NEXTera ENERGY POINT BEACH

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

Response to Request for Additional Information
License Amendment Request 247
Spent Fuel Pool Storage Criticality Control

References
(1) FPL Energy Point Beach Letter to NRC, License Amendment Request 247, Transmittal of Changes to Technical Specifications re: Spent Fuel Pool Storage Criticality Control, dated July 24, 2008 (ML082240685)
(2) FPL Energy Point Beach Letter to NRC, Supplement to License Amendment Request Number 247, Spent Fuel Pool Storage Criticality Control, dated September 19, 2008 (ML082630114)
(3) FPL Energy Point Beach Letter to NRC, Response to Request for Additional Information, License Amendment Request 247, Spent Fuel Pool Storage Criticality Control, dated April 14, 2009 (ML091050499)
(4) NRC letter to FPL Energy Point Beach, Point Beach Nuclear Plant, Units 1 and 2 - Request for Additional Information from Reactor Systems Branch Related to License Amendment Request No. 247 Spent Fuel Pool Storage Criticality Control, dated April 22, 2009 (TAC Nos. MD9321 and MD9322) (ML09090067)

NextEra Energy Point Beach, LLC (formerly known as FPL Energy Point Beach, LLC) submitted a proposed license amendment request for Commission review and approval pursuant to 10 CFR 50.90 for the Point Beach Nuclear Plant (PBNP), Units 1 and 2 (Reference 1). The proposed amendment revises the licensing basis to reflect a revision to the spent fuel pool (SFP) criticality analysis methodology. The revised criticality analysis for the SFP storage racks credits burnup, integral fuel burnable absorber (IFBA), Plutonium-241 decay, and soluble boron, where applicable. FPL Energy provided a supplemental response (Reference 2) containing additional quantitative information to support the fidelity of key methodology aspects described in Reference (1).

On April 14, 2009, a teleconference was held between NRC and NextEra Energy Point Beach (NextEra) personnel to discuss additional information that was requested by the Commission to enable further review of the application. During the teleconference, NextEra stated that the response to the request for additional information would be submitted within 30 days of receipt.

On May 7, 2009, a teleconference was held between NRC and NextEra personnel to discuss PBNP response to Question 1 of Reference (3) on the PBNP boraflex surveillance program.

During the teleconference it was agreed that NextEra would clarify the response and include it in the Reference 4 response to request for additional information.

Enclosure 1 of this letter provides the NextEra response to the request for additional information in Reference (4).

Enclosure 2 provides the clarifying information of the PBNP boraflex surveillance program.

This supplement does not affect the no significant hazards conclusion, as defined in 10 CFR 50.92, provided in Reference (1).

This letter contains no new commitments and no revisions to existing commitments.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing and enclosed information is true and correct. Executed on May 22, 2009.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

**ENCLOSURE 1**

**NEXTERA ENERGY POINT BEACH, LLC**
**POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2**

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**
**LICENSE AMENDMENT REQUEST 247**
**SPENT FUEL POOL STORAGE CRITICALITY CONTROL**

The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's request for additional information dated April 14, 2009.

**Question 1: Code Validation**

*Section 1.4.2 of Enclosure 6 of the letter dated July 24, 2008 (ADAMS Accession No. ML082240685), discusses the validation of the SCALE-PC code used in criticality calculations. To allow the NRC staff to evaluate the adequacy of the validation, please provide the following additional information:*

a. *Discuss and justify the method you used to select the benchmarks identified in Tables 1-1 and 1-2. For example, what parameters were considered to correlate the benchmarks to the systems being analyzed? What ranges were considered for those parameters?*
b. *Please provide additional details characterizing the benchmarks in terms of the parameters cited in Question 1a above, or submit References 9 through 12 of WCAP-16541-P, Revision 2. Currently, the submittal lacks sufficient information to evaluate the applicability of the benchmarks to the systems being analyzed.*
c. *Document and justify the area of applicability for the benchmarks.*
d. *Describe and justify any statistical and trending analyses performed to support the determination of the bias and bias uncertainty.*
e. *How did you account for the measurement uncertainties for the benchmarks?*

**NextEra Response**

Point Beach Nuclear Plant (PBNP) uses low enriched uranium fuel in a water moderated system at relatively low temperatures, with boron as an absorber. The primary structural material of the PBNP spent fuel pool racks is SS-304. The critical benchmarks have similar fuel, absorber, moderator, and structural materials. As outlined in NUREG/CR-6698, the important parameters for valid benchmarking are the fissile isotope, enrichment of the fissile isotope, fuel density, fuel chemical form, the types of neutron moderators and reflectors present, the ratio of moderator to fissile isotope, neutron absorbers, and physical configurations (i.e., geometry). Table 2.3 of NUREG/CR-6698 summarizes the most important parameters and gives guidance as to how to define areas of applicability.

NUREG/CR-6698 notes that, "perhaps the best source of critical benchmarks is found in the International Handbook of Evaluated Criticality Safety Benchmark Experiments for the Nuclear Energy Agency of the Organization for Economic Co-Operation and Development (OECD-NEA). The critical experiments described in this handbook have been found by the ANSI/ANS-8 Subcommittee for NCS to be rigorously peer reviewed and should be accepted as refereed..."

It is important to note that the 30 experiments used in the Benchmarking Suite presented in WCAP-16541-P, Revision 2, are included in the Handbook.

The benchmarks identified in Tables 1-1 and 1-2 of WCAP-16541-P, Revision 2, have been the standard set of benchmarks considered for code validation by Westinghouse for spent fuel pool criticality safety analysis for many years.

A comparison of relevant parameters is shown in Table 1. The first column gives a description of the parameter and the unit of measurement if applicable. The second column gives the range of each parameter for the benchmark experiments. The third column gives the range of each parameter as used in the PBNP analysis. The fourth column paraphrases the guidance given in Table 2.3 of NUREG/CR-6698 relating to the area of applicability for each parameter. The final column specifies if the parameter, as used in the PBNP analysis, is completely covered by the benchmark as defined by the area of applicability guidance. If the PBNP specific parameter is covered by the area of applicability as defined by Table 2.3 of NUREG/CR-6698, the final column will list "Yes" if any part of the PBNP parameter is not covered; it is addressed in the discussion below the table.

**Table 1 – Comparison of Select Parameters between Point Beach Analysis and Benchmark Suite**

| Parameter | Benchmark | Point Beach Analysis | NUREG/ CR-6698 Guidance | Applicable |
|---|---|---|---|---|
| Fissile Isotope | $^{235}U$ | $^{235}U$ | Same | Yes |
| Enrichment of Fissile Isotope (wt%) | 2.5, and 4.31 | 1.33 – 5.0 | 0-2w/o: ±1%<br>2-5w/o: ±1.5% | See Discussion |
| Fuel Chemical Form | $UO_2$ | $UO_2$ | Same or justified | Yes |
| Temperature (K) | 290 - 299 | 283 - 355 | ±50 | See Discussion |
| Neutron Moderators and Reflectors | $H_2O$ | $H_2O$ | Same or justified | Yes |
| Neutron Absorbers | Boron, SS-304 | Boron, SS-304 | Same or justified | Yes |
| Physical Configurations | Thin fuel rods in water | Thin fuel rods in water | As similar as possible | Yes |

For the lowest fresh enrichment credited in the WCAP-16541-P, Revision 2, analysis (1.33 wt% $^{235}U$), Table 2.3 of NUREG/CR-6698 suggests that the benchmark should include experiments with enrichments between 0.33 – 2.33 wt% $^{235}U$. However, the lowest enrichment in the benchmark suite is 2.5 wt% $^{235}U$.

The upper end of the temperature range is 355 K. Table 2.3 of NUREG/CR-6698 suggests that the benchmark should include experiments with temperatures between 305 – 405 K. The highest temperature in the benchmark suite is 299 K.

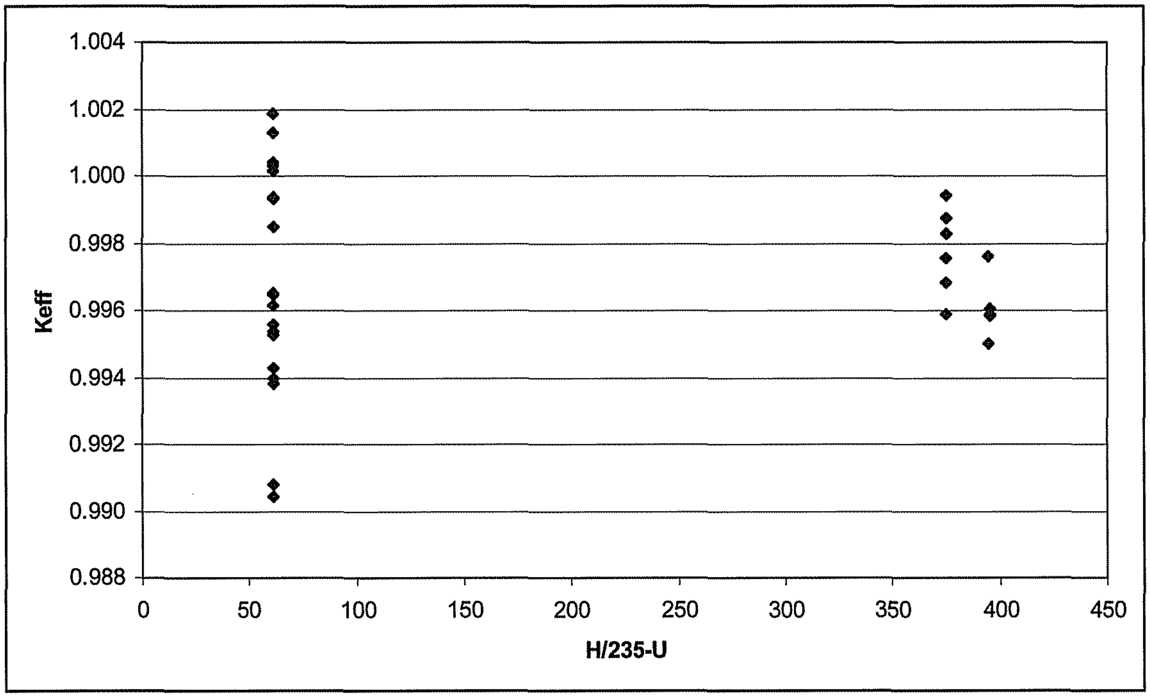
The basis for acceptability of the enrichment and temperature values being outside the range of applicability of the benchmarks is provided by the cross-section library being used. A key purpose of benchmarking is to validate the cross-section library for the proposed application. As discussed in Section 1.4 of WCAP-16541-P, Revision 2, the 44-group Evaluated Nuclear

Data File Version 5 (ENDF/B-V) was used for the PBNP analysis. NUREG/CR-6686 states that the ENDF/B-V library is particularly suited for light water reactor lattice applications; Section 7.1 notes that the library was developed for use in the analysis of fresh and spent light water reactor fuel.

The Shapiro-Wilks test for normality was applied to both the 44 and 238 group library results of the benchmark suite presented in Tables 1-1 and 1-2 of WCAP-16541-P, Revision 2. Both data sets pass the Shapiro-Wilks test and can thus be considered to have a normal distribution.

The calculated $k_{eff}$, using the 44-group cross-section library, of each of the thirty benchmark experiments was plotted versus the $H/^{235}U$ value of the experiment, and is shown in Figure 1.

**Figure 1 - Calculated $k_{eff}$ (44-group Cross-Section Library) vs. $H/^{235}U$**




The data is well distributed and no trend is apparent as a function of $H/^{235}U$. The results using the 238-group cross-section library showed similar distributions.

The measurement uncertainties in the benchmarks were not explicitly accounted for. This is justified by the fact that the experimental uncertainties are small and the methodology bias and uncertainty calculated with the benchmark data is comparably large.

Fission product critical data is not available. The effect of fission products is accounted for in the burnup uncertainty. In response to the acceptance review questions, cover letter Reference (2), the burnup uncertainty methodology was changed to the 5% decrement method. This method is sufficiently conservative to account for the lack of fission product critical data.

**Question 2: Tolerance/Uncertainty Calculations**

a) *Why did you not include the fuel pellet diameter uncertainty in "All-Cell" and "1-out-of-4, 5.0 w/o Fresh with no IFBA [integral fuel burnable absorber]" when you included it in "1-out-of-4, 4.0 w/o Fresh with IFBA" case?*
b) *How do you determine what manufacturing tolerances to include in the uncertainty study?*
c) *You appear to assume that the sum of biases and uncertainties for a given configuration remains constant for the different combinations of enrichment, burnup, decay period, and number of IFBAs (for the "1-out-of-4, 4.0 w/o Fresh with IFBA"). Please substantiate this assumption quantitatively.*

**NextEra Response**

The fuel pellet diameter for Standard fuel was modeled as a bounding value, meaning the value modeled was greater than the nominal diameter plus the manufacturing tolerance. Because a bounding value was modeled, no uncertainty needed to be evaluated in configurations that only considered Standard fuel. The All-Cell configuration only considered Standard fuel as discussed in the response to Question 3. However, in response to this question, the pellet diameter uncertainty effect was explicitly calculated at both the positive and negative extremes of the manufacturing tolerance from nominal and is reported below.

Manufacturing tolerances that have a statistically significant effect on the calculated eigenvalue and a physical basis are included in the uncertainty rackup. This includes parameters associated with the individual fuel pin characteristics that can propagate across a given fuel assembly and even across an entire fuel batch. Such parameters include fuel enrichment, fuel pellet diameter, and cladding thickness and diameter among others. Neglecting spacer grids has been shown to be conservative.

Manufacturing tolerances on the spent fuel pool rack are also considered. This includes parameters associated with rack pitch, wall thickness, and rack cell inner diameter. If the rack contains neutron absorbing material, and the criticality analysis takes credit for that material, tolerances associated with the neutron poison must also be evaluated. The position of the assembly within the cell is also considered.

The fuel rod pitch tolerance has historically been neglected for a combination of reasons. The primary reason is that the real variability of the pin pitch is small and random. These small pitch changes would yield small reactivity effects. The effect of variability in pin pitch would introduce some slight disarray in the fuel assembly. This disarray would act to lower reactivity slightly as an ordered array is more reactive than a disordered array. The amount of pin pitch variability is also necessarily small given the established tolerances on grid strap parameters and overall assembly dimensions.

The sum of biases and uncertainties for a given configuration does not remain constant over the entire range of allowable enrichments, burnups, decay periods, and number of IFBA rods. Care was taken when determining the conditions at which the biases and uncertainties would be calculated to ensure that the resulting sum would be conservative for the range of conditions over which it is applied. Tables 2 and 3 demonstrate that the total biases and uncertainties documented in WCAP-16541-P, Revision 2, are conservative.

Table 3-4 of WCAP-16541-P, Revision 2, reports the biases and uncertainties calculated for the All-Cell configuration. With the exception of the temperature bias, these tolerances were calculated by modeling fresh Standard fuel at the highest allowable fresh fuel enrichment. Standard fuel was modeled in this configuration because it is the design basis fuel assembly for low enrichment fresh fuel and for all burnt fuel, as discussed in the response to Question 3. The temperature bias was calculated with 5.0 wt% fuel depleted to 25,000 MWD/MTU burnup as discussed in cover letter Reference (2). To show that these calculations are conservative for the range of burnups and decay times over which they are applied, two additional bias and uncertainty rackups were created for the All-Cell configuration; one for 5.0 wt% fuel depleted to 35,000 MWd/MTU with zero decay time and one for 5.0 wt% fuel depleted to 35,000 MWD/MTU with 20-year decay time. SCALE 5.1 was used when creating the new rackups. SCALE 5.1 has gone through the same verification and validation as SCALE-PC. The methodology biases and uncertainties are properly accounted for between the two versions of SCALE allowing for a valid comparison of the total bias and uncertainty terms. The results are shown in Table 2 along with the WCAP-16541-P, Revision 2, results for convenience.

**Table 2 - Bias and Uncertainty Calculations, All-Cell Storage Configuration**

| | WCAP-16541-P Revision 2 | 35 GWd/MTU, 0 yrs decay | 35 GWd/MTU, 20 yrs decay |
|---|---|---|---|
| Case Description | $\Delta k_{eff}$ | $\Delta k_{eff}$ | $\Delta k_{eff}$ |
| Increased Fuel Enrichment | 0.00692 | 0.00688 | 0.00688 |
| Increased Pellet Diameter | -- | 0.00073 | 0.00013 |
| Decreased Pellet Diameter | -- | 0.00025 | -0.00001 |
| Decreased Clad OD & Thickness | 0.00169 | 0.00155 | 0.00119 |
| Decreased Cell Pitch | 0.00171 | 0.00150 | 0.00095 |
| Decreased Rack Thickness | 0.00309 | 0.00208 | 0.00222 |
| Off-Center Assembly position | 0.00708 | 0.00671 | 0.00621 |
| Burnup Uncertainty | 0.00781 | 0.00781 | 0.00781 |
| Methodology Uncertainty | 0.00639 | 0.00677 | 0.00677 |
| Statistical Sum of Uncertainties | 0.01467 | 0.01445 | 0.01414 |
| Methodology Bias | 0.00310 | 0.00318 | 0.00318 |
| Pool Temperature Bias | 0.01036 | 0.00932 | 0.00876 |
| Sum of Uncertainties and Biases | 0.02813 | 0.02695 | 0.02608 |

For the 35,000 MWD/MTU burnup 0 decay case, the result for the decreased pellet diameter is not statistically significant therefore, it is not included in the total. For the 35,000 MWD/MTU burnup 20-year decay case, neither increasing nor decreasing the pellet diameter gives statistically significant results so neither is included in the total. While there is both positive and negative variation in individual uncertainty terms, it can be seen that the total sum of uncertainties and biases as reported in WCAP-16541-P, Revision 2, bounds the range of burnups and decay times over which the uncertainties are applied.

The biases and uncertainties for the 1-out-of-4 4.0 wt% with IFBA configuration were calculated with 4.0 wt% fresh OFA fuel and no IFBA in one cell, and low enriched Standard fuel representing the burnt fuel in the other 3 cells. To show that the sum of uncertainties and biases is conservative over the range of number of IFBA for which it is applied, two additional bias and uncertainty rackups were created for the 1-out-of-4 4.0 wt% with IFBA configuration, one modeling 4.0 wt% fresh fuel and 16 1.5X IFBA rods, and one modeling 4.0 wt% fresh fuel and 32 1.5X IFBA rods. The results are shown in Table 3 along with the WCAP-16541-P, Revision 2, results for convenience.

**Table 3 - Bias and Uncertainty Calculations, 1-out-of-4, 4.0 wt% with IFBA Storage Configuration**

| | WCAP-16541-P Revision 2 | 4.0 wt% 16 IFBA | 4.0 wt% 32 IFBA |
|---|---|---|---|
| Case Description | $\Delta k_{eff}$ | $\Delta k_{eff}$ | $\Delta k_{eff}$ |
| Increased Fuel Enrichment | 0.00549 | 0.00530 | 0.00524 |
| Increased Pellet Diameter | 0.00125 | 0.00074 | 0.00074 |
| Decreased Pellet Diameter | -- | -0.00034 | -0.00017 |
| Decreased Clad OD & Thickness | 0.00198 | 0.00110 | 0.00129 |
| Decreased Cell Pitch | 0.00146 | 0.00148 | 0.00177 |
| Decreased Rack Thickness | 0.00201 | 0.00196 | 0.00230 |
| Off-Center Assembly position | 0.00420 | 0.00336 | 0.00358 |
| Burnup Uncertainty | 0.00589 | 0.00589 | 0.00589 |
| Methodology Uncertainty | 0.00644 | 0.00677 | 0.00677 |
| Statistical Sum of Uncertainties | 0.01165 | 0.01130 | 0.01147 |
| Methodology Bias | 0.00310 | 0.00318 | 0.00318 |
| Pool Temperature Bias | 0.00852 | 0.00769 | 0.00837 |
| Sum of Uncertainties and Biases | 0.02327 | 0.02217 | 0.02302 |

Again, there is both positive and negative variation in individual uncertainty terms, however, it can be seen that the total sum of uncertainties and biases as reported in WCAP-16541-P, Revision 2, bounds the range of number of IFBA over which the uncertainties are applied.

**<u>Question 3: Bounding Fuel Design</u>**

a) *In Section 1.5, you state that the Standard fuel design is bounding for spent fuel and OFA is bounding for fresh. Please quantitatively justify that this assumption is valid for all anticipated storage configurations and burnup/enrichment combinations at Point Beach.*
b) *In Section 3.2, you state, "Westinghouse standard fuel assembly design was modeled as the design basis fuel assembly to represent typical fresh and depleted fuel assemblies residing in all of the fuel assembly storage configurations." Does this contradict the statements in Section 1.5?*
c) *You also state "checkerboard storage configuration utilize the OFA fuel design." What do you mean by "checkerboard?" Are you referring to the 1 out of 4 configuration?*

**NextEra Response**

The optimized fuel assembly (OFA) is designed for improved in-reactor performance relative to the Standard assembly, and given the same assumed in-reactor conditions will be less reactive than a Standard assembly at realistic discharge burnups. Table 4 demonstrates the difference in calculated eigenvalue between the Standard and OFA fuel assemblies at 3.0 wt%, 4.0 wt%, and 5.0 wt% enrichment in the All-Cell storage configuration.

**Table 4 - $\Delta k$ [$k_{STD} - k_{OFA}$] at 3.0 wt%, 4.0 wt%, and 5.0 wt% as a Function of Burnup**

| Burnup (MWd/MTU) | 3.0 wt% $\Delta k$ [$k_{STD} - k_{OFA}$] | 4.0 wt% $\Delta k$ [$k_{STD} - k_{OFA}$] | 5.0 wt% $\Delta k$ [$k_{STD} - k_{OFA}$] |
|---|---|---|---|
| 0 | 0.00502 | 0.00034 | -0.00110 |
| 5,000 | 0.00560 | 0.00168 | -0.00159 |
| 15,000 | 0.01245 | 0.00485 | 0.00000 |
| 25,000 | 0.01759 | 0.01170 | 0.00462 |
| 35,000 | 0.02370 | 0.01395 | 0.00662 |
| 45,000 | 0.02928 | 0.01869 | 0.01090 |
| 55,000 | 0.03472 | 0.02352 | 0.01442 |

For the 5 wt% case, the Standard assembly becomes more limiting at 15,000 MWD/MTU. In all the configurations with 5.0 wt% initial enrichment, the burnups used to determine the burnup limit are at or above 15,000 MWD/MTU. These results also confirm the conservatism of using Standard fuel as the design basis fuel assembly for spent fuel and OFA as the design basis fuel assembly for fresh fuel when modeling 5.0 wt% initial enrichment, as in the 1-out-of-4, 5.0 wt% no IFBA configuration.

For the 3.0 wt% and 4.0 wt% cases, Table 4 shows Standard fuel as more reactive than OFA at all times in life including fresh fuel. These results confirm the conservatism of using Standard fuel as the design basis fuel assembly for spent fuel when modeling 4.0 wt%, or less, initial enrichment, but calls into question the appropriateness of modeling 4.0 wt% fresh OFA in the 1-out-of-4, 4.0 wt% Fresh with IFBA configuration. The four cases shown in Figure 2 were used to determine the design basis fresh fuel assembly for the 1-out-of-4, 4.0 wt% case.

| | |
|---|---|
| 1.6 wt % Fresh STD | 1.6 wt % Fresh STD |
| 4.0 wt % Fresh STD | 1.6 wt % Fresh STD |

| | |
|---|---|
| 5 wt %, STD, 55,000 MWd/MTU | 5 wt %, STD, 55,000 MWd/MTU |
| 4.0 wt % Fresh STD | 5 wt %, STD, 55,000 MWd/MTU |

| | |
|---|---|
| 1.6 wt % Fresh STD | 1.6 wt % Fresh STD |
| 4.0 wt % Fresh OFA | 1.6 wt % Fresh STD |

| | |
|---|---|
| 5 wt %, STD, 55,000 MWd/MTU | 5 wt %, STD, 55,000 MWd/MTU |
| 4.0 wt % Fresh OFA | 5 wt %, STD, 55,000 MWd/MTU |

**Figure 2 – Test Cases to Determine the Design Basis Fresh Fuel Assembly in the 1-out-of-4, 4.0 wt% Fresh Configuration.**

The highest allowable fresh enrichment in 3 of the 4 cells is 1.6 wt% according to Table 3-14 in WCAP-16541-P, Revision 2; and the highest enrichment/burnup combination used was 5.0 wt% at 55,000 MWD/MTU burnup so this covers the range of enrichment/burnup combinations. The single 4.0 wt% assembly was modeled as both an OFA and a Standard assembly for each enrichment/burnup combination; the results are shown in Table 5.

**Table 5 – Δk Results for the 1-out-of-4 4.0 wt% Configurations Shown in Figure 2**

| Description | 4.0 wt% Fresh Δk [$k_{STD} - k_{OFA}$] |
|---|---|
| 1.6 wt%, 0 MWd/MTU | -0.00299 |
| 5.0 wt%, 55,000 MWd/MTU | -0.00360 |

The results in Tables 4 and 5 confirm that the design basis fuel assemblies used in the WCAP-6541-P, Revision 2, analysis are appropriate and conservative. Furthermore, using Standard fuel as the design basis fuel assembly for spent fuel and OFA as the design basis fuel assembly for fresh fuel is consistent with the design basis fuel assemblies used in the previous, approved, PBNP criticality analysis. It is also consistent with other approved criticality analysis for 14x14 fuel lattices (References 1 and 2).

The statement referred to in part b of this question refers to the All-Cell storage configuration. As discussed in the response to Question 2, the bias and uncertainty calculations reported in WCAP-16541-P, Revision 2, with the exception of the temperature bias, were done with fresh fuel. For the All-Cell configuration this was modeled as fresh Standard fuel because the uncertainties needed to be applied to spent fuel. Also, the results in Table 4 show that Standard fuel becomes more limiting relative to OFA as enrichment decreases.

In the two "1-out-of-4" or "checkerboard" configurations the single fresh fuel assembly is modeled as OFA fuel and the 3 burnt assemblies are modeled as Standard fuel.

The term "checkerboard" as used in WCAP-16541-P, Revision 2, refers generically to the 1-out-of-4 storage configurations.

**Question 4: IFBA Depletion Effects**

a) *Letter dated September 19, 2008 (ADAMS Accession No. ML082630114), provided a response to the staff acceptance review. You state in response to Question 4, that the "results demonstrate that including the residual $^{10}B$ provides sufficient reactivity margin to account for the spectral hardening caused by the presence of IFBA during the depletion." This statement conflicts with NUREG/CR-6760 which states that "... the $\Delta k$ values become positive for fuel assembly designs containing IFBA rods but remain negative for gadolinia-bearing fuel assembly designs." NUREG/CR-6760 further states that "... analyses show that there is a negative residual effect for gadolinia-bearing fuel but no such effect for fuel designs with IFBA rods." Please resolve the difference in conclusions between your analysis and that of NUREG/CR-6760.*

b) *What enrichment was used for the calculations in the table titled, "Results of Calculations with IFBA Present During Depletion?" Please justify that the results are based on the limiting enrichment and burnup combinations.*

**NextEra Response**

The analysis presented in the September 19, 2008 letter (cover letter Reference (2)), was performed using site specific information for the fuel assembly design, core operating parameters, and IFBA designs used at PBNP. This is in contrast with the generic analysis of 17x17 fuel presented in NUREG/CR-6760. Not enough information is available in NUREG/CR-6760 for the Licensee or Vendor to know exact differences between the two analyses. However, an attempt will be made to call out the differences that are apparent. The results presented in the cover letter Reference (2), and additional results below are applicable to the PBNP spent fuel pool criticality safety analysis. The results presented here are not necessarily representative of other fuel lattices, assembly designs, IFBA designs or core operating parameters outside of PBNP.

Tables 6 and 7 contain modeling information from Section 3.3 of NUREG/CR-6760. A column is added to each table to show the values used in the PBNP analysis.

**Table 6 - Summary of Parameters Used for the Depletion Calculations (Table 1 of Section 3.3 of NUREG/CR-6760)**

| Parameter | NUREG/CR-6760 | Point Beach Analysis |
|---|---|---|
| Moderator Temperature (K) | 600 | 561.3 |
| Fuel Temperature (K) | 1000 | 1079.3 |
| Fuel Density (g/cm$^3$) | 10.44 ($UO_2$) | 10.69 ($UO_2$) |
| Clad Temperature (K) | 600 | 593.7 |
| Clad Density (g/cm$^3$) | 5.78 (Zr) | 6.56 (Zircaloy-4) |
| Power Density (MW/MTU) | 60 | 37 |
| Moderator Boron Concentration (ppm) | 650 | 800 |

Most of the calculations in NUREG/CR-6760 were performed assuming a uniform burnup profile. The PBNP analysis used the axial power distribution shown in Figure 3-9 of WCAP-16541-P, Revision 2; the fuel, moderator, and cladding temperatures varied as a function of power. The values shown in Table 6 correspond to the values at a relative power of 1.0.

The footnote to Table 1 of NUREG/CR-6760 states that cases were also calculated using a power density of 30 MW/MTU and that the $\Delta k$ results were not sensitive to variations in power density.

A significant difference is the moderator soluble boron concentration. By assuming a lower value, the results in NUREG/CR-6760 will exaggerate the effect of the IFBA. Per NUREG-6665, using a high soluble boron concentration during depletion is conservative due to spectral hardening. The IFBA also provides spectral hardening, but the effect is localized. Using 800 ppm for a cycle average value is high for the PBNP units. The NUREG/CR-6760 approach of using a lower soluble boron concentration will indicate a larger IFBA worth, but may underpredict the overall reactivity.

Standard fuel dimensions are presented in Table 7 because Standard fuel is the design basis assembly for burnt fuel.

**Table 7 - Fuel Assembly Specifications (Table 2 of Section 3.3 of NUREG/CR-6760)**

| Parameter | NUREG/CR-6760 | Point Beach Analysis |
|---|---|---|
| Rod Pitch (cm) | 1.260 | 1.412 |
| Assembly Pitch (cm) | 21.5 | 19.8 |
| Cladding Outside Diameter (cm) | 0.8898 | 1.0719 |
| Cladding Inside Diameter (cm) | 0.8001 | 0.9484 |
| Pellet Outside Diameter (cm) | 0.7840 | 0.9294 |
| Guide/Instrument Tube Outside Diameter (cm) | 1.204 | 1.369 / 1.072 |
| Guide/Instrument Tube Inside Diameter (cm) | 1.124 | 1.2827 / 0.94996 |
| Array Size | 17x17 | 14x14 |
| Number of Fuel Rods | 264 | 179 |
| Number of Guide/Instrument Tubes | 25 | 17 |

NUREG/CR-6760 examines 17x17 assemblies containing 80, 104, and 156 IFBA rods with boron loadings of 1.57 and 2.355 mg $^{10}B$/inch. The PBNP analysis uses 14x14 assemblies containing 120 IFBA rods with a 1.5X IFBA loading. The PBNP analysis IFBA pattern contains a significantly higher percentage of IFBA rods than those considered in NUREG/CR-6760 (67% for PBNP versus 59% in NUREG/CR-6760).

In the table titled "Results of Calculations with IFBA Present During Depletion," in the cover letter Reference (2), the calculations were performed with 5.0 wt% enriched fuel. The study reported in cover letter Reference (2) was performed with IFBA modeled over the entire axial length of the fuel.

Section 3.3.5.5 of NUREG/CR-6760 states that modeling a shorter IFBA stack can result in larger differences in calculated eigenvalues between cases depleted with and without IFBA present. Tables 10 and 11 of NUREG/CR-6760 show the Δk effects when a non-uniform axial burnup profile is modeled with IFBA modeled over the entire axial length of the fuel and with IFBA modeled over 120 inches, centered axially in the fuel rod. The more realistic IFBA model, 120 inches centered axially, is more limiting.

Therefore, the PBNP specific study was re-performed modeling a 120 inch IFBA region centered axially in the fuel assembly. Results of this new study are presented and discussed below.

Some similar trends are seen between the two analyses. Figure 41 in Section 3 of NUREG/CR-6760 shows Δk values versus burnup for the 156 IFBA pattern, with varying $^{235}U$ enrichments. Table 8 below shows similar trends for the 120 IFBA pattern, with varying $^{235}U$ enrichments in the All-Cell configuration of the PBNP analysis. The values shown in this table are plotted in Figures 3 and 4.

**Table 8 - Δk in the All-Cell Configuration**

| | 4.0 wt% | 5.0 wt% |
|---|---|---|
| MWd/MTU | $\Delta k\ [k_{(IFBA)} - k_{(no\_IFBA)}]$ | $\Delta k\ [k_{(IFBA)} - k_{(no\_IFBA)}]$ |
| 5,000 | -0.07924 | -0.08292 |
| 15,000 | -0.01332 | -0.02247 |
| 25,000 | 0.00128 | -0.00172 |
| 35,000 | 0.00267 | 0.00164 |
| 45,000 | 0.00237 | 0.00206 |

The data was fit using a fifth order polynomial and the difference in eigenvalues was found for the All-Cell configuration burnup limits for 4.0 wt% and 5.0 wt%, reported in Table 4-1 of WCAP-16541-P, Revision 2. The difference in eigenvalues at the burnup limits is shown in Table 9. For this site specific analysis, at the burnup limits specified in WCAP-16541-P, Revision 2, the residual IFBA is enough to overcome the effect of spectral hardening. The results confirm that neglecting the presence of IFBA is conservative in the All-Cell storage configuration.

**Table 9 - Δk in the All-Cell Configuration at the Burnup Limit**

| Limiting Burnup | 4.0 wt% | Limiting Burnup | 5.0 wt% |
|---|---|---|---|
| MWd/MTU | $\Delta k\ [k_{(IFBA)} - k_{(no\_IFBA)}]$ | MWd/MTU | $\Delta k\ [k_{(IFBA)} - k_{(no\_IFBA)}]$ |
| 18,475 | -0.00512 | 27,349 | -0.00006 |

**Figure 3 - Δk in the All-Cell Configuration**

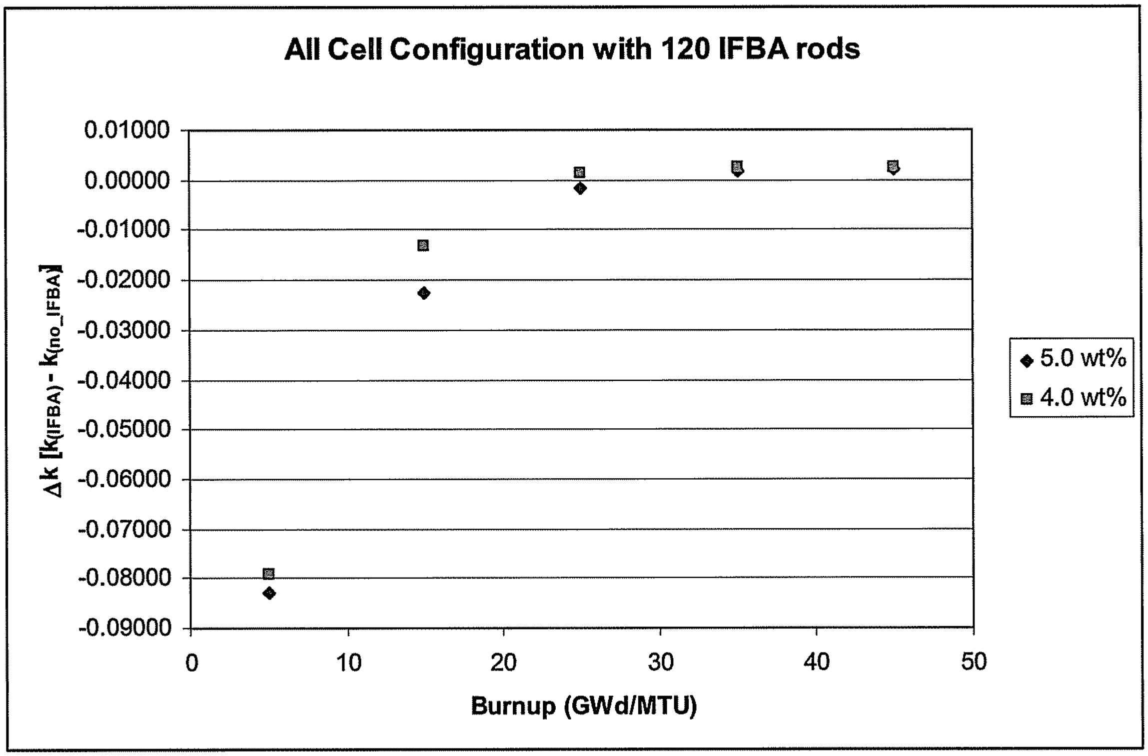
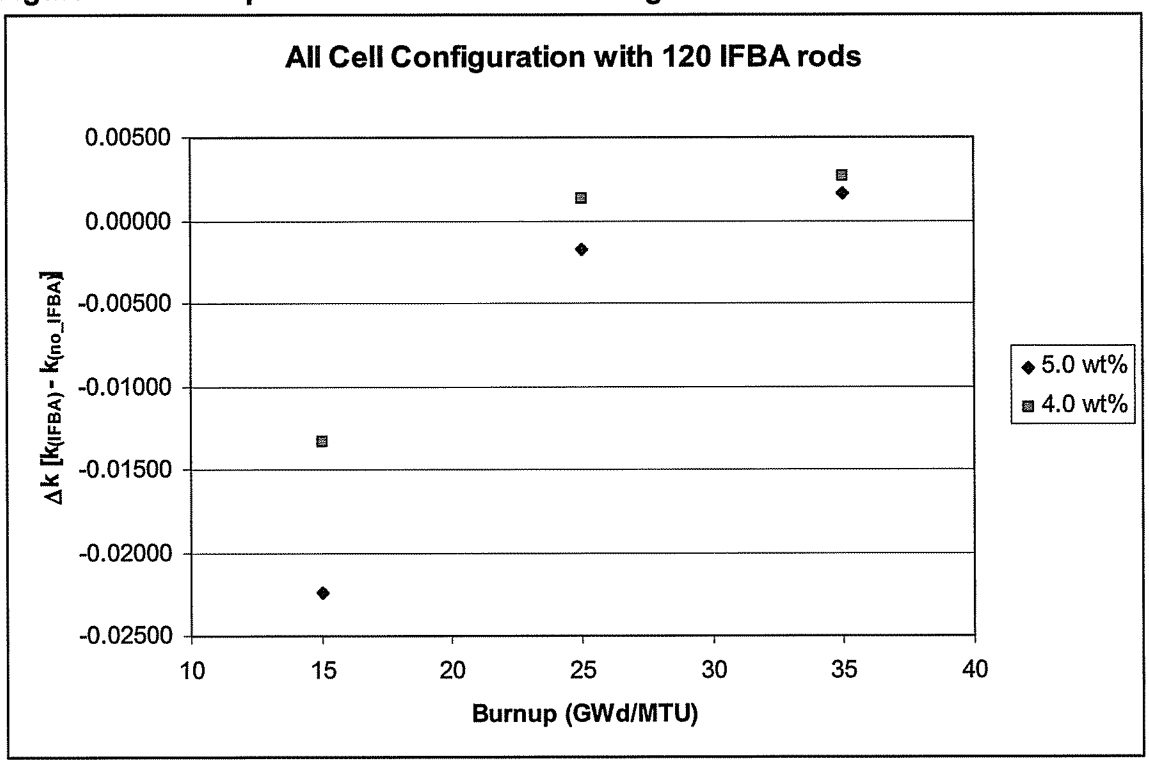



**Figure 4 - Close-up of the Area of Interest in Figure 3**



In the All-Cell configuration all four of the assemblies have been depleted with IFBA thereby maximizing the spectral hardening effect. However, since the All-Cell configuration also has the lowest burnup limits, the 1-out-of-4 configurations were examined.

The All-Cell case and NUREG/CR-6760 demonstrate that the effect at 4.0 wt% is stronger than at 5.0 wt% so the burnt assemblies modeled in the following 1-out-of-4 cases were modeled as Standard assemblies at 4.0 wt% initial enrichment with 120 IFBA rods that contain 120 inches of IFBA centered axially. Comparison of IFBA versus non-IFBA eigenvalues were performed for burnup on the Standard fuel assemblies from 5,000 to 55,000 MWD/MTU.

For the 1-out-of-4, 5.0 wt% Fresh no IFBA configuration, the three burnt assemblies were modeled as described in the preceding paragraph and the single fresh assembly was modeled as an OFA assembly at 5.0 wt% enrichment containing no IFBA rods.

Three additional cases were considered for the 1-out-of-4, 4.0 wt% Fresh with IFBA configuration. The three burnt assemblies were modeled as previously described. The single fresh assembly was modeled in the first case as an OFA assembly at 4.0 wt% enrichment containing 16 IFBA rods. The second case was modeled as an OFA assembly at 4.0 wt% enrichment containing 32 IFBA rods. The third case was modeled as an OFA assembly at 5.0 wt% enrichment containing 32 IFBA rods.

Because of the higher burnup limits in the 1-out-of-4 configurations, there is no longer enough residual IFBA to offset the effect of increased plutonium production. Once again the $\Delta k$ [$k_{(IFBA)} - k_{(no\_IFBA)}$] data was fit with a fifth order polynomial. Using the polynomial fits, the largest positive difference in eigenvalues was identified for each configuration, shown in Table 10.

**Table 10 – Largest Positive Difference in Eigenvalues for each Configuration**

| 1-out-of-4, 5.0 wt% Fresh, No IFBA $\Delta k$ [$k_{(IFBA)} - k_{(no\_IFBA)}$] | 1-out-of-4, 4.0 wt% Fresh, 16 IFBA $\Delta k$ [$k_{(IFBA)} - k_{(no\_IFBA)}$] | 1-out-of-4, 4.0 wt% Fresh, 32 IFBA $\Delta k$ [$k_{(IFBA)} - k_{(no\_IFBA)}$] | 1-out-of-4, 5.0 wt% Fresh, 32 IFBA $\Delta k$ [$k_{(IFBA)} - k_{(no\_IFBA)}$] |
|---|---|---|---|
| 0.00149 (41,000 MWD/MTU) | 0.00218 (45,000 MWD/MTU) | 0.00187(41,000 MWD/MTU) | 0.00053 (35,000 MWD/MTU) |

It can be seen from Table 10 that the largest positive difference in calculated eigenvalue for the 1-out-of-4, 5.0 wt% Fresh, with no IFBA configuration is 0.00149. The response to Question 1 in the cover letter Reference (2) used 0.00302 of the 0.00500 $\Delta k$ administrative margin in this configuration, which still leaves enough to cover the additional 0.00149 $\Delta k$.

It can be seen from Table 10 that the largest positive difference in calculated eigenvalue for the 1-out-of-4, 4.0 wt% Fresh, with IFBA configuration is 0.00218. The response to Question 1 in the cover letter Reference (2) used 0.00203 of the 0.00500 $\Delta k$ administrative margin in this configuration, which still leaves enough to cover the additional 0.00218 $\Delta k$. The remaining administrative margin for each configuration is shown in Table 11.

**Table 11 – Remaining Administrative Margin for each Configuration**

| | Remaining Administrative Margin |
|---|---|
| All-Cell | 0.00369 |
| 1-out-of-4, 5.0 wt% Fresh, No IFBA | 0.00049 |
| 1-out-of-4, 4.0 wt% Fresh, With IFBA | 0.00079 |

This study showed trends similar to those reported in NUREG/CR-6760, but was performed with PBNP specific conditions and concludes that analysis reported in WCAP-16541, Revision 2, remains conservative.

**Question 5: Soluble Boron Credit**

*Letter dated September 19, 2008 (ADAMS Accession No. ML082630114), provided a response to the staff acceptance review. Response to Question 2 discussed the effect of "parallel" accounting method on the boron concentration required for accident conditions. Please justify the effect of "parallel" accounting method on the boron concentration required for nominal conditions.*

**NextEra Response**

WCAP-16541, Revision 2, reports 402 ppm as the soluble boron concentration, assuming 19.4 atom percent $^{10}B$ abundance, required to maintain $k_{eff}$ less than or equal to 0.95 including all biases and uncertainties for nominal conditions.

Isotopics are not available at the burnup limits specified in WCAP-16541, Revision 2, so the available isotopics for burnups closest to the limits assuming 5.0 wt% initial enrichment were used. The limits associated with 5.0 wt% initial enrichment were used because this maximizes the required burnup. As burnup increases, soluble boron worth decreases due to the harder neutron spectrum. The isotopics at the burnup closest to, but less than the limit, are used because the effect of additional burnup on reactivity is expected to be larger than the reduction in boron worth due to the additional burnup. Therefore, if there is margin to the Upper Subcritical Limit, or target eigenvalue, for fuel that does not meet the burnup limit specified in WCAP-16541, Revision 2, it proves the required boron is conservatively high. Results using the isotopics at the burnup closest to, but over the burnup limit, are included to show that the reactivity effect due to additional burnup is larger than the reduction in boron worth, i.e., there is more margin to the target eigenvalue even though the soluble boron is worth slightly less.

The isotopics used in this analysis are summarized for the three configurations in Table 12.

**Table 12 – Burnup Limits of Interest and Isotopics Used to Calculate Margin**

| Configuration | Burnup Limit Specified in WCAP-16541, Revision 2 | Isotopics Used |
|---|---|---|
| All-Cell | 27,349 MWd/MTU | 25,000 / 35,000 MWd/MTU |
| 1-out-of-4, 5.0 wt% Fresh, no IFBA | 51,169 MWd/MTU | 45,000 / 55,000 MWd/MTU |
| 1-out-of-4, 4.0 wt% Fresh, with IFBA | 41,361 MWd/MTU | 35,000 / 45,000 MWd/MTU |

The three configurations were modeled as 2x2 infinite arrays as described in Section 3 of WCAP-16541, Revision 2. This is conservative by not accounting for leakage that would be present in the actual pool. WCAP-16541, Revision 2, reports 402 ppm as the required soluble boron concentration, assuming 19.4 atom percent $^{10}B$ abundance, so the moderator was modeled containing 400 ppm of soluble boron with 19.4 atom percent $^{10}B$ abundance which is close to, but conservatively less than, the required amount.

The calculated eigenvalues were compared to the target eigenvalues documented in cover letter Reference (2) in response to Question 5. These target eigenvalues considered borated (648 ppm) biases and uncertainties and the 5% decrement method for burnup uncertainty documented in response to Question 1 in the same letter.

Margin is demonstrated by subtracting the calculated eigenvalue from the target eigenvalue; there is positive margin when the calculated value is less than the target. Margins to the target eigenvalue for each configuration at each burnup considered are shown in Table 13.

**Table 13 - Margin to Target Eigenvalue**

| Burnup (MWD/MTU) | Margin to Target Eigenvalue |
|---|---|
| All-Cell | |
| 25,000 | 0.01321 |
| 35,000 | 0.06164 |
| | |
| 1-out-of-4, 5.0 wt% Fresh, no IFBA | |
| 45,000 | 0.01551 |
| 55,000 | 0.03213 |
| | |
| 1-out-of-4, 4.0 wt% Fresh, with IFBA | |
| 35,000 | 0.01450 |
| 45,000 | 0.04023 |

Margin is shown to the target eigenvalue for every configuration, thereby justifying the "parallel" accounting method at nominal conditions.

The target eigenvalues are not expected to be significantly different at 400 ppm than at 648 ppm. The smallest amount of margin to the target eigenvalue is more than 20 times larger than the largest difference between the borated and unborated target eigenvalues shown in cover letter Reference (2).

As discussed in the response to Question 2 documented in cover letter Reference (2), this margin is the results of the conservatisms included in the soluble boron concentration equation shown below and in the method used to determine the soluble boron worth.

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

Where: $SBC_{Total}$ is the total soluble boron concentration 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)

The reactivity uncertainty of the fuel assembly is accounted for in the burnup uncertainty included in the determination of the burnup limits. The soluble boron worth is also conservatively determined in the full pool model loaded with depleted fuel which is less sensitive to the addition of soluble boron. These two conservatisms, which are implicit in the methodology used in the analysis presented in WCAP-16541-P, Revision 2, provide sufficient conservatism in the determination of the required boron concentration.

**References:**

1. Letter to Dr Robert C. Mecredy (RG&E) from Guy S. Vissing (NRC), dated December 7, 2000 "R.E. Ginna Nuclear Power Plant – Amendment RE: Revision to the Storage Configurations Requirements within the Existing Storage Racks and Taking Credit for a Limited Amount of Soluble Boron" (ML003761578)

2. Letter to Thomas J. Palmisano (NMC) from Mahesh L. Chawla (NRC), dated February 5, 2006 "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: 'Spent Fuel Pool Storage" (ML060250208)

# ENCLOSURE 2

## NEXTERA ENERGY POINT BEACH, LLC
## POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

## PBNP BORAFLEX
## SURVEILLANCE PROGRAM

As discussed in the cover letter, a teleconference was held between NRC and NextEra personnel to discuss PBNP response to Question 1 of Reference (3). It was agreed that NextEra would formally respond to the staff's query on the boraflex surveillance program. That response follows.

**<u>Question 1</u>**

*Reference 3, question 1, where the licensee states "The Boraflex Surveillance Program described in the letter dated 10/23/96 was discontinued on 12/21/06. Commitments to implement a Boraflex Monitoring Program for the period of extended operation remain in place." Please clarify whether a Boraflex Surveillance Program is currently in effect (especially for the period of 12/21/06 to the beginning of the period of extended operation).*

**<u>NextEra Response</u>**

The PBNP Boraflex Surveillance Program includes blackness testing conducted every five (5) years. The blackness testing in 2006 was deferred. The next required test is areal density testing which must be performed prior to entering the period of extended operation in 2010. Currently PBNP continues to:

1. Maintain a database that is updated each time fuel movements are performed
2. Monitor industry OE in accordance with PBNP Operating Experience program
3. Notify the NRC if the program is to be modified
4. Checkerboard fresh and spent fuel assemblies with burnup less than 38,400 MWD/MTU, and if significantly degraded Boraflex is found.

The current analysis bounds current and future gap formation. The trends of silica are monitored have not shown an upward acceleration, and remain steady around 19 ppm.

The following license renewal commitments were made and are documented in NUREG 1839:

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 desntiy 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% 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% 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 Reanalysis, Repair and/or Replacement.