043	0.91542 ± 0.00043	0.91378 ± 0.00045

Table 3-15
Fuel Assembly Burnup versus Initial Enrichment for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)				
	0 yr decay	5 yr decay	10 yr decay	15 yr decay	20 yr decay
1.627	0	0.00	0	0	0
3.000	17,723	16,590	16,208	15,936	15,550
4.000	28,382	26,654	25,431	24,655	24,262
5.000	38,288	36,330	34,625	33,793	33,392

Note that the assembly burnups as a function of initial enrichment for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $170.36\ e^3 - 2420.63\ e^2 + 21300.07\ e - 28991.46$
Assembly Burnup (5 yr decay) = $195.63\ e^3 - 2541.74\ e^2 + 20617.59\ e - 27668.70$
Assembly Burnup (10 yr decay) = $318.85\ e^3 - 3841.38\ e^2 + 24315.99\ e - 30776.68$
Assembly Burnup (15 yr decay) = $423.62\ e^3 - 4874.30\ e^2 + 27165.59\ e - 33130.16$
Assembly Burnup (20 yr decay) = $389.42\ e^3 - 4464.01\ e^2 + 25551.28\ e - 31442.11$

Table 3-16
K_{eff} Values versus Number of IFBA Pins (1.0X) Contained in the 4.5 w/o ^{235}U Fresh Fuel of the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Enrichment of Fresh Fuel (w/o ^{235}U)	Burnup of Depleted Fuel (MWD/MTU)	Number of IFBA Pins in Fresh Fuel a,b,c	k_{eff}
4.5	35,000		0.99624 ± 0.00041
4.5	35,000		0.98656 ± 0.00038
4.5	35,000		0.97936 ± 0.00040
4.5	35,000		0.97443 ± 0.00041
4.5	45,000		0.96951 ± 0.00039
4.5	45,000		0.95977 ± 0.00037
4.5	45,000		0.95283 ± 0.00042
4.5	45,000		0.94796 ± 0.00043
4.5	55,000		0.94674 ± 0.00041
4.5	55,000		0.93499 ± 0.00045
4.5	55,000		0.92823 ± 0.00043
4.5	55,000		0.92199 ± 0.00044

Table 3-17
K_{eff} Values versus Number of IFBA Pins (1.0X) Contained in the 5.0 w/o ^{235}U Fresh Fuel of the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Enrichment of the Fresh Fuel (w/o ^{235}U)	Burnup of the Depleted Fuel (MWD/MTU)	Number of IFBA Pins a,b,c	k_{eff}
5.0	35,000		1.00658 ± 0.00039
5.0	35,000		0.99684 ± 0.00039
5.0	35,000		0.99104 ± 0.00037
5.0	35,000		0.98438 ± 0.00041
5.0	45,000		0.98022 ± 0.00043
5.0	45,000		0.97081 ± 0.00042
5.0	45,000		0.96442 ± 0.00045
5.0	45,000		0.95786 ± 0.00045
5.0	55,000		0.95941 ± 0.00048
5.0	55,000		0.94775 ± 0.00044
5.0	55,000		0.93998 ± 0.00046
5.0	55,000		0.93411 ± 0.00044

Table 3-18
Number of IFBAs versus Initial Enrichment for the Fresh Fuel Assembly in the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Number of IFBAs (1.0X)
4.000	0
4.500	23
5.000	56

Required Number of IFBA pins as a function of enrichment is given by the following polynomials:

Number of IFBA Pins = $20.0e^2 - 124.0e + 176.0$

Table 3-19
Entire Spent Fuel Pool k_{eff} Results for the Interface Configurations

	All-Cell		1-out-of-4 5.0 w/o Fresh with no IFBA		1-out-of-4 4.0 w/o Fresh with IFBA	
	k_{eff}	Max. Infinite Array k_{eff}	k_{eff}	Max. Infinite Array k_{eff}	k_{eff}	Max. Infinite Array k_{eff}
All-Cell						
1-out-of-4 5.0 w/o Fresh with no IFBA	0.97386 ± 0.00031	0.97558 ± 0.00045				
1-out-of-4 4.0 w/o Fresh with IFBA	0.96954 ± 0.00026	0.97410 ± 0.00040	0.97368 ± 0.00029	0.97558 ± 0.00045		

Table 3-20
Assembly Loading Requirements at the Interface between Different Storage Configurations

Configuration	Assembly that Must be Loaded at the Interface with Another Configuration[28]
"All-Cell"	Any
"1-out-of-4 5.0 w/o Fresh with no IFBA"	Only Depleted Fuel Assemblies
"1-out-of-4 4.0 w/o Fresh with IFBA"	Only Depleted Fuel Assemblies

Instructions:

1. Identify which storage configurations will be interfaced.
2. Look up the assembly loading requirements for <u>both</u> storage configurations.

[28] An empty storage location is always permitted.

Table 3-21
K_{eff} Values for the Failed Fuel Rod Storage Basket with 5.0 w/o ^{235}U Fresh Fuel in the All-Cell Storage Configuration

Fuel Pin Pitch in the FFRSB (inches)	k_{eff} with FFRSB	Nominal k_{eff} for All-Cell
P = 1.016" (fuel pins touching)	0.90101 ± 0.00039	0.96950 ± 0.00035
P = 2.3811" (typical storage basket design pitch)	0.93310 ± 0.00034	0.96950 ± 0.00035
P = 2.99357" (maximum pitch in the cell)	0.94417 ± 0.00036	0.96950 ± 0.00035

Table 3-22
K_{eff} Values as a Function of Soluble Boron Concentration for the Spent Fuel Pool with Depleted Fuel Assemblies in the "All-Cell" Storage Configuration

Configuration	k_{eff}			
	0 ppm	**200 ppm**	**400 ppm**	**600 ppm**
Depleted Fuel (5.0 w/o, 45,000 MWD/MTU)	0.87565 ± 0.00024	0.83753 ± 0.00022	0.80412 ± 0.00020	0.77528 ± 0.00020

Note that the following polynomial describes an amount of boron as a function of Δk_{eff} for the entire spent fuel pool:

$ppm = 58842.012\Delta k_{eff}^3 + 3053.398\Delta k_{eff}^2 + 5112.229\Delta k_{eff}$

Table 3-23
Summary of Burnup Reactivity Uncertainties for the Storage Configurations

Configuration	Maximum Burnup (MWD/MTU)	5% Burnup Uncertainty	Δk_{eff}
All-Cell	27,000	1,350	0.00714
1-out-of-4 5.0 w/o Fresh with no IFBA	47,000	2,350	0.00528
1-out-of-4 4.0 w/o with IFBA	39,000	1,950	0.00569

Table 3-24
K_{eff} Values for Various Accident Scenarios in the Spent Fuel Pool

	All-Cell		1-out-of-4 5.0 w/o Fresh with no IFBA		1-out-of-4 4.0 w/o Fresh with IFBA	
Accident Scenarios	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
Misloaded fresh fuel assembly into burnup storage rack location	0.94835 ± 0.00032	0.07154[29]	1.01243 ± 0.00031	0.05597[30]	0.99475 ± 0.00034	0.06564[31]
Misloaded fresh fuel assembly in the cask area between storage racks	0.96492 ± 0.00021	0.00019	0.97658 ± 0.00022	0.00334	0.97065 ± 0.00028	-0.00020
Spent fuel pool temperature greater than normal operating range (240°F)	0.98586 ± 0.00022	0.02410	0.96196 ± 0.00021	-0.00810	0.97640 ± 0.00023	0.00894

[29]Based on the nominal k_{eff} value of 0.87692 ± 0.00021 for a pool filled with "All-Cell" storage configuration containing depleted fuel with 5.0 w/o ^{235}U initial enrichment at 45,000 MWD/MTU

[30]Based on the nominal k_{eff} value of 0.95648 ± 0.00029 for a pool filled with "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration containing depleted fuel with 5.0 w/o ^{235}U initial enrichment at 55,000 MWD/MTU

[31]Based on the nominal k_{eff} value of 0.92919 ± 0.00026 for a pool filled with "1-out-of-4 4.0 with IFBA" storage configuration containing depleted fuel with 5.0 w/o ^{235}U initial enrichment at 55,000 MWD/MTU.

Table 3-25
Soluble Boron required to Mitigate Various Accidents in the Spent Fuel Pool

	All-Cell	1-out-of-4 5.0 w/o Fresh with no IFBA	1-out-of-4 4.0 w/o Fresh with IFBA
Accident Scenarios	**[ppm]**	**[ppm]**	**[ppm]**
Misloaded fresh fuel assembly into burnup storage rack location	402.9	306.0	365.4
Misloaded fresh fuel assembly in the cask area between storage racks	1.0	17.1	-
Spent fuel pool temperature greater than normal operating range (240°F)	125.8	-	46.0

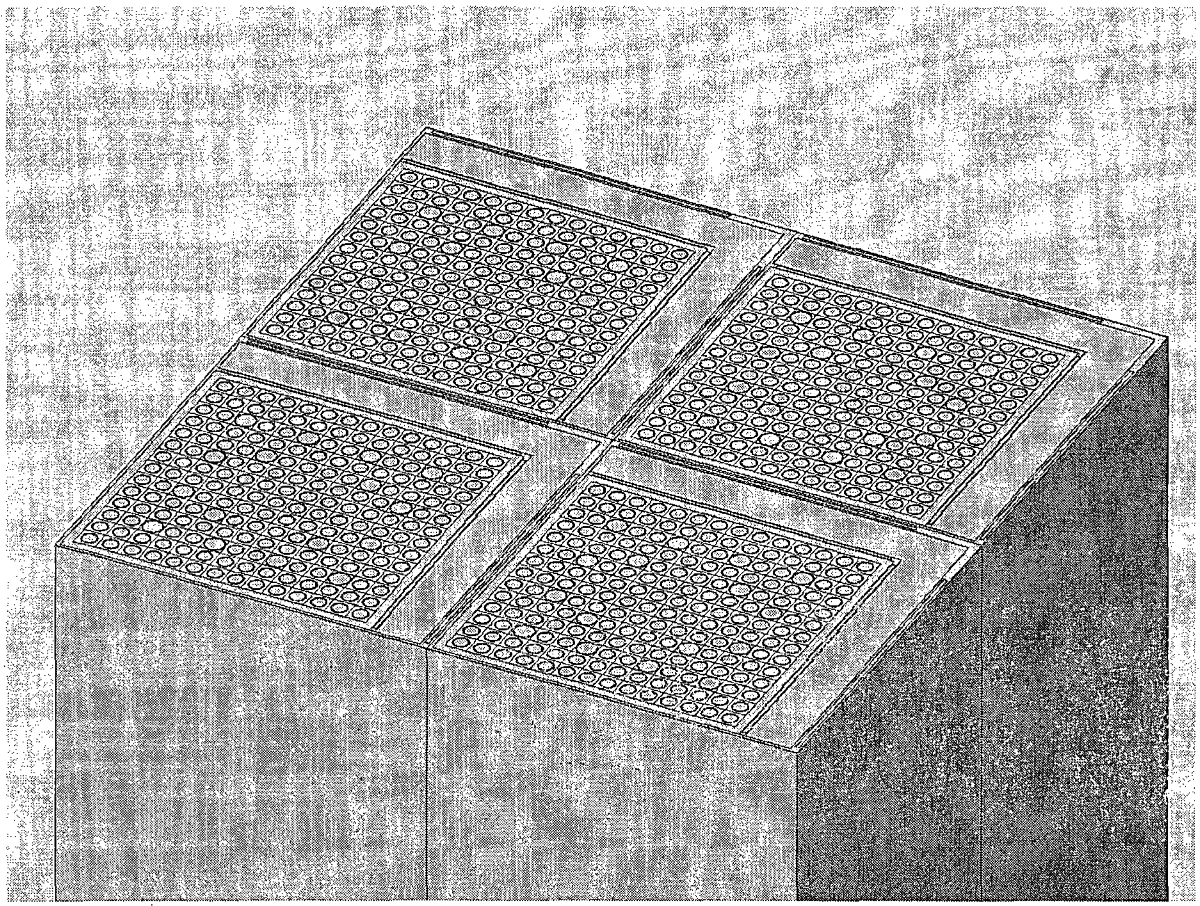

Figure 3-1 KENO Plot for the "All-Cell" Storage Configuration

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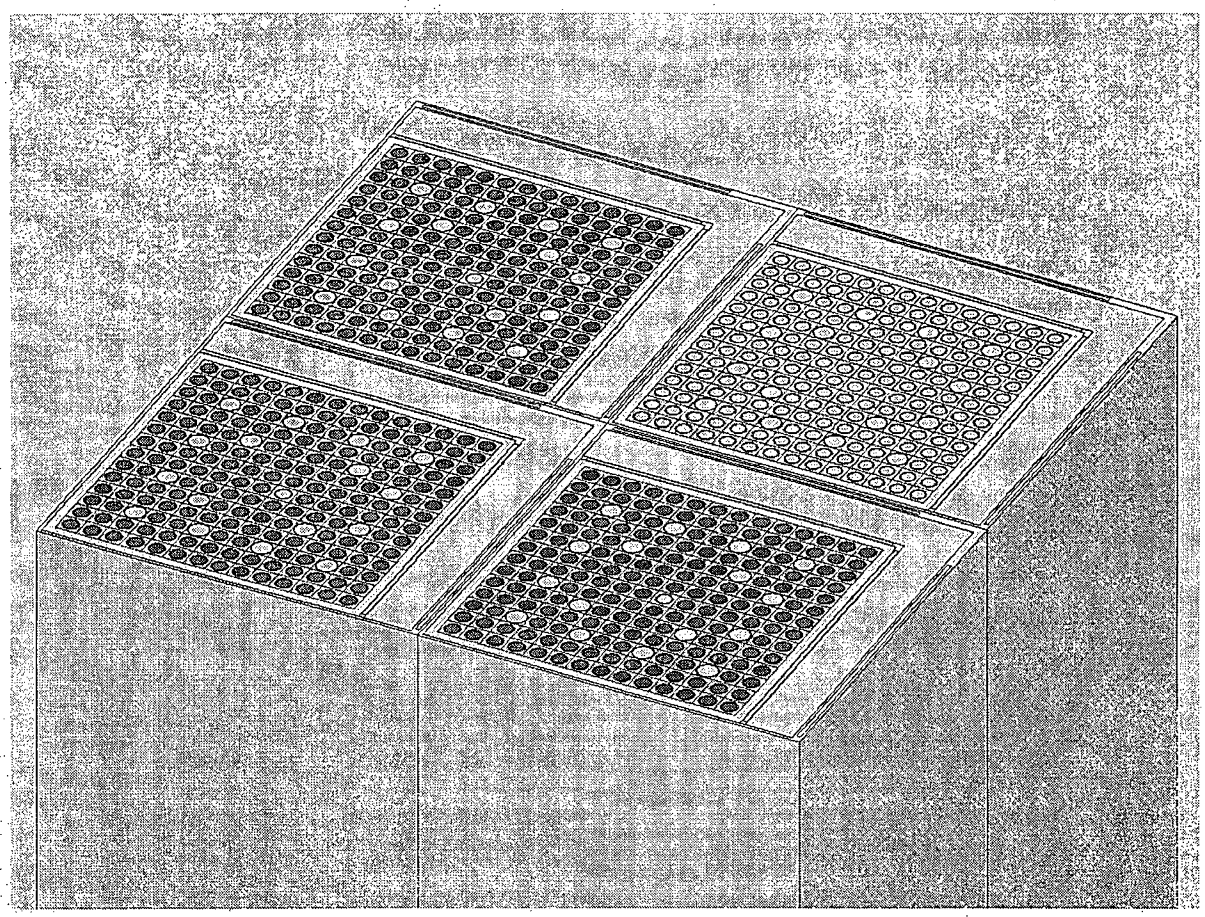

Figure 3-2 KENO Output Plot for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

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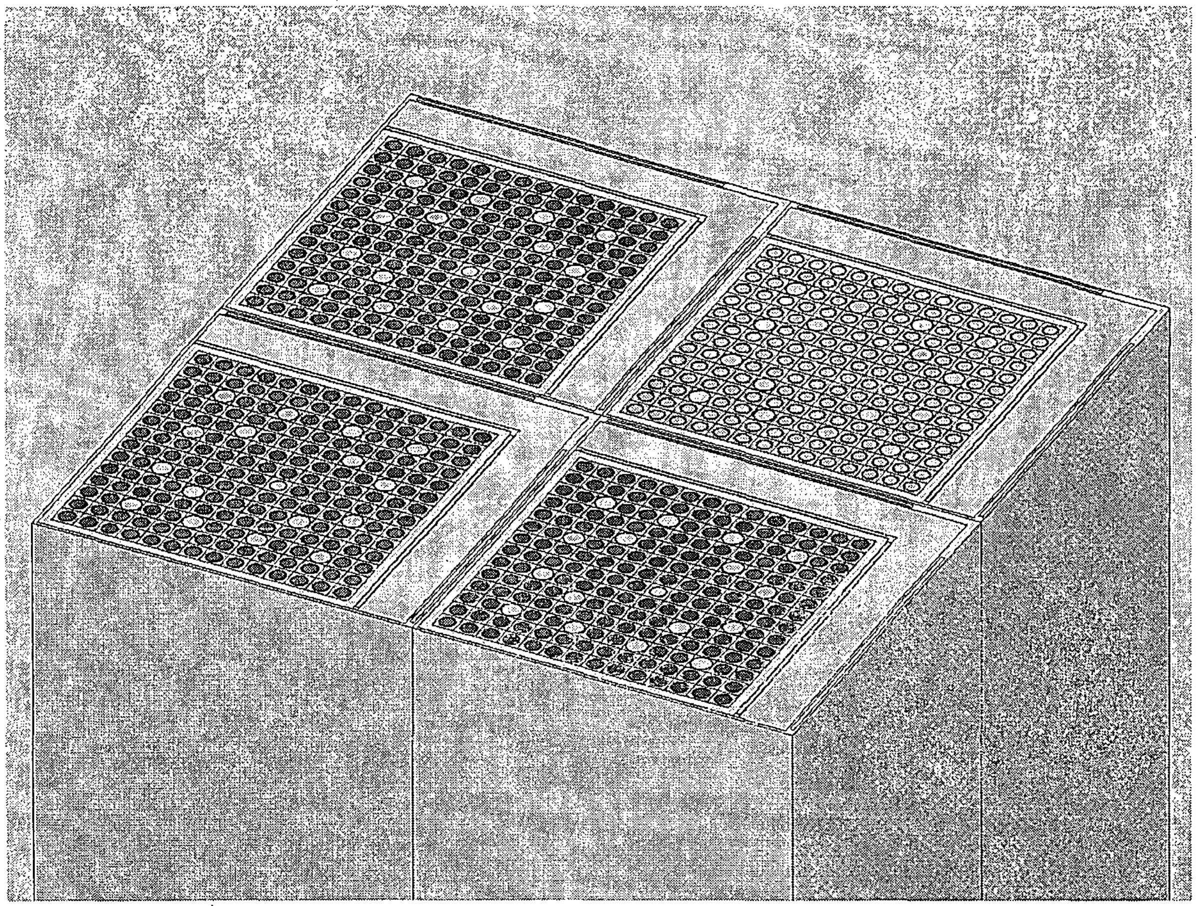

Figure 3-3 KENO Output Plot for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

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a,b,c

Figure 3-4 IFBA Patterns Used in the Point Beach Criticality Analysis

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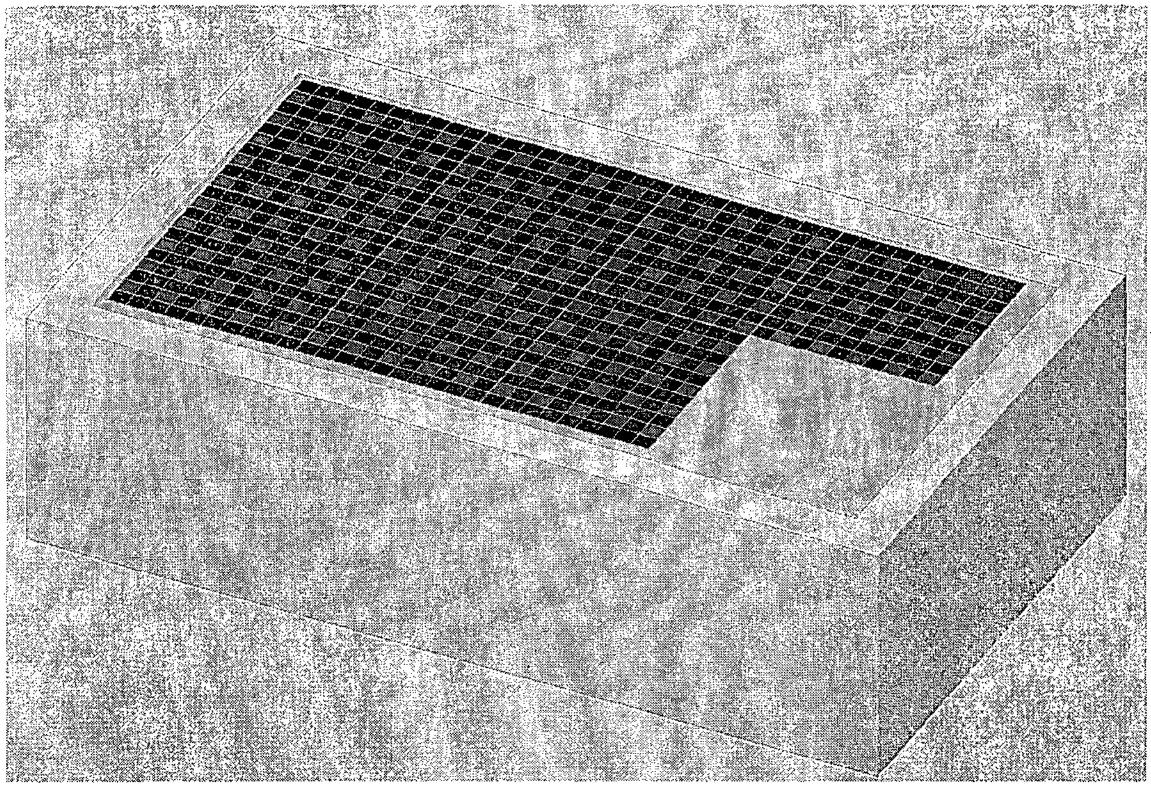

Figure 3-5 KENO Output Plot for the Spent Fuel Pool Loaded with the "1-out-of 4 4.0 w/o Fresh with IFBA" Storage Configurations

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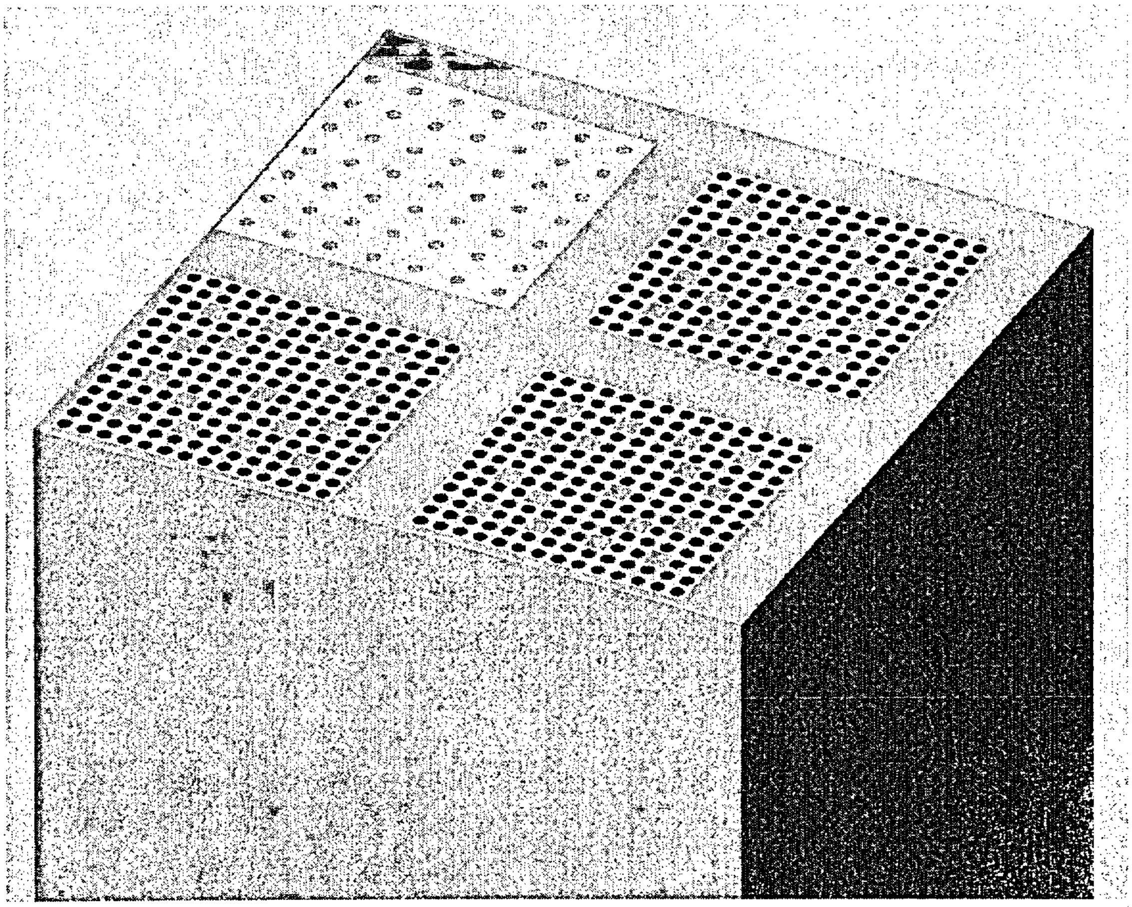

Figure 3-6 KENO Output Plot of the Failed Fuel Rod Storage Basket in the "All-Cell" Storage Configurations

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[ ] a,b,c

Figure 3-7 Westinghouse 14x14 Standard and (OFA) Fuel Dimensions

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a,b,c

Figure 3-8 Sketch of Axial Zones Used in Fuel Assembly

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4.0 Summary of Results

The following sections contain the criticality analysis results for the Point Beach spent fuel pools with soluble boron credit.

4.1 Allowable Storage Configurations

4.1.1 "All-Cell" Storage Configuration

Figure 4-1 displays the allowable storage configurations for the "All-Cell" storage. The "All-Cell" storage rack configuration will be employed to store depleted fuel assemblies which meet the requirements of Table 4-1 and Figure 4-7.

4.1.1.1 Storage of Fuel Pins in the Guide Tubes in an All-Cell Storage Configuration

Fuel pins stored in the guide tubes of an assembly in an All-Cell configuration shall meet the burnup requirements of the All-Cell configuration with zero year decay time and without any fuel pins in the guide tubes. Fuel pins that do not meet the requirements of the All-Cell storage configuration (i.e., pins that have less burnup by a certain amount) can be stored in the guide tubes only if the same or greater number of pins that exceed the burnup requirement by the same or larger amount are also loaded in the guide tubes to offset the excess reactivity.

4.1.2 "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

Figure 4-2 displays the allowable storage configurations for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage. The "1-out-of-4 5.0 w/o Fresh with no IFBA" storage rack will be employed to store fresh assemblies with enrichments up to 5.0 w/o ^{235}U and depleted fuel assemblies which meet the requirements of Table 4-2 and Figure 4-8.

4.1.3 "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Figure 4-3 displays the allowable storage configurations for the "1-out-of-4 4.0 w/o Fresh with IFBA" storage. The "1-out-of-4 4.0 w/o Fresh with IFBA" storage rack will be employed to store fresh fuel assemblies with enrichments up to 4.0 w/o ^{235}U and depleted fuel assemblies which meet the requirements of Table 4-3 and Figure 4-9. The fresh fuel assemblies with enrichments greater than 4.0 w/o ^{235}U and up to 5.0 w/o ^{235}U shall meet the requirements of Table 4-4 and Figure 4-10. [ ]^{a,b,c}

4.1.4 Burnup Requirements for Intermediate Decay Time Points

For all the storage configurations in the Point Beach Spent Fuel pool crediting ^{241}Pu decay, burnup requirements for intermediate decay time points should be determined using at least a second order polynomial.

4.2 Interface Requirements in Spent Fuel Pool Storage Racks

Fuel storage patterns used at the interface of storage configurations shall comply with the assembly loading requirements provided in Table 4-5 and Figure 4-4 to Figure 4-6. Note that it is acceptable to leave a storage cell empty.

4.3 Failed Fuel Rod Storage Basket with 5.0 w/o ^{235}U Fuel

Failed Rod Storage Baskets filled with fresh fuel rods with a maximum enrichment of 5.0 w/o ^{235}U and no burnable absorbers can be stored in the All-Cell storage configuration.

4.4 Empty Cells

For all configurations at Point Beach, an empty cell is permitted in any location of the spent fuel pool to replace an assembly since the water cell will not cause any increase in reactivity in the spent fuel pool. Non-fissile material and debris canisters may be stored in empty cells of All-Cell storage configuration provided that the canister does not contain fissile materials.

4.5 Non-Fissile Equipment

Non fissile equipment, such as UT cleaning equipment is permitted on top of the fuel storage racks, as these equipments will not cause any increase in reactivity in the spent fuel pool.

4.6 Total Soluble Boron Requirement

The soluble boron (with 19.9% ^{10}B abundance) necessary to maintain k_{eff} less than or equal to 0.95 (including all biases and uncertainties) is 389.5 ppm. The soluble boron concentration required for a ^{10}B atom percent equal to 19.6 (expected lowest pool value crediting ^{10}B depletion) is 395.5 ppm. A total of 792.4 ppm of soluble boron (with 19.9% ^{10}B abundance) is required to maintain k_{eff} less than or equal to 0.95 (including all biases and uncertainties) and assuming the most limiting single accident. The soluble boron concentration required for a ^{10}B atom percent equal to 19.6 (expected lowest pool value crediting ^{10}B depletion) is 804.5 ppm. The recommended minimum boron level is 804.5 ppm and is sufficient to accommodate all the design requirements.

Table 4-1
Fuel Assembly Burnup versus Initial Enrichment for the "All-Cell" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)				
	0 yr decay[32]	5 yr decay[32]	10 yr decay[32]	15 yr decay[32]	20 yr decay[32]
2.147	0	0	0	0	0
3.000	7,697	7,308	7,225	7,222	7,227
4.000	16,925	16,286	15,741	15,403	15,148
5.000	26,085	25,080	24,161	23,662	23,402

Note that the assembly burnups as a function of initial enrichment for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $-49.58\ e^3 + 561.12\ e^2 + 7134.44\ e - 17417.89$

Assembly Burnup (5 yr decay) = $-108.67\ e^3 + 1212.54\ e^2 + 4510.52\ e - 14201.98$

Assembly Burnup (10 yr decay) = $-24.62\ e^3 + 247.20\ e^2 + 7696.65\ e - 17424.68$

Assembly Burnup (15 yr decay) = $68.81\ e^3 - 786.56\ e^2 + 11140.74\ e - 20978.70$

Assembly Burnup (20 yr decay) = $163.70\ e^3 - 1797.81\ e^2 + 14448.84\ e - 24359.00$

[32] Decay time is defined as the number of years since fuel assembly discharge

Table 4-2
Fuel Assembly Burnup versus Initial Enrichment for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)				
	0 yr decay[33]	5 yr decay[33]	10 yr decay[33]	15 yr decay[33]	20 yr decay[33]
1.361	0	0	0	0	0
3.000	24,892	23,442	21,919	22,356	21,347
4.000	36,323	33,615	32,170	30,934	29,957
5.000	46,918	43,944	42,500	41,382	40,252

Note that the assembly burnups as a function of initial enrichment for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $276.83\ e^3 - 3740.62\ e^2 + 27373.43\ e - 31037.19$

Assembly Burnup (5 yr decay) = $452.23\ e^3 - 5348.68\ e^2 + 30881.71\ e - 33275.57$

Assembly Burnup (10 yr decay) = $524.70\ e^3 - 5948.94\ e^2 + 31863.58\ e - 33682.58$

Assembly Burnup (15 yr decay) = $651.51\ e^3 - 7101.87\ e^2 + 34622.21\ e - 35621.37$

Assembly Burnup (20 yr decay) = $692.01\ e^3 - 7461.55\ e^2 + 35236.72\ e - 35893.48$

[33] Decay time is defined as the number of years since fuel assembly discharge

Table 4-3
Fuel Assembly Burnup versus Initial Enrichment for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Limiting Burnup (MWD/MTU)				
	0 yr decay[34]	5 yr decay[34]	10 yr decay[34]	15 yr decay[34]	20 yr decay[34]
1.627	0	0	0	0	0
3.000	17,723	16,590	16,208	15,936	15,550
4.000	28,382	26,654	25,431	24,655	24,262
5.000	38,288	36,330	34,625	33,793	33,392

Note that the assembly burnups as a function of initial enrichment for each decay period are described by the following polynomials:

Assembly Burnup (0 yr decay) = $170.36\ e^3 - 2420.63\ e^2 + 21300.07\ e - 28991.46$
Assembly Burnup (5 yr decay) = $195.63\ e^3 - 2541.74\ e^2 + 20617.59\ e - 27668.70$
Assembly Burnup (10 yr decay) = $318.85\ e^3 - 3841.38\ e^2 + 24315.99\ e - 30776.68$
Assembly Burnup (15 yr decay) = $423.62\ e^3 - 4874.30\ e^2 + 27165.59\ e - 33130.16$
Assembly Burnup (20 yr decay) = $389.42\ e^3 - 4464.01\ e^2 + 25551.28\ e - 31442.11$

[34] Decay time is defined as the number of years since fuel assembly discharge

Table 4-4
Number of IFBAs versus Initial Enrichment for the Fresh Fuel Assembly in the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Number of IFBAs (1.0X)
4.000	0
4.500	23
5.000	56

Required Number of IFBA pins as a function of enrichment is given by the following polynomials:

Number of IFBA Pins = $20.0e^2 - 124.0e + 176.0$

Table 4-5
Assembly Loading Requirements at the Interface between Different Storage Configurations

Configuration	Assembly that Must be Loaded at the Interface with Another Configuration[35]
"All-Cell"	Any
"1-out-of-4 5.0 w/o Fresh with no IFBA"	Only Depleted Fuel Assemblies
"1-out-of-4 4.0 w/o Fresh with IFBA"	Only Depleted Fuel Assemblies

Instructions:

1. Identify which storage configurations will be interfaced.
2. Look up the assembly loading requirements for both storage configurations.

[35] An empty storage location is always permitted.

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A	**A**
A	**A**

A : Fuel Assembly meeting the requirements of Table 4-1 or Figure 4-7 for the "All-Cell" Configuration

Figure 4-1 Allowable Fuel Assemblies in the "All-Cell" Storage Configuration

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L1	**H1**
L1	**L1**

L1 : Fuel Assembly meeting the requirements of Table 4-2 and Figure 4-8 for the "1-out-of 4 5.0 w/o Fresh with no IFBA" Configuration

H1 : Fuel Assembly with 5.0 w/o Fresh in the "1-out-of 4 5.0 w/o Fresh with no IFBA" Configuration

Note: The 2x2 array is repeated with the same orientation

Figure 4-2 Allowable Fuel Assemblies in the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

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L2	**H2**
L2	**L2**

L2 : Fuel Assembly meeting the requirements of Table 4-3 or Figure 4-9 for the "1-out-of 4 4.0 w/o Fresh with IFBA" Configuration

H2 : Fuel Assembly with 4.0 w/o Fresh in the "1-out-of 4 4.0 w/o Fresh with IFBA" Configuration

Note: The 2x2 array is repeated with the same orientation.

Figure 4-3 Allowable Fuel Assembly Categories in the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

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1-out-of-4 5 w/o Fresh

All-Cell

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
L1	L1	L1	L1	A	A	A
H1	L1	H1	L1	A	A	A
L1	L1	L1	L1	A	A	A
H1	L1	H1	L1	A	A	A

A : Fuel Assembly meeting the requirements of Table 4-1 or Figure 4-7 for the "All-Cell" Configuration

L1 : Fuel Assembly meeting the requirements of Table 4-2 and Figure 4-8 for the "1-out-of 4 5.0 w/o Fresh with no IFBA" Configuration

H1 : Fuel Assembly with 5.0 w/o Fresh in the "1-out-of 4 5.0 w/o Fresh with no IFBA" Configuration

Figure 4-4 Allowable Interface between "All-Cell" and "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configurations

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1-out-of-4 4.0 w/o Fresh with IFBA

A	A	A	A	A	A	A
A	A	A	A	A	A	A
A	A	A	A	A	A	A
L2	L2	L2	L2	A	A	A
H2	L2	H2	L2	A	A	A
L2	L2	L2	L2	A	A	A
H2	L2	H2	L2	A	A	A

All-Cell

A : Fuel Assembly meeting the requirements of Table 4-1 or Figure 4-7 for the "All-Cell" Configuration

L2 : Fuel Assembly meeting the requirements of Table 4-3 or Figure 4-9 for the "1-out-of 4 4.0 w/o Fresh with IFBA" Configuration

H2 : Fuel Assembly with 4.0 w/o Fresh in the "1-out-of 4 4.0 w/o Fresh with IFBA" Configuration

Figure 4-5 Allowable Interface between "All-Cell" and "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configurations

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1-out-of-4 4.0 w/o Fresh with IFBA

L1	L1	L1	L1	L1	L1	L1
L1	H1	L1	H1	L1	H1	L1
L1	L1	L1	L1	L1	L1	L1
L2	L2	L2	L2	L1	H1	L1
H2	L2	H2	L2	L1	L1	L1
L2	L2	L2	L2	L1	H1	L1
H2	L2	H2	L2	L1	L1	L1

1-out-of-4 5 w/o Fresh

L1 : Fuel Assembly meeting the requirements of Table 4-2 or Figure 4-8 for the "1-out-of 4 5.0 w/o Fresh with no IFBA" Configuration

H1 : Fuel Assembly with 5.0 w/o Fresh in the "1-out-of 4 5.0 w/o Fresh with no IFBA" Configuration

L2 : Fuel Assembly meeting the requirements of Table 4-3 or Figure 4-9 for the "1-out-of 4 4.0 w/o Fresh with IFBA" Configuration

H2 : Fuel Assembly with 4.0 w/o Fresh in the "1-out-of 4 4.0 w/o Fresh with IFBA" Configuration

Figure 4-6 Allowable Interface between "1-out-of-4 5.0 w/o Fresh with no IFBA" and "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configurations

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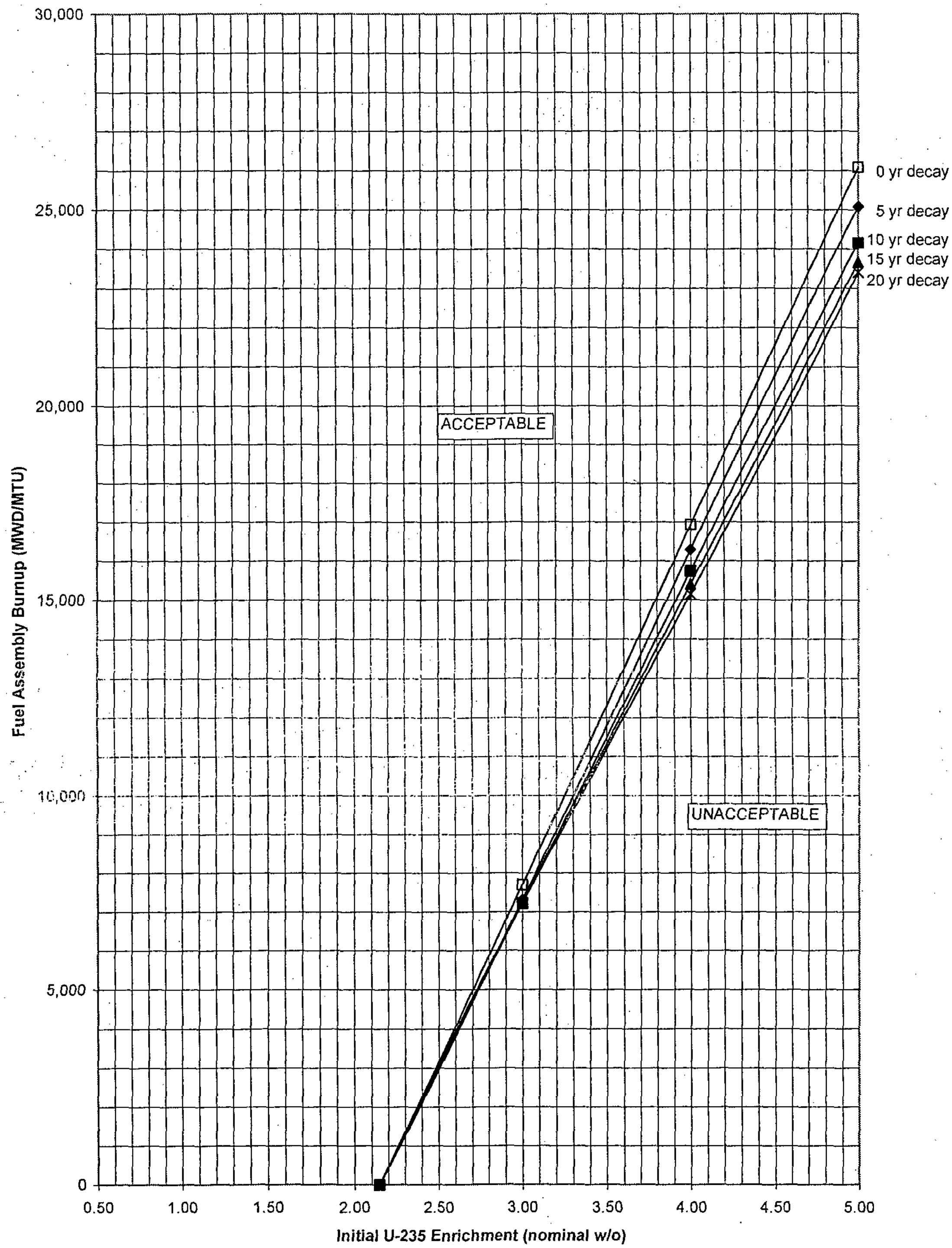

Figure 4-7 Fuel Assembly Burnup versus Initial Enrichment for the "All Cell" Storage Configuration

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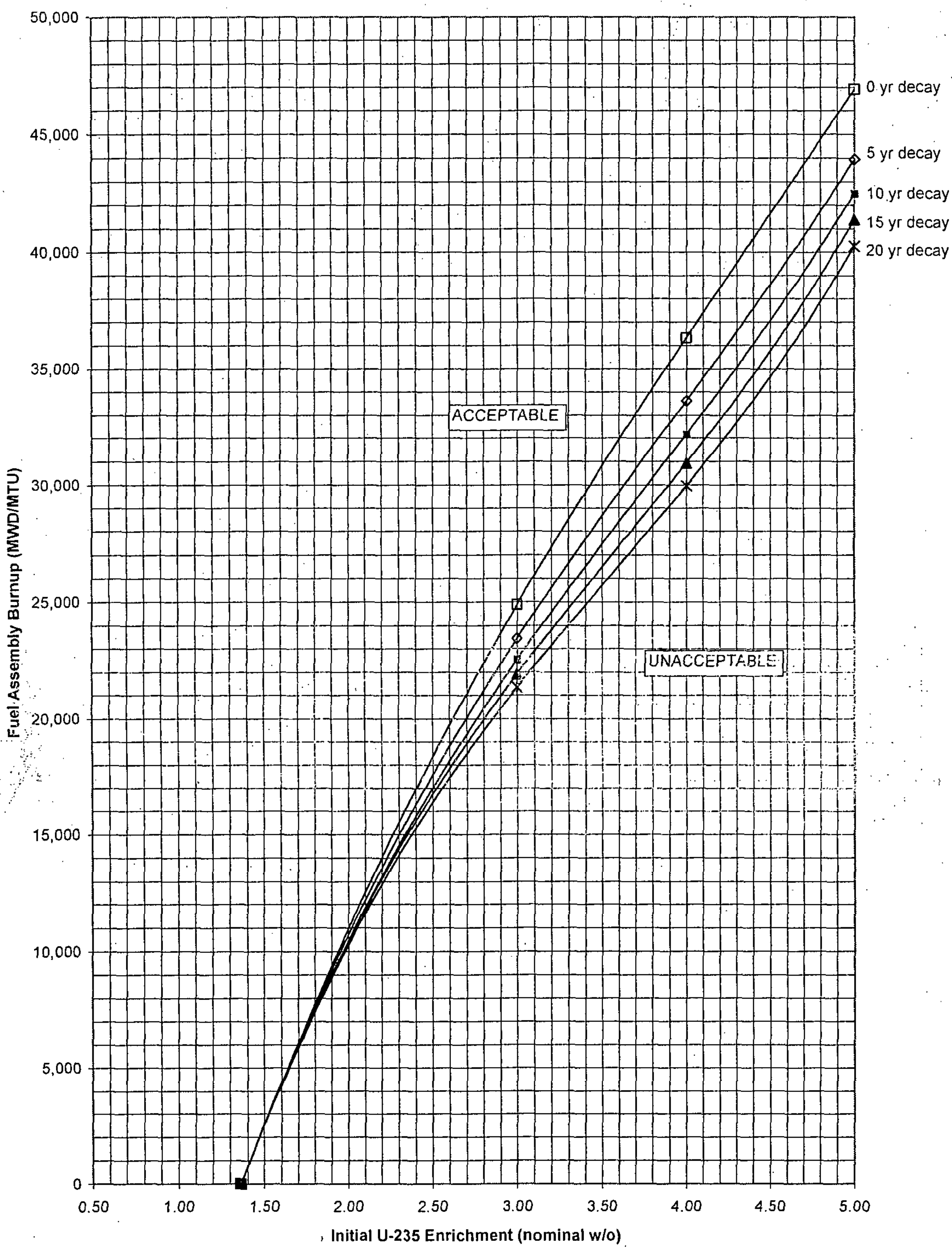

Figure 4-8 Fuel Assembly Burnup versus Initial Enrichment for the "1-out-of-4 5.0 w/o Fresh with no IFBA" Storage Configuration

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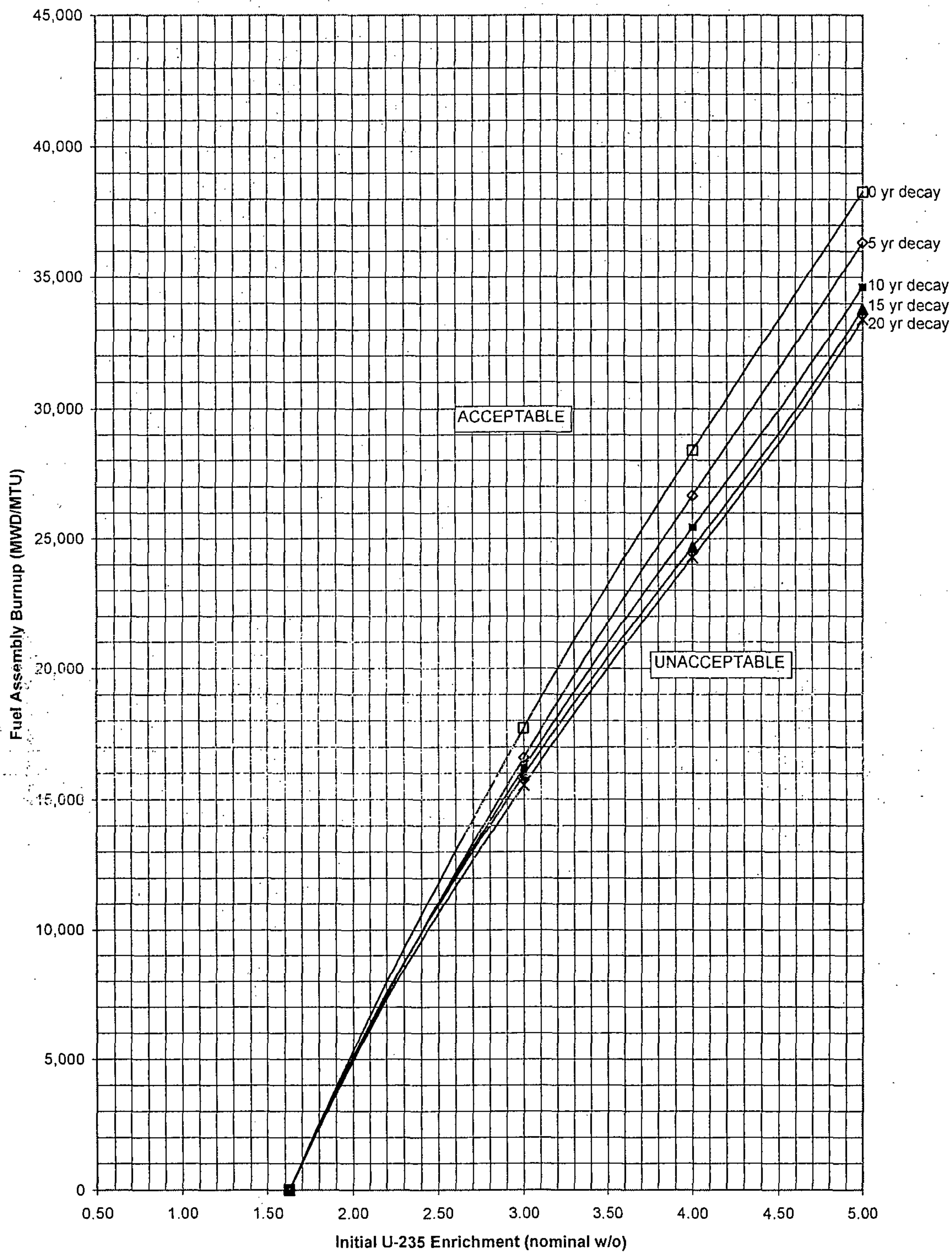

Figure 4-9 Fuel Assembly Burnup versus Initial Enrichment for the "1-out-of-4 4.0 w/o Fresh with IFBA" Storage Configuration

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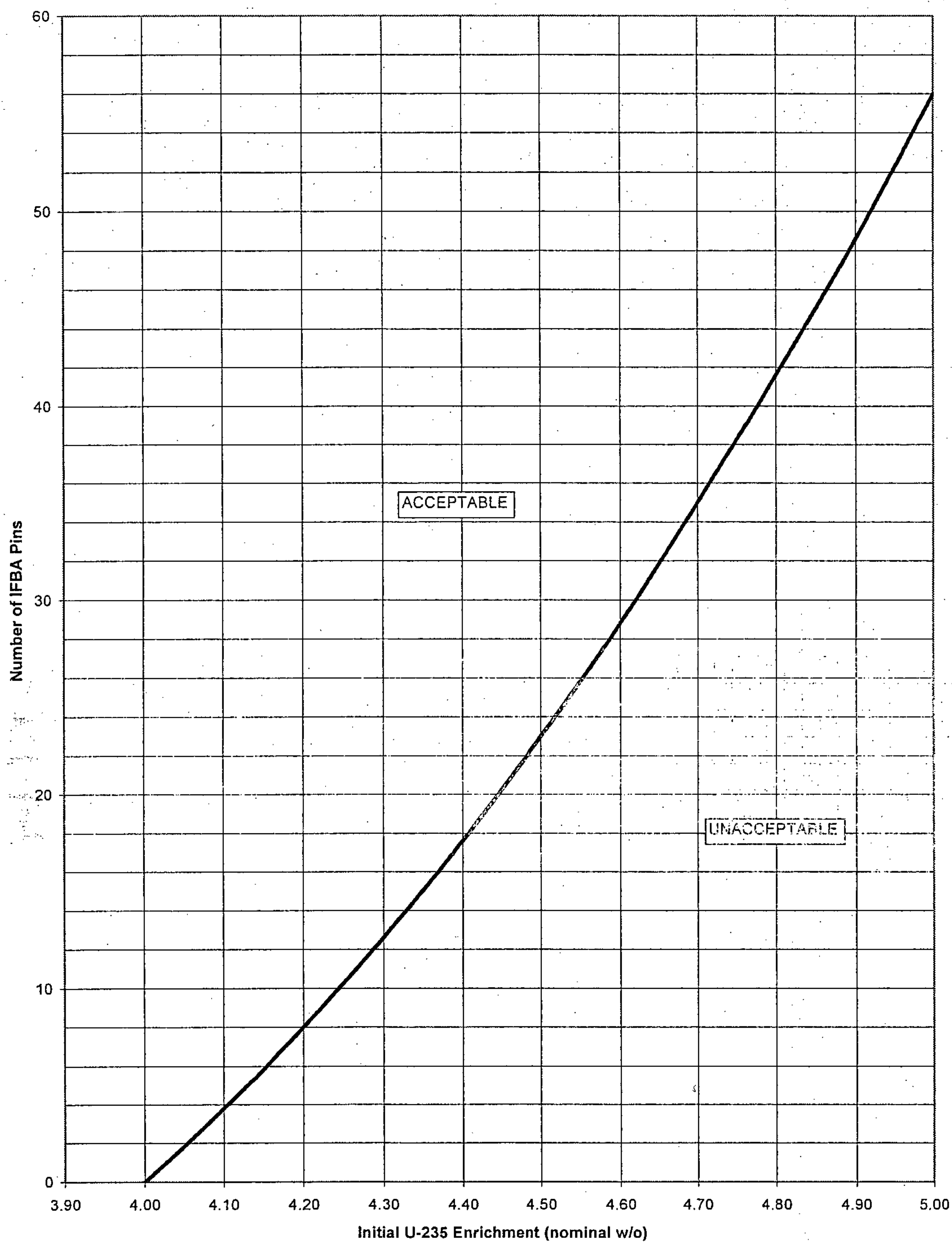

Figure 4-10 IFBA Requirements for the Fresh Fuel Assembly with Enrichments Greater than 4.0 w/o ^{235}U in the "1-out-of-4 4.0 w/o with IFBA" Storage Configuration

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5.0 Computer Codes Used In Calculation

Table 5-1
Summary of Computer Codes Used in Point Beach Spent Fuel Pool Criticality Calculations

Code No.	Code Name	Code Version	Verified and Configured per EP-310 or EP-313? (Yes/No) or Configuration Control Reference	Basis (or reference) that supports use of code in current calculation	Outstanding Category A Error? (Yes/No). If Yes, how acceptable?
1	SCALE-PC	4.4a	See Footnote[36]	See Footnote[36]	See Note

Notes:

1. NRC Information Notice 2005-13, "Potential Non-Conservative Error in Modeling Geometric Regions in the KENO-V.A Criticality Code", May 17, 2005 notifies of an error in SCALE associated with cylindrical holes with shared boundaries. In the standard spent fuel pool analysis, none of the input files involve cylindrical holes with shared boundaries; therefore, the analysis is not affected from this code error.
2. NRC Information Notice 2005-31, "Potential Non-Conservative Error in Preparing Problem-Dependent Cross Sections for use with the KENO-V.a or KENO-VI Criticality Code", November 17, 2005". This programming error in SCALE version 5 does not cause erroneous results in current Westinghouse criticality analyses for the following reasons:
   a. Westinghouse has not implemented SCALE version 5 for criticality analyses. All current analyses have been performed using SCALE 4.4 or earlier versions.
   b. These options are only used for slab geometry (e.g., plate-type fuel), and Westinghouse analyses that are applicable to pressurized water reactor (PWR) and boiling water reactor (BWR) fuel lattices do not utilize this functionality.

[36] Validation and benchmarking of the SCALE-PC Code package version 4.4a installation was performed as described in subsection 1.4.2. Verification of SCALE-PC Version 4.4a was achieved by running the sample test problems provided in the software package. Only differences in the outputs are due to time/date information and the header lines.

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6.0 References

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2. Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)," October 25, 1996.
3. Not Used.
4. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."
5. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 1998.
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7. "SCALE 4.4a- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC CODE PACKAGE CCC-545, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2000.
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9. M. N. Baldwin, et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel; Summary Report," BAW-1484-7, July 1979.
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15. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandia Corporation Monograph, March 1963.
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17. Not Used.
18. "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801, March 2003.

19. "Point Beach Nuclear Plant Reactor Engineering Surveillance Procedures: Defective Fuel Rod Replacement", Resp 2.2 Major Ipte Revision 1, March 12, 1993

ENCLOSURE 7

WCAP-16541-P (PROPRIETARY)
APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION
AFFIDAVIT
PROPRIETARY INFORMATION NOTICE
COPYRIGHT NOTICE

POINT BEACH UNITS 1 AND 2
SPENT FUEL POOL CRITICALITY ANALYSIS
DATED FEBRUARY 2006

LICENSE AMENDMENT REQUEST 247
SPENT FUEL POOL STORAGE CRITICALITY CONTROL

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

(120 pages follow)

UPON REMOVAL OF THE ATTACHMENT THE REMAINDER OF THIS LETTER MAY BE DECONTROLLED