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

March 5, 2010

Mr. Larry Meyer
Site Vice President
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: SPENT FUEL POOL STORAGE CRITICALITY CONTROL (TAC NOS. MD9321 AND MD9322)

Dear Mr. Meyer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 236 to Renewed Facility Operating License No. DPR-24 and Amendment No. 240 to Renewed Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated July 24, 2008, as supplemented by letters dated September 19, 2008, April 14, May 22, August 7, August 27, November 20, 2009, and February 2, 2010.

These amendments revise the Point Beach Nuclear Plant licensing basis to reflect a revision to the spent fuel pool (SFP) criticality analysis methodology. The changes to TS 3.7.12, "Spent Fuel Pool Storage," and 4.3.1, "Criticality," impose new storage requirements reflecting the new SFP criticality analysis. Upon approval of these changes, Boraflex will no longer be credited in the criticality analysis and, therefore, the Boraflex Surveillance Program will be discontinued.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Justin C. Poole, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 236 to DPR-24
2. Amendment No. 240 to DPR-27
3. Safety Evaluation

cc w/encls: Distribution via ListServ

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

FPL ENERGY POINT BEACH, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 236
License No. DPR-24

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

   A. The application for amendment by FPL Energy Point Beach, LLC (the licensee), dated July 24, 2008, as supplemented by letters dated September 19, 2008, April 14, May 22, August 7, August 27, November 20, 2009, and February 2, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 4.B of Renewed Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 236, are hereby incorporated in the renewed operating license. FPLE Point Beach shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications
and Facility Operating License

Date of issuance: March 5, 2010

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

FPL ENERGY POINT BEACH, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 240
License No. DPR-27

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

   A. The application for amendment by FPL Energy Point Beach, LLC (the licensee), dated July 24, 2008, as supplemented by letters dated September 19, 2008, April 14, May 22, August 7, August 27, November 20, 2009, and February 2, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 4.B of Renewed Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 240 , are hereby incorporated in the renewed operating license. FPLE Point Beach shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Tony A. Brito for

Robert J. Pascarelli, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications
and Facility Operating License

Date of issuance: March 5, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 236

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 240

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Replace the following pages of the Facility Operating Licenses and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Unit 1 License Page 3	Unit 1 License Page 3
Unit 2 License Page 3	Unit 2 License Page 3
TS Page 3.7.12-1	TS Page 3.7.12-1
TS Page 3.7.12-2	TS Page 3.7.12-2
TS Page 4.0-2	TS Page 4.0-2
TS Page 4.0-3	TS Page 4.0-3
------	TS Page 4.0-4
------	TS Page 4.0-5
------	TS Page 4.0-6
------	TS Page 4.0-7
------	TS Page 4.0-8
------	TS Page 4.0-9
------	TS Page 4.0-10
------	TS Page 4.0-11

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, FPLE Point Beach to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, FPLE Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

4. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Levels

FPLE Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 236 , are hereby incorporated in the renewed operating license. FPLE Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, FPLE Point Beach to receive, possess and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed source for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, FPLE Point Beach to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, FPLE Point Beach to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

4. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Levels

FPLE Point Beach is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 240 are hereby incorporated in the renewed operating license. FPLE Point Beach shall operate the facility in accordance with Technical Specifications.

C. Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

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.

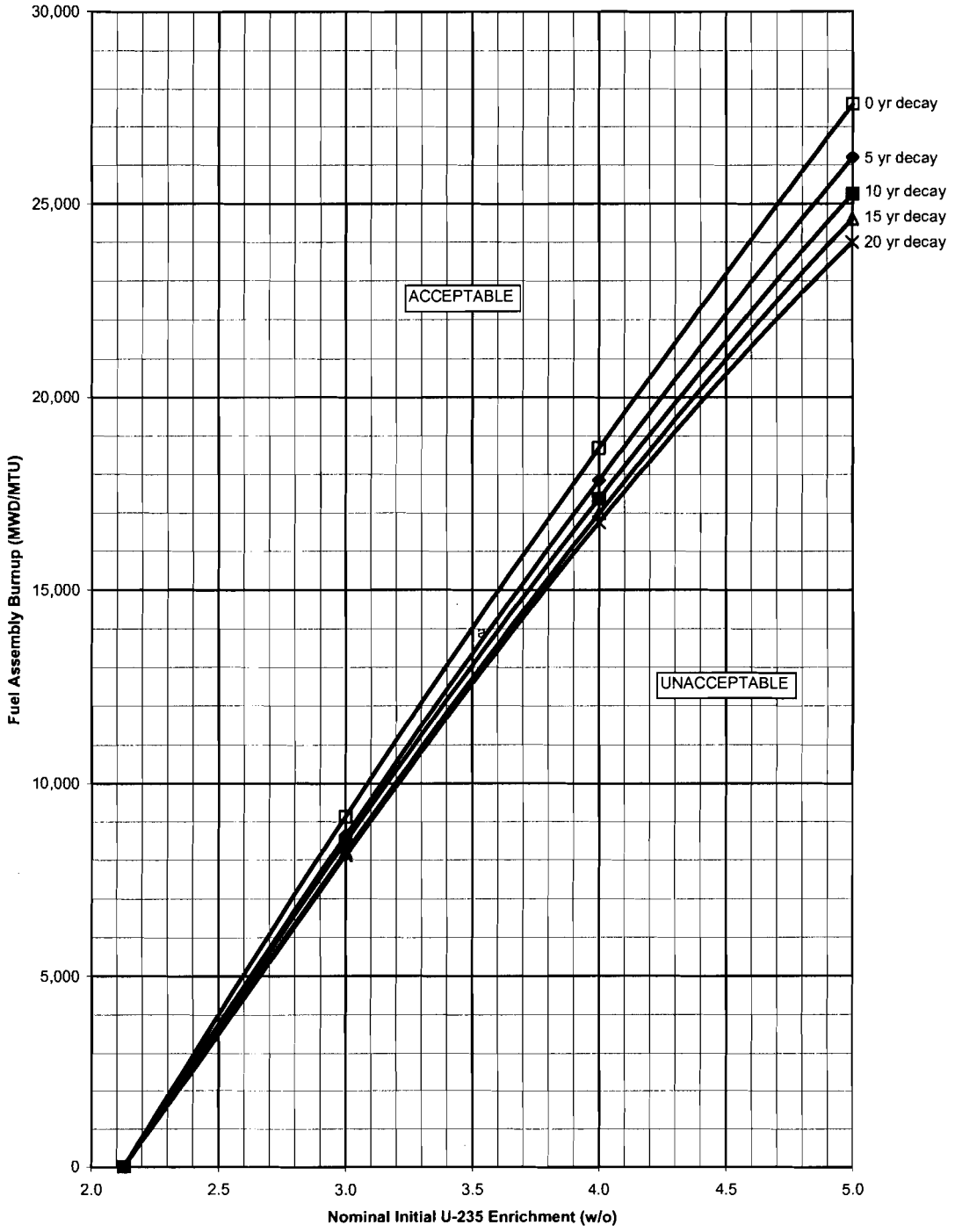
APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

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 =	$71.56e^3 - 1166.69e^2 + 15059.19e - 27474.43$
5 yr decay =	$-11.14e^3 - 287.32e^2 + 11630.42e - 23361.60$
10 yr decay=	$-16.28e^3 - 308.68e^2 + 11655.01e - 23267.42$
15 yr decay=	$-85.22e^3 + 436.68e^2 + 8884.52e - 20081.62$
20 yr decay=	$-113.81e^3 + 669.67e^2 + 8161.13e - 19321.65$

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

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 402 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.

4.3.2 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.

REFERENCES

1. "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," WCAP-16541-P, Revision 2 Westinghouse Electric Company, June 2008.

2. "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis - Addendum," WCAP-16541-NP, Revision 2, Addendum 1, Westinghouse Electric Company, November 2009.

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

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

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

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

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

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

No restriction on burnup.

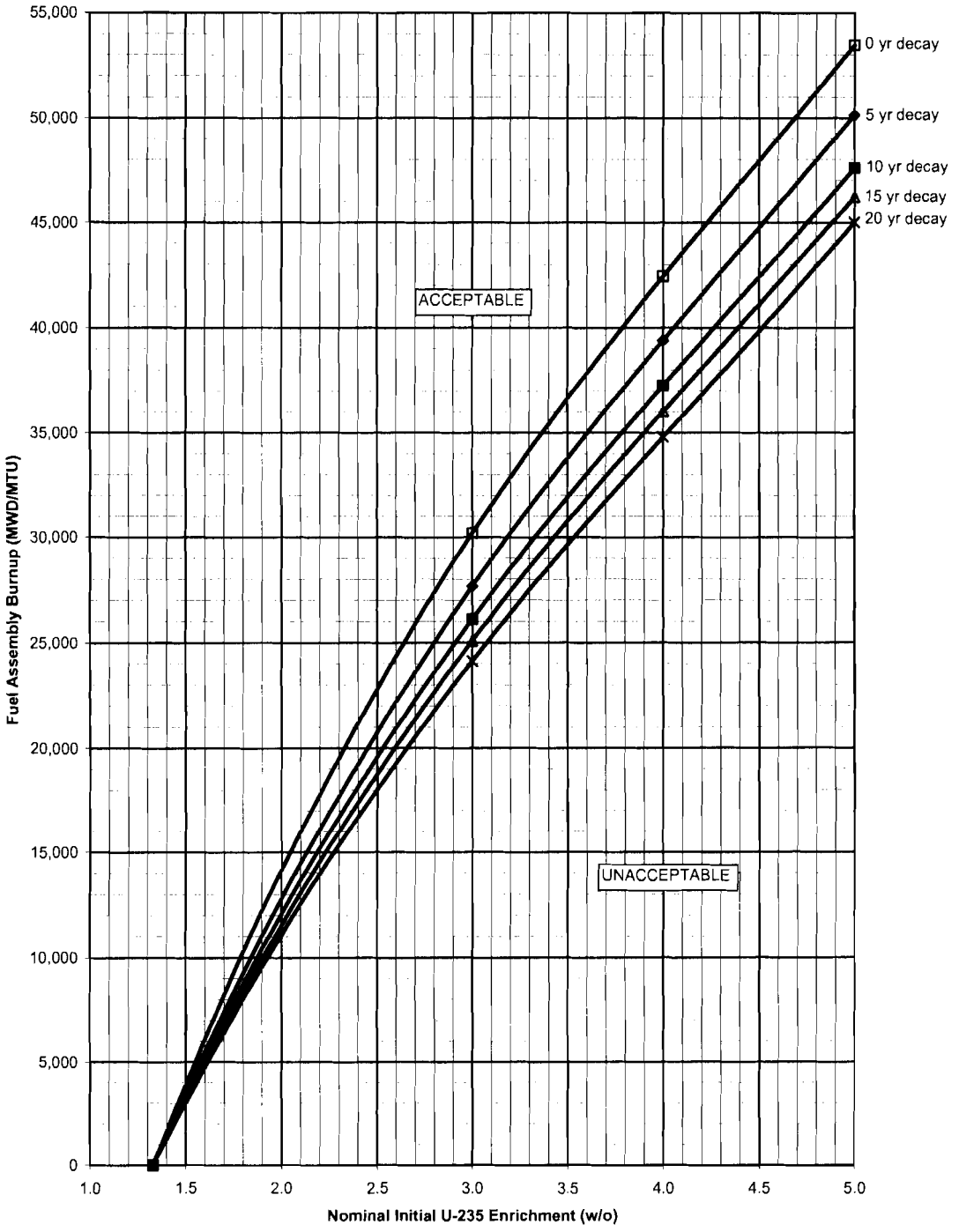
L1: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-6.

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.

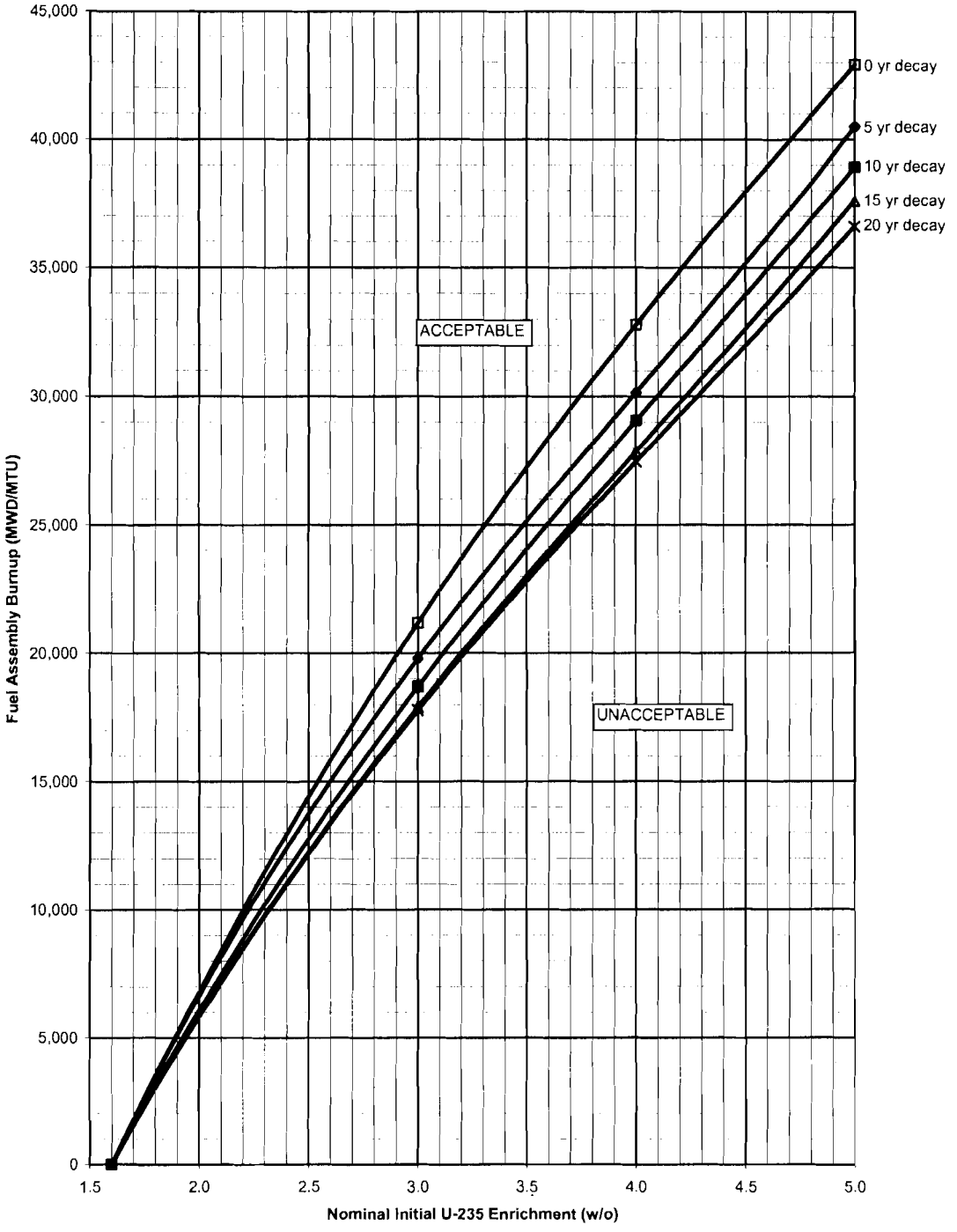
L2: Spent fuel assemblies in the "Acceptable" range of Figure 4.3.1-7.

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



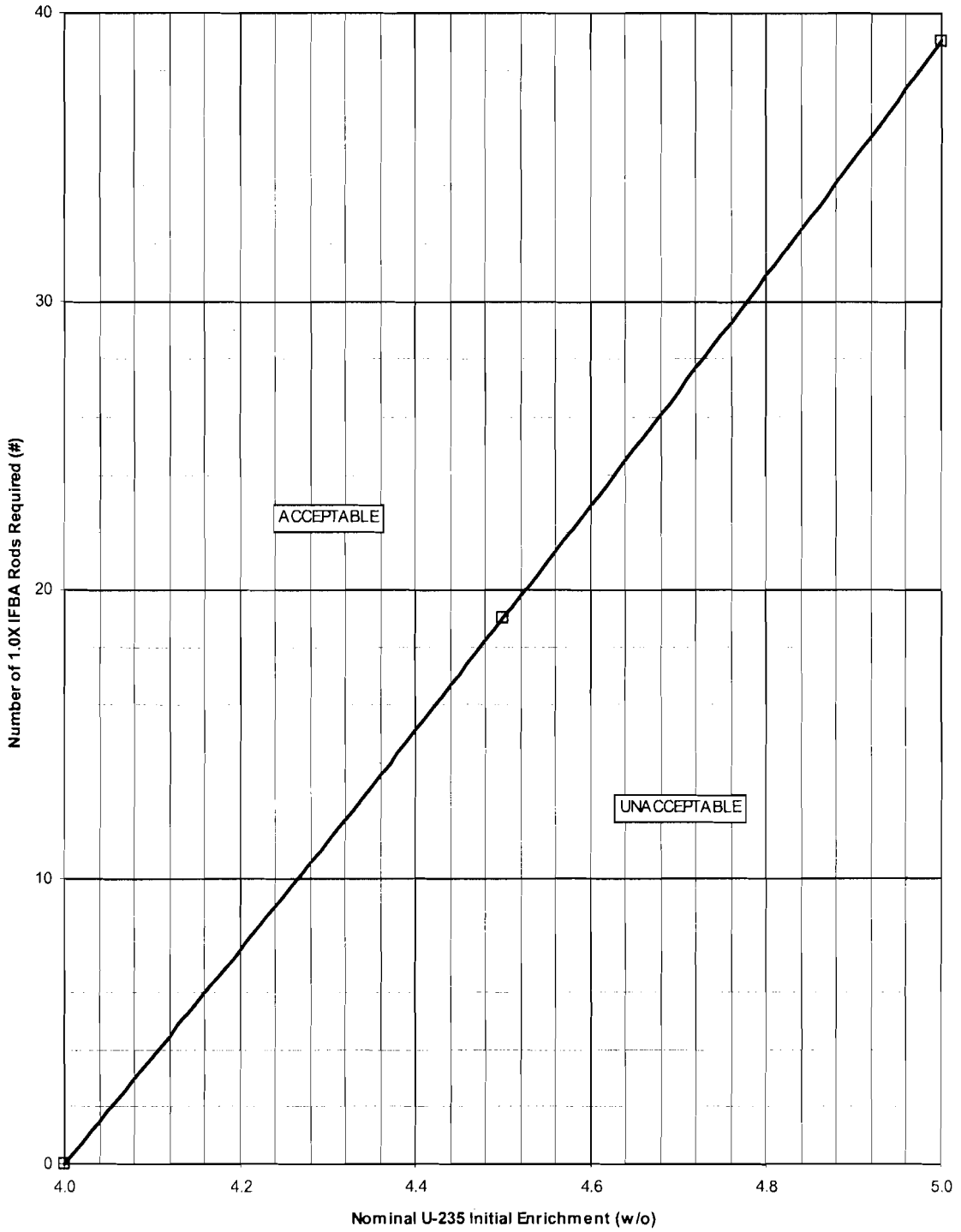
0 yr decay = $428.92e^3 - 5757.04e^2 + 36677.22e - 39606.18$
5 yr decay = $354.71e^3 - 4760.55e^2 + 31941.49e - 34895.75$
10 yr decay= $346.35e^3 - 4566.17e^2 + 30296.34e - 33031.86$
15 yr decay= $321.26e^3 - 4212.08e^2 + 28522.05e - 31239.39$
20 yr decay= $326.36e^3 - 4138.31e^2 + 27551.88e - 30091.55$

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



0 yr decay = $212.05e^3 - 3290.58e^2 + 26798.27e - 35321.90$
5 yr decay = $461.10e^3 - 5538.19e^2 + 32070.67e - 39023.97$
10 yr decay= $290.71e^3 - 3744.49e^2 + 25822.24e - 32920.44$
15 yr decay= $316.33e^3 - 3898.92e^2 + 25546.37e - 32188.62$
20 yr decay= $286.15e^3 - 3716.74e^2 + 25117.81e - 31845.90$

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



Number of IFBA Pins = $2e^2 + 21e - 116$

Figure 4.3.1-8
Fresh Fuel IFBA Requirements

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 236 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 240 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-27

FPL ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, Commission) dated July 24, 2008, as supplemented by letters dated September 19, 2008, April 14, May 22, August 7, August 27, November 20, 2009, and February 2, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML082240685, ML082630114, ML091050499 ML091420436, ML092220273, ML092400262, ML093270080, and ML100331643), FPL Energy Point Beach, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Point Beach Nuclear Plant (PBNP), Units 1 and 2. The supplements dated September 19, 2008, April 14, May 22, August 7, August 27, November 20, 2009, and February 2, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 9, 2008 (73 FR 74759).

The proposed changes would revise the PBNP licensing basis to reflect a revision to the spent fuel pool (SFP) criticality analysis methodology. The revised SFP criticality analysis credits burnup, integral fuel burnable absorber (IFBA), Plutonium-241 (^{241}Pu) decay, and soluble boron, where applicable. The proposed license amendment request (LAR) creates three 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." Each storage configuration has a geometric arrangement which must be maintained so that the SFP criticality analysis remains valid. The storage configurations may be interspersed with each other throughout the SFP, provided that the interface requirements are met. Each storage configuration has a burnup verses enrichment requirement that must be met for a fuel assembly to be stored in that configuration. The changes to TS 3.7.12, "Spent Fuel Pool Storage," and 4.3.1, "Criticality," impose new storage requirements reflecting the new SFP criticality analysis. Upon approval of these changes, Boraflex will no longer be credited in the criticality analysis and therefore the Boraflex Surveillance Program will be discontinued.

Currently, the licensee is committed to periodic sampling of the Boraflex in the PBNP spent fuel pool by commitments to NRC Generic Letter (GL) 96-04 in letter dated October 23, 1996 and for the period of extended operation as explained in the letter from FPL Energy to the NRC dated April 1, 2005 (ML051020357) as part of their license renewal application. Since the current request does not include credit for Boraflex, these commitments will no longer exist upon approval of the proposed request.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 68(b)(4) requires, in part, that:

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

In order to take credit for soluble boron, the licensee must show that k_{eff} remains less than 0.95 when the spent fuel pool is flooded with borated water. The amendment credits a concentration of 664 parts per million (ppm) boron in an accident scenario. The TS required minimum concentration of boron in the spent fuel pool water is 2100 ppm. The licensee has performed a boron dilution analysis to provide confidence that adequate margin exists to preclude inadvertent dilution of the spent fuel pool below 664 ppm boron. The NRC staff has reviewed this analysis and assumptions made therein to determine if there are sufficient measures to detect and stop any dilution event that could reduce the spent fuel pool boron concentration below the amount credited in this amendment.

10 CFR Part 50 Appendix A Criterion 62, "Prevention of criticality in fuel storage and handling," requires that:

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

10 CFR Part 50.68(b)(1) requires that:

> 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 Part 50.36(c)(4) requires that:

> Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

The NRC staff issued an internal memorandum on August 19, 1998, containing guidance for performing the review of SFP criticality analysis (Reference 1). This memorandum is known as the "Kopp Letter," after the author. The Kopp Letter provides guidance on salient aspects of a criticality analysis. The NRC staff used the Kopp Letter as guidance for the review of the LAR. This document is publicly available and was used as an exhibit before the Atomic Licensing Review Board for contention in July of 2000.

The NRC staff reviewed the licensee's request dated July 24, 2008, using NUREG-0800, "Standard Review Plan, Chapter 18.0, Human Factors Engineering," Revision 1.

Shrinkage, gap formation, and dissolution of the Boraflex poison material in the spent fuel racks is a phenomenon addressed in several generic communications from the NRC staff. In GL 96-04, the NRC staff requested that all licensees of power reactors with installed racks containing Boraflex provide an assessment of the physical condition of the Boraflex and state whether a subcritical margin of 5 percent can be maintained for the racks in unborated water. In addition, the licensees were requested to submit a description of any proposed actions to monitor or confirm that this subcriticality margin can be maintained for the lifetime of the storage racks and to describe any corrective actions in the event that it cannot be maintained. The descriptions of the GL 96-04 commitments for PBNP are provided in a letter from Wisconsin Electric to the NRC dated October 23, 1996. The current PBNP Boraflex Surveillance Program provides for condition monitoring of the Boraflex through inspection of full length Boraflex panels in five year intervals, measurement of the boron-10 (^{10}B) areal density of the Boraflex, and corrective action measures. The program tests for gap formations in the panels that experience accelerated exposures due to their relative location in the pool. This would provide sufficient time to perform corrective actions before the majority of the pool experienced the same degradation. The method used by the licensee to assess the condition of the Boraflex is neutron attenuation testing. This procedure confirms the physical presence of the Boraflex panels in terms of gap formation, gap distribution, and gap growth. The licensee also has license renewal commitments to perform the Boraflex Surveillance Program. The commitments include:

1. Certain accelerated Boraflex panels will be areal density and blackness tested every two years during the period of extended operation.
2. The first Boraflex areal density testing of the Boraflex panels will be performed prior to the period of extended operation.
3. A new procedure to schedule and perform Boraflex areal density and blackness testing will be created.
4. 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 percent 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.
5. Corrective actions will be taken to ensure that the 5 percent 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 percent subcriticality margin cannot be maintained because of current or projected future degradation. Corrective actions may include, but are not necessarily limited to, reanalysis, repair and/or replacement.

3.0 TECHNICAL EVALUATION

3.1 Spent Fuel Pool Criticality Analysis

Currently, there is not a generically approved methodology for performing SFP criticality analysis. Therefore, the licensee must submit a plant-specific SFP criticality analysis which meets regulatory requirements and includes technically supported margins (WCAP-16541-P, Rev. 2, "Point Beach Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis"). The NRC staff reviewed the analysis to ensure that the assumptions made, both stated and unstated, are technically substantiated. Assumptions affecting the neutron multiplication factor, k_{eff}, should generally be conservative. Non-conservative assumptions may be acceptable if they are quantified and are reasonably offset by other conservative assumptions. The NRC staff reviewed the application and issued requests for additional information (RAIs) to conclude with reasonable assurance that the regulatory requirements (10 CFR 68(b)(4) states a 95 percent probability, 95 percent confidence level) will be met.

The licensee's criticality analysis approach aims to maintain an acceptance criterion of k_{eff} <0.995 if flooded with unborated water, reserving 0.005 Δk_{eff} analytical margin to the regulatory requirement. While this acceptance criterion is not a regulatory requirement, it aids the NRC staff in reaching a reasonable assurance determination. Intact margin may allow the NRC staff to preclude the need to substantiate every detail of the criticality analysis and accept certain qualitative arguments.

Issues identified during the review reduced the licensee's reserved analytical margin from 0.005 Δk_{eff} to 0.00049 Δk_{eff}, resulting in a maximum k_{eff} of 0.99951. The analytical margin is normally reserved to address potential uncertainties that are unaccounted for or are unknown. The reserved analytical margin of 0.00049 Δk_{eff} essentially required that there were no other uncertainties needed to be accounted for and that the remaining portions of the analysis were without issues. With available margin so small, issues that may be otherwise resolved by engineering judgment become more important to consider and quantify. Issues such as the effect of not accounting for fission products and actinides in the criticality code validation become important to address. As a result, the review required additional rounds of RAIs.

When a potential non-conservatism was identified, the licensee generally responded by identifying offsetting conservatisms in the analysis or by crediting, and therefore, reducing, the reserved analytical margin. While this approach may be valid, attempts to remove conservatisms or reduce margin usually lead to lengthier review, especially in the absence of generically approved analytical techniques. The NRC staff requested the licensee to provide quantitative justifications. In certain cases, the response resulted in more questions. The review also required tracking of credited conservatisms. For example, the pellet dishing and chamfering that were conservatively ignored in the original submittal were later explicitly modeled. Axial blankets that were initially ignored were later explicitly modeled. The analysis initially used the bounding value of 97.5 percent theoretical density (TD) for the uranium dioxide (UO_2) density which was later reduced. The constant boron concentration assumed in the depletion calculation was later changed to model the representative let-down curve. The submittal had to be re-reviewed to ensure that these changes in assumptions would not invalidate other areas of the submittal.

In an attempt to address the difficulties described above, the licensee, in letter dated November 20, 2009, submitted a supplement restoring the full 0.005 Δk_{eff} analytical margin for the proposed storage configurations. The licensee revised the loading curves to require additional burnup that would provide the necessary negative reactivity to offset the margin previously lost. Based on the conservatively revised loading requirements in conjunction with the discussions in this safety evaluation, the NRC staff finds reasonable assurance that PBNP will comply with the regulatory requirements.

3.1.1 Code Validation

The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainties (Δk_{bias} and σ_m) are determined for use in the criticality calculation. The NRC staff expects code validation to be consistent with established standards for out-of-reactor criticality safety analyses. Standards require comparison of predicted verses experimental data to obtain the bias and bias uncertainty.

NUREG/CR-6698 states that:

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A sufficient number of experiments with varying experimental parameters should be selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (Reference 2), as guidance for review of the code validation methodology presented in the application. NUREG/CR-6698 outlines the basic elements of validation, including identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the area of applicability.

For the criticality calculation, the licensee used SCALE-PC, a personal computer version of the SCALE-4.4a code system with the updated 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross section library. SCALE-PC, used in both the code benchmark analysis and the fuel storage analysis, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a (KENO). For the depletion calculation to determine the isotopics, the licensee used the two-dimensional PHOENIX-P code with an Evaluated Nuclear Data File, Version 6 (ENDF/B-VI) neutron cross section library. PHOENIX-P was not used for any fuel storage rack eigenvalue calculations, including uncertainty determination.

3.1.1.1 SCALE-PC

The criticality code validation of SCALE-PC is based on a benchmark analysis of 30 selected critical experiments from two experimental programs. Nineteen benchmarks were selected from the Babcock & Wilcox experiments in support of Close Proximity Storage of Power Reactor Fuel (Reference 3) and eleven benchmarks were selected from the Pacific Northwest Laboratory

(PNL) Program in support of the design of Fuel Shipping and Storage Configurations (Reference 4).

In response to NRC staff RAIs (References 5 and 6), the licensee provide additional information regarding the code validation of SCALE-PC. The licensee identified the applicable operating conditions for the validation (e.g., fissile isotope, enrichment of fissile isotope, fuel chemical form, types of neutron absorbers, moderators and reflectors, range of moderator to fissile isotope, and physical configurations). The licensee compared the spectral parameters (e.g., EALF, H/U) between the benchmarks and the PBNP SFP conditions to demonstrate that the selected benchmarks are applicable. The licensee stated that the Shapiro-Wilks test for normality shows that the benchmark k_{eff} results are normally distributed. The licensee also verified that no trends exist.

Based on above, the NRC staff finds that the bias and bias uncertainties (Δk_{bias} and σ_m) determined for SCALE-PC are acceptable for PBNP.

3.1.1.2 PHOENIX-P

The licensee used the PHOENIX-P code to perform the fuel depletion analysis. PHOENIX-P performs a heterogeneous multi-group transport calculation for an explicit representation of a fuel assembly to determine the isotopic composition of the spent fuel as a function of fuel burnup and initial feed enrichment. The uncertainty in the k_{eff} calculation introduced by the PHOENIX-P depletion calculation is addressed by applying the appropriate depletion uncertainty in the determination of the maximum k_{eff}. In response to an NRC staff RAI, the licensee identified the appropriate depletion uncertainty treatment to be consistent with the NRC staff guidance (Reference 1). Therefore, the NRC staff accepts the use of the PHOENIX-P code to determine the isotopics for the subsequent criticality analysis. Section 3.1.5 provides additional evaluation of the burnup uncertainty.

3.1.2 Selection of Bounding Assembly Design

The criticality analysis should be based on the fuel assembly design that results in the highest calculated reactivity. The PBNP SFP contains the 14x14 Standard, Optimized Fuel Assembly (OFA), and 422V+ fuel assemblies. The licensee selected the Standard design to model the spent fuel and the OFA design to model the fresh fuel in the PBNP SFP.

The licensee states that the Standard fuel design bounds the 422V+ fuel. The Standard fuel assembly is 0.75 inches longer than 422V+ and uses Zirc-4 as the cladding material, which is less absorbent than the ZIRLO material used by 422V+. In addition, the 422V+ design contains low-enriched axial fuel blankets, and the Standard fuel design contained active fuel to the top and bottom of the fuel pellet stack. Based on the noted design differences, the NRC staff accepts the use of Standard design to model 422V+.

The licensee initially assumed that the Standard design is bounding for spent fuel and the OFA design is bounding for fresh fuel in the PBNP SFP. The NRC staff finds that the submittal should demonstrate that the selected fuel assembly design would be bounding for all anticipated storage configurations and burnup and enrichment combinations. In response to an NRC staff RAI (Reference 5), the licensee provided explicit calculations to justify the use of the Standard design for spent fuel and OFA fuel for new fuel. The licensee determined the eigenvalue

differences between the Standard and OFA fuel assemblies, $\Delta k = k_{STD} - k_{OFA}$, at 3.0 wt%, 4.0 wt%, and 5.0 wt% enrichment and from 0 to 55,000 Megawatt-Days per Metric Ton (MWD/MTU) in 5000 MWD/MTU increments. The results show that the Δk's are positive, except for the 5.0 percent enrichment cases at burnup less than 15,000 MWD/MTU. However, with 5.0 wt% initial enrichment, the burnup values used to determine the loading requirements are significantly above 15,000 MWD/MTU for all storage configurations. Therefore, the NRC staff finds that the assumption to use the Standard design to model spent fuel is acceptable.

The licensee's calculation showed that at zero burnup, the OFA design was more reactive with enrichment of 5.0 wt%, but the Standard design was slightly more reactive with enrichment of 4.0 wt%. To justify the use of OFA design for fresh fuel, the licensee compared the Standard and OFA fuel designs considering the 1 out of 4 storage configuration. To span the range of burnup/enrichment combinations, 1.6 wt% at 0 burnup and 5.0 wt% at 55,000 MWD/MTU burnup were considered for the other 3 assemblies. The results show that the OFA design is notably more reactive for both burnup/enrichment combinations. Therefore, the NRC staff finds that it is reasonable to assume the OFA design to model the fresh fuel.

3.1.3 Manufacturing Tolerances

The manufacturing tolerances of the storage racks and fuel assemblies contribute to the reactivity. The Kopp Letter states that:

> An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k-eff or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations.

The licensee states that, "[m]anufacturing tolerances that have a statistically significant effect on the calculated eigenvalue and a physical basis are included in the uncertainty rackup." The analysis considered the following uncertainty components: cell pitch, rack thickness, cell internal dimension, UO_2 density, IFBA ^{10}B loading, fuel enrichment, and clad outer diameter and thickness. The analysis also considered the spacer grids and showed that neglecting them was conservative.

The analysis initially used the bounding value of 97.5 percent TD for the UO_2 density. This value was later reduced to 96.5 percent TD to show the applicability of the analysis to the peripheral power suppression assemblies (PPSAs). An explicit uncertainty was included in the maximum k_{eff} calculation in conjunction with the use of 96.5 percent TD. The licensee stated that 96.5 percent TD is the manufacturing process maximum and the pellet density at PBNP is less than this value. IFBA ^{10}B loading was conservatively reduced to account for the manufacturing and calculational uncertainties. The reactivity effects of the other uncertainty components were determined by comparing the nominal case in which all parameters were defined at their nominal values to a perturbed case where parameter *i* was perturbed by the manufacturing tolerance. All tolerance perturbations were applied in the direction that increases reactivity relative to the nominal condition. If the tolerance perturbation resulted in a decrease in reactivity relative to the nominal condition, the reactivity effect for that tolerance was ignored. The Monte Carlo uncertainties were summed and conservatively added to the component uncertainty. In the

initial submittal, the licensee calculated the uncertainties at a single burnup/enrichment combination for a given storage configuration. In response to NRC staff RAI (Reference 5), the licensee recalculated the uncertainties at additional burnup and enrichment combinations to show that the sum of bias and uncertainties reported in WCAP-16541 is conservative.

In response to an NRC staff RAI (Reference 5), the licensee provided information showing the effect of 648 ppm of soluble boron on the biases and uncertainties. A concentration of 648 ppm is the required to meet the regulatory requirements following the limiting accident, per WCAP-16541. The results show an increase in the sum of biases and uncertainties for the "All-Cell" and the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configurations and a decrease for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration. The NRC staff finds the results acceptable based on the available margin to the Technical Specifications minimum boron concentration requirement of 2100 ppm.

3.1.4 SFP Temperature Bias

The Kopp Letter states that the criticality analysis should assume the temperature corresponding to the highest reactivity. If the SFP has a positive moderator temperature coefficient, the temperature corresponding to the highest reactivity would be the highest allowed operating temperature. The licensee performed the analysis at a nominal temperature and then determined a temperature bias to the limiting temperature.

In response to an NRC staff RAI (Reference 7), the licensee provided additional information describing how a conservative temperature bias was determined for each storage configuration. The reactivity increase caused by the temperature difference is determined by comparing the 68 °F case to the 180 °F case. As in the manufacturing tolerance analysis, the Monte Carlo uncertainties were summed and conservatively added to the bias.

The licensee performed explicit calculations in the infinite array models for each storage configuration. For initial enrichments of 3.0, 4.0, and 5.0 wt%, calculations were performed above and below the minimum allowable burnup requirement. The largest bias from the six cases was selected as the temperature bias for the storage configuration at all conditions. The NRC staff accepts the licensee's conservative treatment of the temperature bias.

In response to NRC staff RAI (Reference 5), the licensee provided information showing the effect of 648 ppm of soluble boron on the biases and uncertainties. 648 ppm is the required concentration to meet the regulatory requirements following the limiting accident, per WCAP-16541. The results show an increase in the sum of biases and uncertainties for the "All-Cell" and the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configurations and a decrease for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration. The NRC staff finds the results acceptable based on the available margin to the Technical Specifications minimum boron concentration requirement of 2100 ppm.

3.1.5 Spent Fuel Characterization

To take credit for the reduction in reactivity due to fuel burnup, the spent fuel must be properly characterized with conservative burnup uncertainty, axial burnup profile, and core depletion. The adequacy of the application with respect to each element is discussed.

3.1.5.1 Burnup uncertainty

The Kopp Letter states that:

> A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

The 5 percent reactivity decrement has been used throughout the industry since the issuance of the Kopp Letter. Rather than use the 5 percent reactivity decrement as the burnup uncertainty, the WCAP-16541 analysis used an unapproved technique without sufficient justification. Any deviation from the NRC staff guidance should at least include an appropriate comparison of the predicted isotopic concentrations to actual measured isotopic concentrations.

In response to an NRC staff RAI (Reference 7), the licensee recalculated the burnup uncertainty in accordance with the NRC staff guidance in the Kopp Letter. The burnup uncertainty for each storage configuration has increased. While the burnup uncertainty is included in the statistical combination by taking the square root of the sum of the squares, the burnup uncertainty is the largest uncertainty component and the increase makes a significant impact on the total uncertainty value. The recalculation of the burnup uncertainty resulted in a reduction of the reserved 0.005 Δk_{eff} analytical margin by 0.00131 Δk_{eff} for the "All-Cell" storage configuration, 0.00302 Δk_{eff} for the "1-out-of-4 5.0 w/o Fresh with no IFBA" storage configuration, and 0.00203 Δk_{eff} for the "1-out-of-4 4.0 w/o Fresh with IFBA" storage configuration. The NRC staff finds that the appropriate burnup uncertainties have been incorporated into the maximum k_{eff}.

In its November 20, 2009, letter, the licensee submitted a supplement restoring the full 0.005 Δk_{eff} analytical margin for the proposed storage configurations. The licensee revised the loading curves to require additional burnup that would provide the necessary negative reactivity to offset the margin lost.

3.1.5.2 Burnup Profile

Another important aspect of the spent fuel characterization is the selection of the burnup profile. At the beginning of life, a pressurized-water reactor fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis has shown that this results in an under prediction of k_{eff} and, generally, the under prediction becomes larger as burnup increases. The difference in the k_{eff} between a calculation with explicit representation of the axially distributed burnup and a calculation that assumes an axially uniform burnup is known as the end effect. Judicious selection of the axial burnup profile is necessary to ensure k_{eff} is not under predicted due to the end effect.

WCAP-16541 used the limiting profile from the R. E. Ginna Nuclear Plant LAR (Reference 8), citing the similarities in fuel design and reactor core type (i.e., both plants are Westinghouse 2-

loop reactors with 14x14 fuel lattice). To justify its applicability to PBNP, the licensee used a survey of 14 cycles of recent operation from both units to justify the selected distributed profile. The NRC staff finds that the assumed distributed profile is justified for PBNP.

To ensure that all discharged fuel assemblies are conservatively represented in the PBNP SFP, assemblies with uniform axial burnup profile are also considered. For each calculation, both uniform and distributed profiles were run and the most reactive representation was used to determine the minimum acceptable burnup requirement for safe storage. The NRC staff concludes that the analysis properly accounted for the effects of axial burnup profile for PBNP.

3.1.5.3 Core Depletion

The spent fuel model in the criticality analysis should be based on isotopics generated by bounding depletion parameters. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel" (Reference 9), discusses the treatment of depletion parameters. While NUREG/CR-6665 is focused on criticality analysis in storage and transportation casks, the basic principals with respect to the depletion analysis apply generically to SFPs, since the phenomena occur in the reactor as the fuel is being used. The basic premise is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: (1) fuel temperature, (2) moderator temperature, (3) soluble boron, (4) specific power and operating history, (5) fixed burnable poisons, and (6) integral burnable poisons.

For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize ^{241}Pu production. The WCAP-16541 analysis used the maximum temperatures for the fuel and moderator corresponding to Extended Power Uprate operating conditions. Therefore, the moderator and fuel temperature used is acceptable. For boron concentration, NUREG/CR-6665 recommends using a conservative cycle average boron concentration. The licensee's analysis used a conservative boron concentration throughout the depletion of the fuel assemblies. The licensee states that a review of recent cycles has confirmed that the boron concentration used is conservative relative to operating experience. Therefore, the assumed boron concentration is acceptable.

For specific power and operating history, NUREG/CR-6665 does not have a specific recommendation. NUREG/CR-6665 estimated this effect to be about 0.002 Δk_{eff} using operating histories it considered. Based on the difficulty of reproducing a bounding or even a representative power operating history, NUREG/CR-6665 merely recommends using a constant power level and retaining sufficient margin to cover the potential effect of a more limiting power history. The licensee used a constant core power for the depletion calculations. The licensee's revised analysis maintained 0.005 Δk_{eff} of reserved analytical margin to the regulatory limit and used a constant core power for the depletion calculations. The NRC staff concludes that the licensee's treatment of specific power and operating history is consistent with NUREG/CR-6665 and, therefore, is acceptable.

PBNP has used two different types of fixed poisons: glass burnable absorbers and hafnium flux suppression assemblies. The use of glass burnable absorbers at the PBNP has been discontinued, but these assemblies are present in the SFP. The licensee argues that the core operating temperatures and soluble boron concentrations in these past cycles were lower. The

licensee states that the core outlet temperature was approximately 15°F lower and the average soluble concentration was more than 200 ppm lower than that used in the analysis presented in WCAP-16541. These two effects provide approximately 0.005 Δk_{eff} and 0.006 Δk_{eff}, respectively, based on sensitivities provided in NUREG-6665. The increased reactivity caused by the presence of 20 burnable poison rods is approximately 0.006 Δk_{eff}. This bounds the PBNP's 14x14 lattice which only has 16 guide tubes available. The licensee concludes that either the temperature or boron concentration effects alone would be sufficient to provide the necessary margin relative to explicit consideration of glass burnable absorbers. The NRC staff finds this assertion to be reasonable.

Some assemblies in the PBNP SFP were operated in locations with hafnium flux suppression assemblies in the lower or central six feet of the assembly height to reduce vessel fluence. The licensee states that the use of these assemblies has been discontinued in Unit 1 and is likely to be discontinued in the future in Unit 2. From a criticality safety perspective, the concern posed by the peripheral power suppression assemblies (PPSAs) is the potential for increased reactivity at assembly discharge caused by water displacement or other spectral hardening mechanisms present during depletion. In Reference 10, the licensee provided quantitative information showing that the analysis bounds the storage of PPSAs.

PBNP uses IFBA rods. NUREG/CR-6760 (Reference 11) concluded that there is a positive reactivity effect associated with the depleting fuel with presence of IFBA rods. To determine the reactivity effect, the licensee in Reference 6, provided calculations for IFBA rod pattern used at PBNP for a range of burnup/enrichment included in the loading curve. The reactivity of the fuel assembly with IFBA rods was compared to the reactivity of the respective fuel assembly without IFBA rods. The largest difference was identified and deducted from the reserved analytical margin.

In its November 20, 2009, letter, the licensee submitted a supplement restoring the full 0.005 Δk_{eff} analytical margin for the proposed storage configurations. The licensee revised the loading curves to require additional burnup that would provide the necessary negative reactivity to offset the margin lost.

3.1.6 Determination of Soluble Boron Requirements

If soluble boron is credited, 10 CFR 50.68 requires that the k_{eff} 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.

The licensee's soluble boron methodology is based on calculating a boron worth curve, which is used to determine the amount of soluble boron necessary to address each term in the equation below. The licensee defines the total soluble boron requirement as:

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

where,

SBC_{TOTAL} is the total soluble boron requirement (ppm),
$SBC_{95/95}$ is the soluble boron requirement for 95/95 $k_{eff} \leq 0.95$ (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).

$SBC_{95/95}$ is the amount of soluble boron required to effect a 0.05 decrease in k_{eff} from an unborated condition. An unstated assumption regarding $SBC_{95/95}$ is that the storage configuration under the unborated condition already meets the regulation for unborated conditions which states, the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. Another unstated assumption is that the biases and uncertainties calculated for the unborated analysis do not need to be recalculated for borated conditions. The bias and uncertainties will change with soluble boron present in the water. In response to an NRC staff RAI (Reference 5), the licensee recalculated the biases and uncertainties under borated conditions. As discussed in Sections 3.3 and 3.4, the NRC staff finds that the licensee has addressed this effect appropriately.

The licensee assumes that the soluble boron worth from the one storage configuration is limiting with respect to the other three. The boron worth curve in the PBNP analysis is based on the "All-Cell" storage configuration containing depleted fuel at 55,000 MWD/MTU with 5.0 wt% ^{235}U initial enrichment. The initial enrichment and burnup combination selected minimizes the boron worth as boron worth decreases as burnup increases. The NRC staff finds that 55,000 MWD/MTU reasonably bounds the maximum burnup credited, and therefore provides some conservatism. While the licensee did not explicitly determine the limiting storage configuration relative to boron worth, the NRC staff accepts the use of "All-cell" based on this in conjunction with other identified margins with respect to borated conditions as discussed below.

SBC_{RE} is the soluble boron required to offset additional uncertainties associated with crediting boron from a zero soluble boron condition. The analysis identifies two uncertainties to be accommodated. The first is a depletion uncertainty equal to 0.01 Δk_{eff} per 30 GWD/MTU of burnup credited in the analysis. This value is calculated using the maximum burnup credited from any of the three storage configurations in the unborated analysis. In WCAP-16541, the maximum credited burnup was approximately 51 GWD/MTU for the "1 out-of-4 5.0 w/o Fresh with no IFBA" storage configuration. This uncertainty appears to be an adjunct to the burnup uncertainty used in the unborated analysis, providing some additional margin. The second is a burnup uncertainty equal to the largest burnup uncertainty calculated from any of the three storage configurations in the unborated analysis. This is in addition to a previously applied uncertainty, and therefore, provides additional margin.

SBC_{PA} is the soluble boron required to offset accident conditions. The analysis evaluated several potential off normal conditions for all three storage configurations in unborated water. The analysis determined that a misloading event in the "1-out-of-4 5.0 w/o with no IFBA" storage configuration would have the largest reactivity increase. The boron worth curve generated for the "All-Cell" storage configuration was used to determine the amount of soluble boron required to accommodate the reactivity increase associated with this misloading event, independent of any other soluble boron requirements. Based on WCAP-16541's parallel soluble boron accounting method and considering the largest reactivity increase due to an abnormal condition from the unborated condition, the analysis indicates PBNP requires 664 ppm of soluble boron to achieve a k_{eff} below 0.95 with a 95 percent probability at a 95 percent confidence level, under abnormal conditions. A serial accounting of soluble boron results in a much larger soluble boron requirement. In response to an NRC staff RAI (Reference 5), the licensee recalculated the

required boron concentration based on the "serial" accounting method. The licensee determined the total soluble boron (SBC_{TOTAL}) required to maintain k_{eff} below 0.95 with a 95 percent probability at a 95 percent confidence level, under abnormal conditions is approximately 818 ppm, not considering boron depletion. While application of the parallel accounting method is not as conservative as the serial method, the NRC staff considered the fact that the SFP is normally borated to at least 2100 ppm per TS requirements. Based on the large margin still available to the TS limit, the NRC staff accepts the licensee's determination of the soluble boron requirements.

3.1.7 Summary

The licensee credits soluble boron. Hence, the applicable regulatory requirement is taken from 10 CFR 50.68(b)(4), as stated in Section 2 of this safety evaluation. The NRC staff evaluated the submittal against the criteria for both unborated and borated conditions. The NRC staff reviewed the analysis to ensure that the assumptions and analytical technique used are adequately substantiated to conclude at a 95 percent probability, 95 percent confidence level, that the regulatory requirements will be met.

The licensee resolved the non-conservative assumptions in the analysis by restoring the 0.005 Δk_{eff} analytical margin for the proposed storage configurations. The licensee revised the loading curves to require additional burnup that would provide the necessary negative reactivity to offset the margin previously lost. Hence, the licensee's analysis retains a 0.005 Δk_{eff} analytical margin to the regulatory limits. Based on the conservatively revised loading requirements in conjunction with the discussions in Section 3 of this safety evaluation, the NRC staff finds reasonable assurance that PBNP will comply with the regulatory requirements. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

3.2 Boron Dilution Analysis

The licensee's boron dilution analysis determined the time to reduce the concentration of boron in the spent fuel pool from the minimum TS limit (2100 ppm) to the amount credited to maintain k_{eff} less than 0.95 in an accident scenario (664 ppm). This was done by finding the necessary volume of water to dilute the SFP volume to 664 ppm and then analyzing the rates at which water could be added from sources with flow paths to the SFP.

Assuming that the spent fuel racks were a solid block within the SFP, the volume of the pool was calculated to be 236,406 gallons. This is a conservative assumption because in actuality there would be a large volume of water flowing in and around the racks. The SFP volume is initially assumed to contain the minimum concentration of boron allowed by TS, 2100 ppm. Dilution of this volume of borated water would require the addition of 303,725 gallons of pure water. The total resulting volume of water is greater than the capacity of the SFP and transfer canal, meaning that the SFP would overflow onto the refueling floor and this water would be routed eventually to the waste holdup tank.

The licensee identified the following sources capable of adding water to the SFP: Chemical and Volume Control System (CVCS) Hold Up Tanks, Reactor Makeup Water (RMUW) Storage Tank, Demineralized (DI) Water System, Fire Protection System, and the Monitor Tanks. Of these sources, only the DI water system and fire protection system are capable of providing 303,725 gallons of water without manual actions to provide makeup water. In addition, manual

actions, controlled by administrative procedures, would be required to align flow paths from any of the tanks to the SFP. The staff reviewed the licensee's analysis of the CVCS hold up tanks, the RMUW storage tank, and the monitor tanks, and found their treatment to be conservative. These sources are not credible for independently resulting in a dilution of the SFP below the credited value of 664 ppm.

The fire protection system could provide the required dilution volume, but this system has no direct path to the SFP. Dilution of the SFP using the fire protection system would require unrolling a hose station and positioning the nozzle to the SFP. The licensee estimates that this flow path is capable of providing 100 gallons per minute (gpm) of water. This dilution method would require 994 minutes to reach the high level alarm setpoint, and 50 hours to add the required dilution volume.

The DI water system is the normal source of makeup water to the SFP, and is directly connected to the pool via a 2 inch line. The DI water is rated to provide water at 400 gpm; at this rate it would take 249 minutes to reach the high level alarm setpoint, and 12 hours to add the required dilution volume. Therefore addition of water from the DI water system is the bounding case for dilution of the SFP.

The staff reviewed the information contained in the final safety analysis report and the licensee's submittal, and found no other credible sources for adding sufficient quantities of water to the SFP. As shown by the licensee, the volume of water required to dilute the SFP to 664 ppm is greater than the capacity of the pool. Dilution by this source would result in a high level alarm as water level in the pool rose, as well as another high level alarm in the waste holdup tank as the pool continued to overflow. Even if operators failed to respond to alarms, 12 hours are needed to dilute the pool and licensee staff making rounds in the area would discover and report the condition. The NRC staff concludes that sufficient time exists to identify and suppress the bounding SFP dilution event, and there is confidence that the required concentration of boron will be maintained.

3.2.1 Summary

In consideration of the information discussed above, the NRC staff concludes that the proposed change in the PBNP licensing basis is acceptable.

3.3 Terminating Boraflex Monitoring

The NRC staff asked in an RAI letter dated March 16, 2009 for clarification on whether the Boraflex Surveillance Program would still be implemented upon approval of this request. The licensee responded in letters dated April 14, 2009 and May 22, 2009 that all commitments for a Boraflex Surveillance Program would be discontinued upon approval of this request.

The licensee states in the April 14, 2009 response that even though the Boraflex Surveillance Program would be discontinued, they would still monitor the SFP water for silica monthly. Also, PBNP has an operating experience program which would provide industry information for new Boraflex problems that may impede plant operations. The Boraflex poison panels are also vented which would allow gases to vent before warping would impact safe fuel handling.

Boraflex is used in the spent fuel storage racks for absorption of neutrons. Degradation of the Boraflex in the racks could result in an increase in the reactivity of the spent fuel configuration. The licensee justified their request to stop crediting Boraflex by performing criticality and accident analyses. The licensee will credit soluble boron and fuel placement methodology for SFP criticality control. These analyses were done in accordance with 10 CFR 50.68(b)(4) "Criticality accident requirements" acceptance criteria. This criterion allows the licensee to credit soluble boron to maintain the effective neutron multiplication factor of the SFP at 0.95 or less. The licensee's request, for which the analyses were performed, applies to the two SFP storage rack modules that contain Boraflex.

The NRC staff evaluated whether or not the changes proposed by the licensee are sufficient to stop crediting Boraflex. The proposed methodology for fuel placement in the SFP racks was found to be an acceptable way that the licensee could provide and maintain subcriticality in combination with the soluble boron. The results of these analyses, in which the NRC staff is basing its decision, were provided by Westinghouse. The proprietary license amendment report contained the acceptance criteria, assumptions made, design and input data, the methodology and the results which the NRC staff found to be in agreement with 10 CFR 50.68 requirements for the safe operation of the SFP. The report also included the calculations made to analyze possible risks and accidents that are associated with the proposed methods to control criticality in the SFP, i.e., a boron dilution accident. The NRC staff found that the results from the analyses assure that these probabilities are within the limits acceptable to the NRC. Therefore, the proposed changes are acceptable.

Since the proposed criticality analysis methodology does not take credit for Boraflex, the NRC staff concludes that the GL 96-04 commitments and license renewal commitments will no longer be necessary upon approval of the criticality analysis methodology proposed by PBNP Units 1 and 2.

3.3.1 Summary

The NRC staff has reviewed the impact of the licensee's request to credit soluble boron for reactivity control at PBNP on the GL 96-04 commitments and license renewal commitments to monitor Boraflex degradation for PBNP. The proposed criticality analysis methodology does not take credit for Boraflex; therefore, the NRC staff finds the proposed changes acceptable and concludes that the GL 96-04 commitments and license renewal commitments will no longer be necessary upon the approval of the criticality analysis methodology proposed for PBNP.

3.4 Effects on Human Performance

The staff reviewed PBNP's Updated Final Safety Analysis Report (UFSAR), Chapter 14, Section 14.2.1 "Fuel Handling Accident" to determine whether any actions credited in analyzed accident scenarios could be impacted by the proposed changes to the PBNP TS. Four accident scenarios are postulated:

1. A fuel assembly becomes stuck inside reactor vessel.
2. A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or SFP.
3. A fuel assembly becomes stuck in the penetration valve.
4. A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

None of the operator actions identified in these scenarios, which were previously approved by the NRC, will change due to the licensee's proposed changes to TS.

The NRC staff also reviewed WCAP-16541, to confirm that any actions credited in analyzed accident scenarios in that document were consistent with the proposed changes to the PBNP TS. The scenarios included:

1. Misloaded fresh fuel assembly into burnup storage rack location.
2. Misloaded fresh fuel assembly in the cask area between storage racks.
3. SFP temperature greater than normal operating range (240°F).

The NRC staff concluded that WCAP-16541 and the proposed TS are consistent regarding human actions. The physical actions identified in the WCAP scenarios do not change or affect existing fuel handling processes at PBNP, but the analysis does change the method of characterizing the fuel, i.e., for purposes of determining an appropriate position within the SFP (All-Cell or 1 out of 4 configuration). It should be noted that the licensee has identified scenario No. 1, "Misloaded fresh fuel assembly into burnup storage rack location," as the limiting accident for purposes of determining the required SFP boron concentration to mitigate postulated SFP criticality accidents. Per the WCAP, scenario #1 can be mitigated by a boron concentration of 256 ppm. After adding other sources in the SFP, as well as uncertainties, a concentration of 664 ppm was determined to be required to control k_{eff} less than or equal to 0.95 under accident conditions. The TS value for SFP boron concentration, which is confirmed every seven days via SR 3.7.11.1, is 2100 ppm. Consequently, the limiting misloading event is easily mitigated by the maintenance of the TS-required concentration of boron in the SFP.

The licensee states in its LAR that, "Fuel assembly burnup is a key input for determining how and where a fuel assembly may be stored in the SFP." Because the proposed fuel characterization process is somewhat more limiting and complex, the opportunity for cognitive errors may increase. Additionally, as the licensee states in the LAR, "Because the new criticality analysis will impose additional restrictions, some additional fuel moves to prepare for a refueling outage may be required." Therefore, the NRC staff concludes that there is some small increase in the probability of a misloading event, due to the increased complexity of the fuel characterization and the increased number of fuel moves. Further, if a fuel assembly is mischaracterized and not corrected by the independent verifier, there is also an additional, however small, probability that more than one fuel assembly could be misloaded. However, in addition to the large margin of safety afforded by maintaining SFP boron concentration at 2100 ppm, the licensee has established multiple barriers to prevent misloading events. Misloading events may be caused either by errors in identifying the characteristics of the fuel being moved (mischaracterizing), or by incorrectly moving the fuel (mispositioning).

The licensee has established barriers to prevent mischaracterizing fuel assemblies. Fuel assemblies are characterized using burnup values that are determined using software approved under the licensee's software quality assurance (SQA) program. The software is used only by qualified reactor engineers. Independent reviews of both input and output data are performed by qualified reactor engineers. Burnup data are input into software (ShuffleWorks) for planning fuel movements, and independently reviewed by qualified reactor engineers. The ShuffleWorks fuel movement planning software is approved under the licensee's SQA program. Updates to the software are controlled by procedure and require independent review. Fuel movement sequences are planned and independently reviewed by qualified reactor engineers. Reactor

engineers use an administrative procedure to verify that fuel assemblies meet the requirements of TS 3.7.12 prior to approving storage of fuel assemblies in the SFP.

The process of moving fuel is currently controlled by procedure and fuel movement authorizations. Qualified reactor engineers develop the fuel movement authorizations. All fuel movements are performed by qualified operators under the supervision of a Senior Reactor Operator (SRO). Only one fuel assembly is moved at a time. All fuel handling operations in the SFP are performed by an SRO and two fuel-handling-qualified operators who peer check for each other. An SRO and Reactor Engineer sign off for each move after it has been completed.

The fuel movement process will remain the same after the implementation of the proposed TS change. As stated by the licensee, "This application does not change or modify the fuel, fuel handling processes, spent fuel racks, the number of fuel assemblies that may be stored in the SFP, the decay heat generation rate or the SFP cooling and cleanup system."

After reviewing the submittal and the licensee's supporting documentation, the NRC staff has determined that the proposed change does not impact the current operator actions or the timing to perform those actions as credited in the PBNP UFSAR, and that the proposed TS change is acceptable in the area of human performance. Although the opportunity for error may increase slightly, the licensee has sufficient controls in place to prevent any inadvertent criticality accidents in the SFP due to human error.

3.4.1 Summary

Based on its review, the NRC staff concludes that the proposed TS change does not affect current operator manual actions credited in PBNP's UFSAR. The NRC staff also concludes that the proposed TS change is acceptable in the area of human performance due to the licensee's use of trained reactor engineers to characterize fuel assemblies and identify proper storage locations, and its use of procedures and administrative controls to perform fuel movement activities.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (73 FR 74759). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

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

7.0. REFERENCES

1. NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998. (ADAMS ML003728001)

2. NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," (ADAMS ML010170125)

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

4. NUREG/CR-1547, PNL-3314, "Critical Experiments with Subcritical Clusters of 2.35 Wt% 235U Enriched UO2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6," S. R. Bierman and E.D. Clayton, July 1980.

5. FPL Energy Point Beach Nuclear Plant letter NRC 2009-0037, Larry Meyer, Site Vice President, to U.S. NRC, "Supplement to License Amendment Request Number 247 Spent Fuel Pool Storage Criticality Control," April 14, 2009. (ADAMS Accession No. ML091050499))

6. NextEra Energy Point Beach letter NRC 2009-0057, Larry Meyer, Site Vice President, to U.S. NRC, "Supplement to License Amendment Request Number 247 Spent Fuel Pool Storage Criticality Control," May 22, 2009. (ADAMS Accession No. ML091420436)

7. FPL Energy Point Beach Nuclear Plant letter NRC 2008-0071, Larry Meyer, Site Vice President, to U.S. NRC, "Supplement to License Amendment Request Number 247 Spent Fuel Pool Storage Criticality Control," September 19, 2008. (ADAMS Accession No. ML082630114)

8. Letter, 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," December 7, 2000.

9. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (ADAMS ML003688150)

10. NextEra Energy Point Beach letter NRC 2009-0084, Larry Meyer, Site Vice President, to U.S. NRC, "Supplement to License Amendment Request Number 247 Spent Fuel Pool Storage Criticality Control," August 27, 2009. (ADAMS Accession No. ML092400262)

11. NUREG/CR-6760, " Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," (ADAMS ML020770436)

Principal Contributor: T. Nakanishi, NRR
E. Davidson, NRR
E. Wong, NRR
G. Lapinsky, NRR

Date: March 5, 2010

March 5, 2010

Mr. Larry Meyer
Site Vice President
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: SPENT FUEL POOL STORAGE CRITICALITY CONTROL (TAC NOS. MD9321 AND MD9322)

Dear Mr. Meyer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 236 to Renewed Facility Operating License No. DPR-24 and Amendment No. 240 to Renewed Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated July 24, 2008, as supplemented by letters dated September 19, 2008, April 14, May 22, August 7, August 27, November 20, 2009, and February 2, 2010.

These amendments revise the Point Beach Nuclear Plant licensing basis to reflect a revision to the spent fuel pool (SFP) criticality analysis methodology. The changes to TS 3.7.12, "Spent Fuel Pool Storage," and 4.3.1, "Criticality," impose new storage requirements reflecting the new SFP criticality analysis. Upon approval of these changes, Boraflex will no longer be credited in the criticality analysis and, therefore, the Boraflex Surveillance Program will be discontinued.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Justin C. Poole, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301
Enclosures:
1. Amendment No. 236 to DPR-24
2. Amendment No. 240 to DPR-27
3. Safety Evaluation
cc w/encls: Distribution via ListServ

DISTRIBUTION:

PUBLIC	LPL3-1 r/f	RidsNrrDorlLpl3-1Resource	EDavidson, NRR
RidsNrrPMPointBeach Resource		RidsNrrLABTully Resource	EWong, NRR
RidsOgcRp Resource		RidsAcrsAcnw_MailCTR Resource	GLapinsky, NRR
RidsNrrDirsItsb Resource		RidsRgn3MailCenter Resource	
RidsNrrDorlDpr Resource		TNakanishi, NRR	

Amendment Accession Number: ML100400106 *Minor editorial changes from staff provided SEs

OFFICE	LPL3-1/PM	LPL3-1/LA	SRXB/BC	SBPB/BC	CSGB/BC
NAME	JPoole	BTully	GCranston*	GCasto*	MGavrilas*
DATE	03/05/10	02/24/10	01/25/10	02/01/10	06/08/09
OFFICE	DIRS/IOLB	DIRS/ITSB	OGC	LPL3-1/BC	
NAME	NSalgado*	RElliott	MSmith (NLO)	RPascarelli	
DATE	01/12/09	03/05/10	03/04/10	03/05/10	

OFFICIAL RECORD COPY