August 7, 2009

NRC 2009-0073
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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) (ML090900617)
(5) NextEra Energy Point Beach Letter to NRC, Response to Request for Additional Information, License Amendment Request 247, Spent Fuel Pool Storage Criticality Control, dated May 22, 2009 (ML091420436)
(6) NRC letter to NextEra 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 - Round 2, dated July 9, 2009 (TAC Nos. MD9321 and MD9322) (ML091770550)

NextEra Energy Point Beach, LLC (NextEra) (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 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. NextEra provided a supplemental response (Reference 2) containing additional quantitative information to support the fidelity of key methodology aspects described in Reference (1).

The enclosure of this letter provides the NextEra response to the request for additional information in accordance with Reference (6). Question 4a) from Reference (2) will be provided by August 28, 2009, in accordance with discussions held between representatives of NextEra and NRC on August 5 and 6, 2009.

Summary of Regulatory Commitments

The following new Regulatory Commitment is proposed for submittal of the response to Question 4a) from Reference (2).

- By August 28, 2009, NextEra will provide a quantitative justification to demonstrate that power suppression assemblies do not result in a more reactive assembly.

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 August 7, 2009.

Very truly yours,

NextEra Energy Point Beach, LLC

FOR

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

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 (RAI) dated July 9, 2009, to support continued review of Point Beach Nuclear Plant (PBNP) License Amendment Request 247.

Regarding licensee letter dated, May 22, 2009:

Regarding Question 1 Code Validation:

a) *Table 1 of Request for Additional Information (RAI) 1 response does not include any spectral parameters (e.g., energy of the average lethargy causing fission, hydrogen to uranium atomic ratio (H/U)) to show how the validation is applicable to the Point Beach analysis. Figure 1 of RAI 1 response provides the H/U range for the validation but does not show how it compares to the Point Beach analysis. What are the H/U values for the system analyzed for Point Beach?*

b) *Figure 1 shows a large range of H/U that is not supported by benchmarks. How is this justified?*

c) *How did you conclude that the Figure 1 "data is well distributed and no trend is apparent as a function of $H/^{235}U$?" Did you use any quantitative methods such as the regression analysis?*

NextEra Response

Table 1 and Figure 1 show the calculated SCALE-PC neutron multiplication factors (k_{eff}) versus $H/^{235}U$ for the benchmark experiments, the nominal model for each of the PBNP configurations, and the All-Cell depletions surrounding the calculated burnup limit for 3.0, 4.0, and 5.0 wt%. The k_{eff} values shown for the PBNP analysis include the Total Biases and Uncertainties term for each configuration. The benchmark $H/^{235}U$ ratios have been recalculated to be consistent with the calculation of $H/^{235}U$ for the PBNP models. SCALE Version 5.1 k_{eff} values are also shown in Table 1 because this version of the code is used to support several of the RAI responses. SCALE Version 5.1 use for k_{eff} calculations is always indicated, except for Question 4 of the May 22, 2009 letter.

Table 1 – $H/^{235}U$, EALF, and K_{eff} for Benchmarks and PBNP Models

	$H/^{235}U$	EALF	SCALE-PC k_{eff}	SCALE-5.1 k_{eff}
Benchmark X	217.3	0.15	0.99648	0.99688
Benchmark XI	217.3	0.20	1.00016	0.99914
Benchmark XI	217.3	0.20	0.99937	0.99999
Benchmark XI	217.3	0.20	1.00030	1.00005
Benchmark XI	217.3	0.20	0.99931	0.99944
Benchmark XI	217.3	0.20	1.00129	0.99986
Benchmark XI	217.3	0.20	1.00043	1.00020
Benchmark XI	217.3	0.20	1.00186	1.00114
Benchmark XII	217.3	0.17	0.99847	0.99736
Benchmark XIII	217.3	0.19	0.99653	0.99742
Benchmark XIIIA	217.3	0.20	0.99537	0.99513
Benchmark XIV	217.3	0.20	0.99559	0.99348
Benchmark XIX	217.3	0.17	0.99045	0.98945
Benchmark XV	217.3	0.20	0.99078	0.99044
Benchmark XVI	217.3	0.17	0.99397	0.99346
Benchmark XVII	217.3	0.20	0.99527	0.99354
Benchmark XVIII	217.3	0.20	0.99614	0.99574
Benchmark XX	217.3	0.17	0.99428	0.99422
Benchmark XXI	217.3	0.15	0.99384	0.99333
Benchmark 43	105.4	0.29	0.99757	0.99799
Benchmark 44	105.4	0.29	0.99945	1.00039
Benchmark 45	105.4	0.29	0.99875	0.99971
Benchmark 46	105.4	0.29	0.99826	0.99829
Benchmark 61	105.4	0.24	0.99681	0.99686
Benchmark 62	105.4	0.24	0.99588	0.99678
Benchmark 64	105.4	0.24	0.99591	0.99741
Benchmark 71	105.4	0.25	0.99760	0.99894
Benchmark 79	105.4	0.34	0.99500	0.99573
Benchmark 87	105.4	0.32	0.99582	0.99435
Benchmark 93	105.4	0.32	0.99604	0.99788
Point Beach Nominal Model 1oo4, 4.0 wt% Fresh with IFBA	217.68	0.14	0.99566	N/A
Point Beach Nominal Model 1oo4, 5.0 wt% Fresh no IFBA	215.84	0.16	0.99534	N/A
Point Beach Nominal Model All-Cell	208.31	0.14	0.99550	N/A
Point Beach All-Cell Depletion 3.0 wt%, 5k	182.64	0.18	1.02523	N/A
Point Beach All-Cell Depletion 3.0 wt%, 15k	269.27	0.20	0.93965	N/A
Point Beach All-Cell Depletion 4.0 wt%, 15k	177.16	0.22	1.01124	N/A
Point Beach All-Cell Depletion 4.0 wt%, 25k	244.37	0.23	0.94846	N/A
Point Beach All-Cell Depletion 5.0 wt%, 25k	169.54	0.25	1.00431	N/A
Point Beach All-Cell Depletion 5.0 wt%, 35k	225.15	0.25	0.95348	N/A

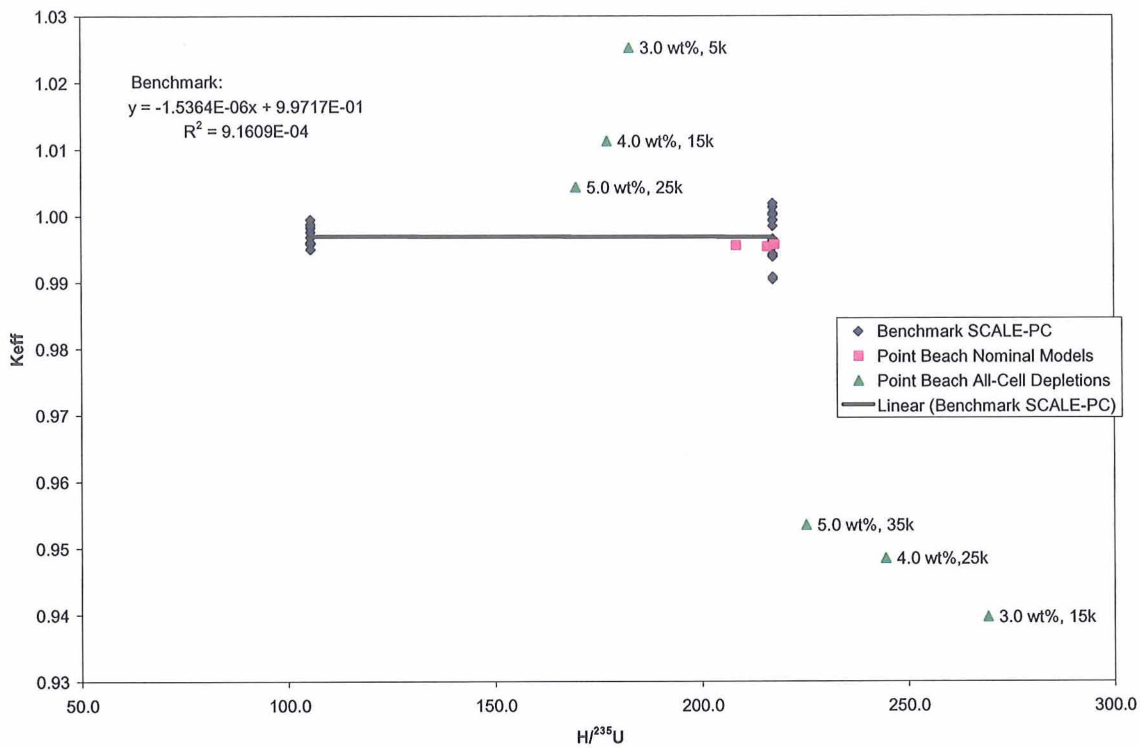

Figure 1 – Calculated k_{eff} versus $H/^{235}U$

$H/^{235}U$ is an indicator of what the neutron energy spectrum in a system will be; a direct calculation of that spectrum is Energy of the Average Lethargy causing Fission (EALF). Table 1 and Figure 2 show the calculated k_{eff} versus EALF for the same dataset as that shown in Figure 1.

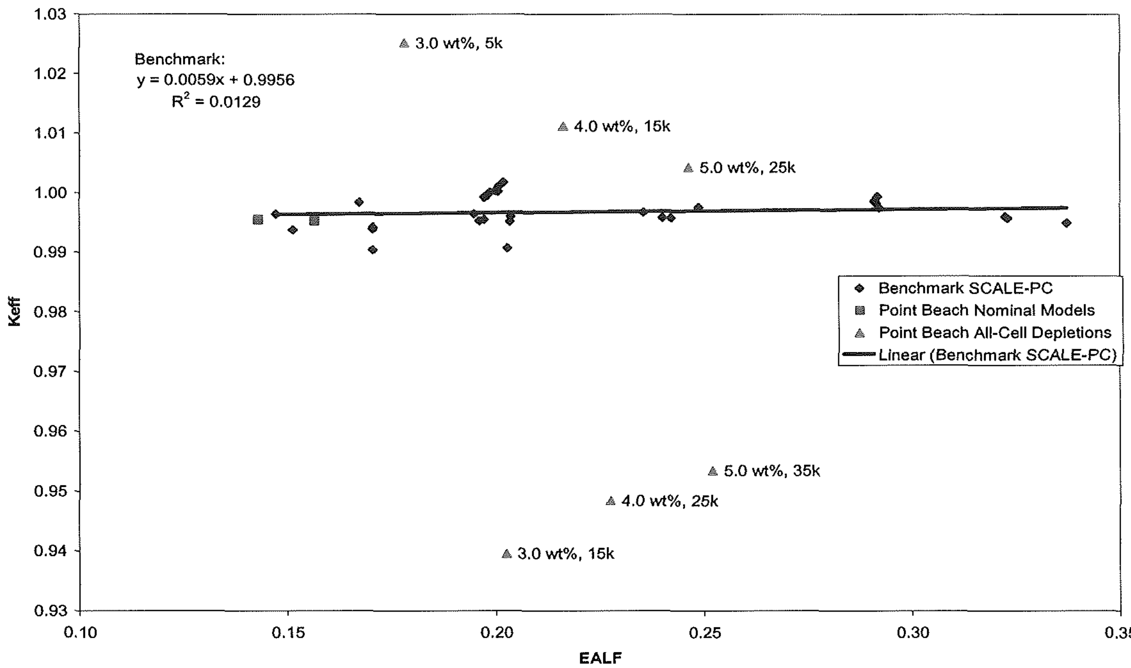

Figure 2 – Calculated k_{eff} versus EALF

Figure 2 shows that the benchmark covers the relevant energy spectrum better than might be indicated by Figure 1. The justification for the use of this benchmark suite is that it adequately covers the range of important parameters defined in NUREG/CR-6698. Figure 2 shows that the neutron energy spectrum of the PBNP analysis is similar to the neutron energy spectrum of the benchmark suite.

It is important to note that this benchmark suite has been used by Westinghouse Electric Company, LLC, to benchmark codes used in criticality safety analyses for many years and there are several NRC approved analyses, including Beaver Valley, Prairie Island and Vogtle that use this same benchmark suite.

This suite has been, and remains, applicable because: 1) the experiments in the benchmark suite were specifically designed to be criticality benchmarks for storage of light water reactor fuel; 2) the benchmark suite and PBNP and other analyses have similar fuel, moderator, poison, and structural materials, all at similar temperatures. Therefore it can be concluded that the energy spectrum is necessarily similar; and 3) the other important parameters defined in NUREG/CR-6698 are similar.

The conclusion that the data in Figure 1 of cover letter Reference (5), was well distributed and no trend was apparent was reached by examining the figure; no quantitative methods were used. Figures 1 and 2 of this document include linear fits of the benchmark data. The equation and R^2 value of each linear fit is included on the figure. The R^2 value is the coefficient of determination and measures the proportion of variation that is explained by the fit. An R^2 value of 1.0 would indicate a perfect fit. Both R^2 values are much less than 0.5 indicating that there is no trend in the data.

The following discussion does not directly relate to this RAI but is provided for clarification and information:

The analysis reported in WCAP-16541-P, Revision 2, uses the PHOENIX-P lattice code to generate depleted isotopic number densities and SCALE-PC for three-dimensional Monte Carlo calculations to determine k_{eff} in the spent fuel pool (SFP) environment. SCALE-PC uses the BONAMI and NITAWL modules for cross-section processing and the KENO V.a module for transport calculations. Reference 1 of this submittal explicitly allows the use of NITAWL-KENO V.a and PHOENIX-P for spent fuel pool criticality safety calculations.

The reactivity determinations made using SCALE Version 5.1 use the BONAMI and NITAWL modules for cross section processing and the KENO V.a module for three-dimensional Monte Carlo calculations. SCALE-PC uses the same modules for the same purposes as SCALE Version 5.1. The 44-group neutron cross section library based on ENDF/B-V data is used in both codes. The same codes and methods accepted in Reference 1 of this submittal are used in both versions of SCALE. Therefore, SCALE Version 5.1 is an appropriate code for use. SCALE Version 5.1 has gone through the same verification and validation process, using the same benchmark experiments as SCALE-PC.

The PARAGON lattice code was approved for use in reactor physics calculations by the NRC in Reference 2 of this enclosure. PARAGON is intended for use as a standalone code or as a replacement for PHOENIX-P. Reference 2 states, "The PARAGON code can be used as a replacement for the PHOENIX-P lattice code." Therefore, PARAGON is an acceptable code for the generation of depleted isotopic number densities.

Regarding licensee letter dated, May 22, 2009:

Regarding Question 2 Tolerance and Uncertainty Calculations:

a) What enrichment value was the enrichment reactivity uncertainty based on?

NextEra Response

The "1-out-of-4 4.0 wt% with IFBA" configuration considers an increase from 4.0 to 4.05 wt% and an increase from 1.578 to 1.628 wt% ^{235}U independently. The "All-Cell" configuration considers an increase from 2.147 to 2.197 wt% ^{235}U for fresh fuel.

The uncertainty of fresh fuel was used in the depleted cases as it is conservative compared to the effect of two depletions performed 0.05 wt% apart. Also, the effect of the enrichment uncertainty at lower enrichments is higher than it would be at higher enrichments. Regardless of these conservatisms, the enrichment uncertainty is treated consistently as unchanging throughout the response.

The "1-out-of-4 4.0 wt% with IFBA" configuration considers an increase from 4.0 to 4.05 wt% and an increase from 1.578 to 1.628 wt% ^{235}U independently. The two uncertainties are then root sum squared to determine the overall enrichment uncertainty. Root sum squaring the individual uncertainties increases the final uncertainty to a value greater than either of the individual uncertainties. These calculations are performed for each IFBA pattern. This is a more conservative approach than was used in WCAP-16541-P, Revision 2. The reduction in uncertainty compared to WCAP-16541-P, Revision 2 is caused by using more neutron histories in the calculations supporting the RAI responses. The increased histories reduced the Monte Carlo uncertainties by approximately 0.00020 Δk_{eff} in each calculation. This translates to a reduction of approximately 0.00040 Δk_{eff} in each portion of the enrichment uncertainty since the computational uncertainties from both results are added to determine the reported enrichment uncertainty. The reduction is magnified again because there are two separate enrichment evaluations combined. The two separate enrichment evaluations are combined statistically, so the impact is likely on the order of 0.00060 Δk_{eff}. Thus, a more conservative approach evaluated more rigorously resulted in a slightly lower overall enrichment uncertainty.

In WCAP-16541-P, Revision 2, the "All-Cell" configuration considered an increase from the maximum allowable fresh fuel enrichment of 2.13 to 2.18 wt% ^{235}U. In cover letter Reference (5), the "All-Cell" configuration considered an increase from 2.147 to 2.197 wt% ^{235}U for fresh fuel. In response to this question, the enrichment uncertainty for the "All-Cell" configuration was recalculated considering an increase from 2.13 to 2.18 wt% ^{235}U. This resulted in a statistically insignificant decrease in the calculated enrichment uncertainty.

The PBNP analysis reported in WCAP-16541-P, Revision 2 and in the subsequent supporting documents did not use fresh fuel to represent depleted fuel for determining burnup credit, spent fuel pool temperature effects or soluble boron worth.

NUREG/CR-6683 states that the practice of equating the reactivity of spent fuel to fresh fuel is acceptable, provided the conditions for which the reactivity equivalent fresh fuel enrichment was determined remain unchanged. This is the case for the bias and uncertainty calculations for which fresh fuel was used in place of spent fuel in the PBNP analysis. When calculations were performed where the conditions for the configuration changed, i.e., the determination of soluble boron credit, burnup credit, or temperature bias, isotopics representing spent fuel were modeled.

In cover letter Reference (5), in response to Question 2, the sum of biases and uncertainties were calculated with isotopics representing spent fuel modeled, except in the case of the enrichment uncertainty, which is described above. The resulting calculation showed that the sum of biases and uncertainties calculated with depleted fuel was lower with than that originally reported in WCAP-16541-P, Revision 2 which was calculated with fresh fuel.

Regarding licensee letter dated, September 19, 2008:

Regarding Question 1:

a) *For each storage configuration, what burnup value was the burnup reactivity uncertainty based on?*

NextEra Response

Configuration	Burnup
All-Cell	27,349
1oo4 5.0 wt% Fresh	51,169
1oo4 4.0 wt% Fresh with IFBA	41,361

The 5.0 wt% burnup limit is used for all configurations as it is the largest credited burnup and will therefore yield the largest value of the burnup reactivity uncertainty.

New Question:

Question 1 (Interface Analysis)

a) Please justify how the k-eff comparison is valid when the interface model assumes radial leakage and the individual storage models consider no radial leakage (i.e. repeating 2x2 array).

b) Please show that placing a more reactive storage configuration (e.g., 1-out-of-4 Fresh 5 percent no integral fuel burnable absorber) next to a less reactive storage configuration (e.g. All-cell) will not lead to unacceptable increase in reactivity of the less reactive storage configuration (e.g. All-cell).

NextEra Response

The evaluations are performed in a model of the SFP. As such, any credited radial leakage exists in the actual spent fuel pool. The effect of the radial leakage on the interface condition is insignificant more than one row in from the edges of the rack modules near the SFP walls. The use of infinite array models to determine the burnup limits provides a conservative reactivity determination by neglecting all radial leakage. The use of these models to determine biases and uncertainties is also conservative by maximizing the impact of each condition being considered. Fuel which meets these conservative limits in the infinite array models will inherently meet the reactivity requirements in the real, finite pool. The interface models therefore credit the actual state of the fuel which meets the regulatory requirements in 10 CFR 50.68 of a k_{eff} value less than 1.0 (unborated) or 0.95 (borated) in the SFP.

All three configurations are designed to have the same absolute reactivity of 0.945 accounting for biases and uncertainties in borated conditions. The interface requirements defined in WCAP-16541-P, Revision 2, require that high reactivity (i.e. fresh) fuel in either of the "1-out-of-4" configurations cannot be in the interface row. This ensures that all 2x2 arrays containing an interface store only depleted fuel. Each of the assemblies in these arrays is therefore required to meet the "All-Cell" storage requirements as no fresh assemblies are in these locations. Any assembly which meets the requirements for the configurations containing fresh fuel will far exceed the minimum burnup requirements for the "All-Cell" configuration. This ensures that the configuration interfaces are actually regions of low reactivity.

In order to provide quantitative evidence that a finite interface model is justifiable and that the allowable interface configurations do not result in an increase in reactivity, the allowable interface configurations were modeled in an infinite array using SCALE Version 5.1.

Interfaces between each of the configurations were considered. The interfaces between the "All-Cell" and "1-out-of-4" configurations were modeled as infinitely repeating 5x2 arrays. A 5x2 array was necessary instead of a 4x2 array because only allowable interface configurations were considered and the fresh assembly in an interface row is not an allowable configuration. A 6x2 array was needed to model the interface of the two "1-out-of-4" configurations for the same reason. An example for a "1-out-of-4"/"All-Cell" interface is shown on the following page.

If a “1-out-of-4” and “All-Cell” interface were modeled as a 4x2 (shown in blue) repeating array, an illegal configuration would result:

All-Cell	All-Cell	1oo4 Fresh	1oo4 Depleted	All-Cell	All-Cell	1oo4 Fresh	1oo4 Depleted
All-Cell	All-Cell	1oo4 Depleted	1oo4 Depleted	All-Cell	All-Cell	1oo4 Depleted	1oo4 Depleted
All-Cell	All-Cell	1oo4 Fresh	1oo4 Depleted	All-Cell	All-Cell	1oo4 Fresh	1oo4 Depleted
All-Cell	All-Cell	1oo4 Depleted	1oo4 Depleted	All-Cell	All-Cell	1oo4 Depleted	1oo4 Depleted

Illegal Interfaces

Therefore, a repeating 5x2 (shown in blue) array is needed so only allowable interfaces result:

All-Cell	All-Cell	1oo4 Depleted	1oo4 Fresh	1oo4 Depleted	All-Cell	All-Cell
All-Cell	All-Cell	1oo4 Depleted	1oo4 Depleted	1oo4 Depleted	All-Cell	All-Cell
All-Cell	All-Cell	1oo4 Depleted	1oo4 Fresh	1oo4 Depleted	All-Cell	All-Cell
All-Cell	All-Cell	1oo4 Depleted	1oo4 Depleted	1oo4 Depleted	All-Cell	All-Cell

Isotopics are not available at the burnup limits specified in WCAP-16541, Revision 2, so the available isotopics for burnups closest to, but less than the limits assuming 5.0 wt% initial enrichment were used. The burnups less than the limit were chosen because they create models that are more reactive than what is allowed to be stored in the spent fuel pool and provide a bounding analysis. No soluble boron was included in these models.

The isotopics used in this analysis are summarized for the three configurations in Table 2.

Table 2 – Burnup Limits of Interest and Isotopics Used

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

The results of the interface calculations are shown in Table 3.

Table 3 – Interface Results from Infinite Models using SCALE 5.1

Interface Configuration	k_{eff}	σ
1oo4 5.0 wt% 45k / All-Cell 25k	0.97411	0.00020
1oo4 4.0 wt% 35k / All-Cell 25k	0.97815	0.00017
1oo4 5.0 wt% 45k / 1oo4 4.0 wt% 35k	0.97493	0.00022

In order to allow for a valid comparison, the 2x2 infinite array base cases were rerun in SCALE 5.1, the results are shown in Table 4.

Table 4 – 2x2 Base Case Results from Infinite Models using SCALE 5.1

Configuration	k_{eff}	σ
All-Cell	0.98035	0.00015
1-out-of-4, 5.0 wt% Fresh, no IFBA	0.98758	0.00020
1-out-of-4, 4.0 wt% Fresh, with IFBA	0.99113	0.00019

Examining Tables 3 and 4 provides quantitative evidence that the allowable interfaces are regions of low reactivity.

The depleted fuel in each configuration was modeled with isotopics representing spent fuel as described in Table 2. In the WCAP-16541, Revision 2 interface analysis, the depleted fuel in each configuration was modeled with the highest allowable fresh fuel enrichment.

As discussed previously, all configurations considered have the same absolute reactivity when accounting for all applicable biases and uncertainties. Therefore, there is no mismatch in reactivities of adjacent configurations in the analysis.

The biases and uncertainties for each individual configuration are created using a conservative mix of fresh and spent fuel isotopics. Spent fuel isotopics are used for the temperature bias and the largest credited burnup is used for determining the burnup reactivity uncertainty. Fresh fuel is modeled in the determination of mechanical uncertainties related to fuel and rack manufacture and assembly positioning.

The total sum of biases and uncertainties is calculated and the nominal conditions for each configuration are defined such that the reactivity of the configuration is set in the fresh fuel condition. The configurations take into account the spent fuel conditions as evidenced by the depletion uncertainty and the temperature bias determination, but in the interface analysis the fresh fuel isotopics are being used in the conditions for which they were determined.

References

1. Laurence Kopp (USNRC), Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, August 19, 1998 (ML003728001)

2. Letter from H. N. Berkow (NRC) to J.A. Gresham (Westinghouse), Final Safety Evaluation for Westinghouse Topical Report WCAP-16045-P, Revision 0, 'Qualification of the Two-Dimensional Transport Code PARAGON' (TAC No. MB8040), dated March 18, 2004 (ML040780402)