FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077

Roy K. Brosi
Director, Performance Improvement

724-682-5210
Fax: 724-682-1840

March 18, 2010
L-10-082

10 CFR 50.90

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

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Response to NRC Staff Request for Additional Information Regarding Criticality Analyses Supporting a Spent Fuel Pool Re-rack for Unit 2 (TAC No. ME1079)

A license amendment request that would revise the Beaver Valley Power Station, Unit No. 2 Technical Specifications to support installation of high density fuel storage racks in the Unit 2 spent fuel pool was submitted on April 9, 2009 (Accession No. ML091210251), as supplemented by letters dated June 15, 2009 (Accession No. ML091680614) and January 18, 2010 (Accession No. ML100191805). The Nuclear Regulatory Commission (NRC) staff provided a request for additional information (RAI) regarding the criticality analysis that supports the requested amendment by a letter dated February 16, 2010. An audit was conducted by the NRC at the offices of Holtec International during the week of March 1, 2010. Based on the results of the audit, responses to a majority of the RAI items listed in the February letter are provided in Attachment 1 to this letter. Future correspondence will provide responses to the remaining RAI items and appropriate supplements to the license amendment request, as noted in the attachment.

A regulatory commitment is identified in Attachment 2 to this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 18, 2010.

Sincerely,

Roy K. Brosi

A001
NRR

Attachments:
1. Response to RAI on Unit 2 Spent Fuel Pool Re-rack Criticality Analysis
2. Regulatory Commitment List

cc: NRC Region I Administrator
NRC Resident Inspector Office
NRC Project Manager
Director BRP/DEP
Site Representative (BRP/DEP)

The Nuclear Regulatory Commission (NRC) staff provided a request for additional information (RAI) regarding the FirstEnergy Nuclear Operating Company (FENOC) license amendment request (LAR) for Unit 2 of the Beaver Valley Power Station (BVPS-2), regarding criticality analyses in support of a Unit 2 spent fuel pool re-rack. The NRC staff requests are presented below in bold type, followed by the FENOC response.

1. **Does BVPS-1 and 2 share any resources related to fresh or spent fuel handling and storage? If so, it may be necessary to evaluate related normal and potential abnormal conditions.**

Response

BVPS-1 and 2 have separate spent fuel storage pools in physically separated buildings. The units do not share spent fuel storage pools for fresh or spent fuel storage. There is no equipment to transfer spent fuel between the buildings that house the spent fuel storage pools.

BVPS-1 and 2 fresh and spent fuel handling operations share other resources such as personnel, tools and equipment, and vendor-supplied personnel and equipment, which do not impact the criticality analysis.

2. **Table 2.5.1 provides some key storage rack design information, but does not provide enough information to assess the adequacy of the storage rack model. Provide the dimensions and tolerances from the bottom of the storage cell to the bottom of the storage rack absorber plates.**

Response

The distance from the bottom of the storage cell to the bottom of the storage rack absorber plates is 2.875 +/- 0.125 inches.

3. **In Section 3.4, on page 3-7, the first bulleted item states, "A decrease of no more than 5% in Boron-10 (B-10), as determined by neutron attenuation, is acceptable. (This is equivalent to a requirement for no loss in boron within the accuracy of the measurement.)" This statement implies that [it] is acceptable to ignore a loss of up to 5% of the B-10 in the Metamic. It is not clear from the text provided in Section 4 that this acceptable loss of B-10 has been considered in the analysis.**

Amend the analysis to consider the "acceptable" loss of B-10 or provide justification for not doing so.

Response

This request for additional information will be resolved in future correspondence.

4. **Section 4.5.3 describes the method used to calculate the axial burnup distribution used in the criticality analysis. From the text and Table 4.5.5, it is not clear whether this was done using core average axial burnup or if individual assembly burnup distributions were used. Use of core average axial burnup would minimize the importance of outliers that may have a significantly different axial burnup shape, and therefore, may not be appropriate.**

   Confirm that assembly-specific axial burnup distributions were utilized, or justify the use of core average axial burnup distributions.

Response

Assembly-specific axial burnup distributions were not utilized in the license amendment request (LAR). The axial burnup distributions described in Section 4.5.3 and Table 4.5.5 of the LAR were based on core average axial burnup distributions. The burnup profile shown in Table 4.5.6 in the LAR was developed based on the methodology explained in response to RAI item 5. However, axial burnup profiles developed from a population of axial burnup profiles based on assembly-specific data more accurately reflect the reactivity of a fuel assembly. Therefore, new axial burnup profiles were developed in response to this RAI, using the same methodology that was used to develop the axial burnup profiles in the LAR (as discussed in response to RAI item 5) but that are instead based on fuel assembly-specific burnup information.

The population of fuel assemblies that was used to develop the new axial burnup profiles was based on end of cycle core follow calculations that modeled every fuel cycle operated to date at Beaver Valley Unit 2. The end of cycle fuel assembly data include once burned and twice burned fuel assemblies, in addition to data for fuel assemblies being permanently discharged to the spent fuel pool.

Three new profiles were developed based on their axial enrichment type:

- A profile for assemblies that have enriched (2.6 wt% U-235) blankets,
- A profile for assemblies that have natural (0.71 wt% U-235) blankets,
- A profile for assemblies that have no blankets (constant axial enrichment).

LAR Profile (Table 4.5.6)	Enriched Blankets Profile	Natural Blankets Profile	No Blankets Profile
0.2089	0.3495	0.1537	0.4430
0.7163	0.7140	0.6518	0.6803
0.8480	0.8497	0.8259	0.8203
0.9658	0.9634	0.9722	0.9485
1.0250	1.0175	1.0384	1.0108
1.0405	1.0380	1.0550	1.0283
1.0560	1.0601	1.0713	1.0458
1.0736	1.0663	1.1075	1.0801
1.0853	1.0829	1.1129	1.0852
1.0610	1.0700	1.0818	1.0548
1.0884	1.0866	1.1154	1.0928
1.0911	1.0868	1.1159	1.0946
1.0808	1.0750	1.0992	1.0716
1.0973	1.0868	1.1141	1.0912
1.1020	1.0876	1.1163	1.1036
1.1020	1.0854	1.1152	1.0975
1.0850	1.0763	1.1041	1.0704
1.1112	1.0898	1.1203	1.0967
1.1107	1.0879	1.1180	1.0910
1.0638	1.0646	1.0858	1.0471
1.0639	1.0542	1.0722	1.0437
0.9915	0.9935	0.9729	0.9804
0.7916	0.8293	0.7304	0.7983
0.2217	0.4000	0.1746	0.5084

The three new profiles shown above will be used in the revised criticality analysis. When used in the revised criticality analysis, the profiles will be applied to fuel with the corresponding axial enrichment distribution. Therefore, the revised analysis will have three sets of loading curves, one for each profile. A future supplement to the LAR will reflect this revised criticality analysis.

To demonstrate the reactivity effect of the new profiles, sensitivity calculations were performed using the three profiles above as well as the profile from the LAR. The sensitivity studies were performed using a single rack cell model in infinite lattice geometry over a range of initial enrichment and burnup combinations.

For each specific calculation, the axial enrichment distribution is modeled (for example, for the enriched blanket profile, the end nodes have an enrichment of 2.6 weight percent while the middle nodes are at the weight percent in the left column). Therefore, the profile used is applied appropriately to the fuel assembly group from which it was developed, and the importance of the end nodes is maintained. These calculations are presented in the table below.

wt% U-235	End Nodes Enrichment	gwd/mtu	Profile	Input File	kcalc	2*σ
3	2.6	10	Enriched Blankets	prb3010	0.9834	0.0014
3	0.7	10	Natural Blankets	prc3010	0.9803	0.0010
3	3	10	No Blankets	prd3010	0.9853	0.0012
3	3	10	LAR Profile	pre3010	0.9834	0.0014
5	2.6	55	Enriched Blankets	prb5055	0.8668	0.0014
5	0.7	55	Natural Blankets	prc5055	0.8615	0.0014
5	5	55	No Blankets	prd5055	0.8973	0.0014
5	5	55	LAR Profile	pre5055	0.9127	0.0016

A process will be established prior to receipt of the next reload batch of BVPS Unit 2 fuel to ensure that the design features and operating parameters of fuel used in the future at BVPS Unit 2 are consistent with the assumptions of the criticality analysis.

5. The analysis takes the lowest relative burnup at each node to create a fictitious profile for use. This makes the relative burnup for the entire fuel assembly less than 1.00. In this particular case, the total relative burnup is 0.96, which means the fuel assembly in the analysis actually has 4% less burnup than is stated. A fuel assembly with 30 GWD/MTU actually has 28.8 GWD/MTU. Attributing the k-effective (k_{eff}) of a fuel assembly with 28.8 GWD/MTU to one with 30 GWD/MTU should be conservative. However, it is not clear that the derived profile is actually a limiting profile. It is not clear that one of the actual, or future, profiles may be more reactive, even with the slightly higher burnup. According to various tables in the analysis 1 GWD/MTU is worth roughly a half percent Δk_{eff}. According to NUREG/CR-6801 the axial profile can be worth several percent Δk_{eff}.

Demonstrate that the method of creating the profile is bounding.

Response

In response to RAI item 4, it was discussed that in the LAR, core average burnup profiles were developed and used. The response to RAI item 4 also states that in the revised criticality analysis, assembly-specific burnup profiles will be used. Both the axial burnup profile in the LAR and the new assembly-specific profiles

discussed in the response to RAI item 4 were created using the same methodology. This methodology is therefore explained in this response in a general manner.

The method used to create the axial burnup profiles is consistent with the guidance provided in NUREG/CR-6801. Bounding axial burnup profiles are developed by determining the limiting burnup for each axial node. This is done by first normalizing each of the axial burnup profile nodes in the population being used to develop the bounding profile to a value of 1.0 by dividing the nodal burnup by the assembly's average burnup. The next step was to search the entire population of normalized profiles to find the minimum normalized burnup for each of the 24 axial nodes. This created a composite axial burnup profile for which the normalized axial burnup value for any axial node is less than or equal to the normalized axial burnup value for that axial node over the fuel in the population. The bounding axial burnup profile that results from this method is not re-normalized to an average value of 1.0.

The effect of not re-normalizing to 1.0 is demonstrated in the following example using the axial burnup profile in the LAR, applied to a fuel assembly with an average burnup of 30 GWD/MTU.

[Response to RAI item 5 is continued on the next page]

The table below shows three scenarios. The first scenario applies the bounding axial burnup profile from the LAR against a 30 GWD/MTU fuel assembly. In the second scenario, the bounding profile has been re-normalized to an average value of 1.0, which preserves the modeled assembly-average burnup of 30 GWD/MTU. In the third scenario, the relative burnup profile from the LAR has been preserved for the top and bottom 6 nodes, but the middle 12 nodes have been set to a constant value such that the relative axial profile still produces an average value of 1.0, again forcing the assembly-average burnup to remain at 30 GWD/MTU.

	Scenario 1 – Limiting Profile from Licensing Report		Scenario 2 – Re-normalized Limiting Profile		Scenario 3 – Flat Center Profile	
	Shape	Burnup	Shape	Burnup	Shape	Burnup
Top 24	0.2089	6.267	0.2172	6.516	0.2089	6.267
23	0.7163	21.489	0.7448	22.344	0.7163	21.489
22	0.8480	25.440	0.8817	26.452	0.8480	25.440
21	0.9658	28.974	1.0042	30.127	0.9658	28.974
20	1.0250	30.750	1.0658	31.974	1.0250	30.750
19	1.0405	31.215	1.0819	32.457	1.0405	31.215
18	1.0560	31.680	1.0980	32.941	1.1627	34.881
17	1.0736	32.208	1.1163	33.490	1.1627	34.881
16	1.0853	32.559	1.1285	33.855	1.1627	34.881
15	1.0610	31.830	1.1032	33.097	1.1627	34.881
14	1.0884	32.652	1.1317	33.951	1.1627	34.881
13	1.0911	32.733	1.1345	34.036	1.1627	34.881
12	1.0808	32.424	1.1238	33.714	1.1627	34.881
11	1.0973	32.919	1.1410	34.229	1.1627	34.881
10	1.1020	33.060	1.1459	34.376	1.1627	34.881
9	1.1020	33.060	1.1459	34.376	1.1627	34.881
8	1.0850	32.550	1.1282	33.845	1.1627	34.881
7	1.1112	33.336	1.1554	34.663	1.1627	34.881
6	1.1107	33.321	1.1549	34.647	1.1107	33.321
5	1.0638	31.914	1.1061	33.184	1.0638	31.914
4	1.0639	31.917	1.1062	33.187	1.0639	31.917
3	0.9915	29.745	1.0310	30.929	0.9915	29.745
2	0.7916	23.748	0.8231	24.693	0.7916	23.748
Bottom 1	0.2217	6.651	0.2305	6.916	0.2217	6.651
Avg.	**0.9617**	**28.852**	**Avg.** **1.0000**	**30.000**	**Avg.** **1.0000**	**30.000**

Shape values are dimensionless quantities. Burnup values are in GWD/MTU.

As can be seen from the above table, the use of the limiting profile (Scenario 1) always produces absolute burnups for each individual axial node that are at least as low, and, in many cases, lower, than in either of the other two scenarios. This conclusion will be true at all burnups, not just at 30 GWD/MTU. Thus, the axial burnup profile in Scenario 1 will always produce higher reactivities than those from the other scenarios, and is, therefore, conservative.

Although the actual average burnup of the fuel assembly in Scenario 1 is 28.852 GWD/MTU in the criticality analysis model, it is treated, for the purposes of determining the final burnup versus enrichment curves for fuel assembly storage categorization, as a fuel assembly with an average burnup of 30 GWD/MTU, thus preserving the conservatism of this methodology. Therefore, the methodology used to develop the axial burnup profiles that were used in the LAR and that will be used in the revised criticality analysis is bounding.

6. **In Section 4.5.5, first paragraph, the last sentences states, "Neutron absorber panels are installed on all exterior walls facing other racks." This statement seems to imply that absorber panels are not installed on exterior walls that are not facing other racks.**

   The text in Section 4.7.13 stated that the mislocated assembly is modeled adjacent to two poisoned faces of the racks. A higher k_{eff} value may be calculated if the mislocated assembly is modeled next to a fuel assembly in a position that does not have an external [~~as~~] poison panel. Confirm that all external faces have poison panels or justify the model used for the mislocated assembly analysis conservatively bound[s] the other possible locations.

Response

The text in Section 4.5.5 should have stated: "Neutron absorber panels are installed on all exterior walls facing other racks and along the exterior of any rack location where it is physically possible to mislocate a fuel assembly." Section 4.7.13 is therefore correct. A future supplement to the LAR will reflect the revised wording of Section 4.5.5.

7. **The text in Section 4.5.6 stated that racks are separated by a minimum 1.5-inch gap, but does not clearly define this dimension. This might be from the outside of the sheathing that holds the Metamic in place or it might be from the outside of each stainless steel box. Provide a clear definition of the 1.5-inch gap between racks and confirm that this definition is used in the calculations.**

Response

The 1.5-inch gap between racks is defined by the 0.75-inch (+0.1875, –0.0 inch) base plate extensions and therefore is the rack wall to rack wall gap between the stainless steel boxes. The calculational models have been verified to represent this geometry correctly.

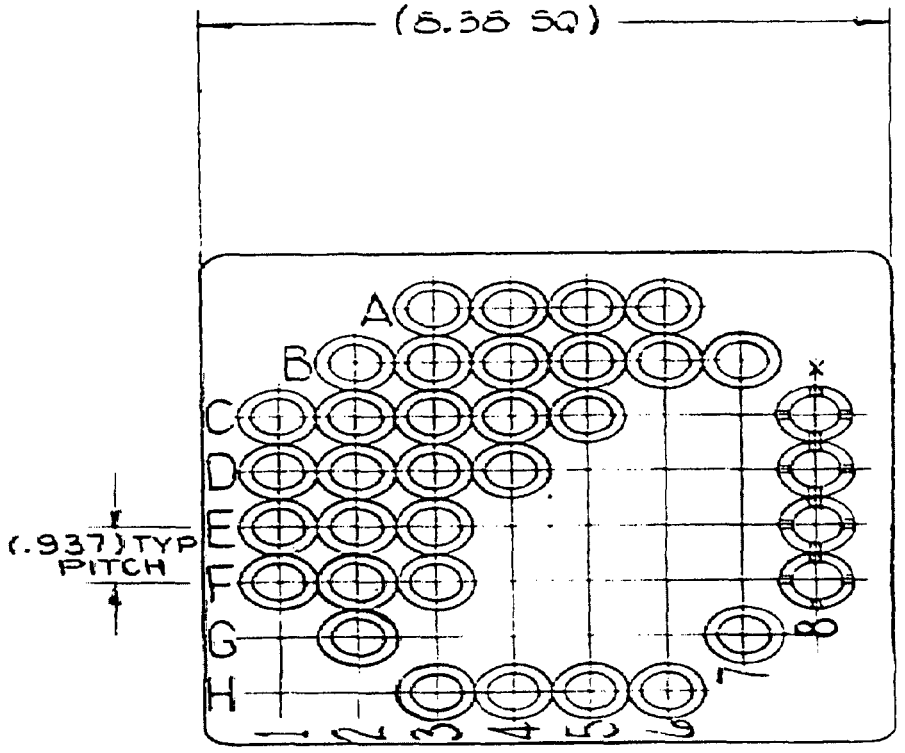
8. In Section 4.5.7, insufficient information is provided to justify the model used for the fuel rod storage basket (FRSB). Provide additional description of the FRSB and explain why the FRSB model used is conservative.

Response

The fuel rod storage basket contains two tube designs:

1. 48 tubes with an outer diameter of 0.625 inch, a wall thickness of 0.035 inch, and a length of 149.375 inches.
2. 4 tubes with an outer diameter of 0.75 inch, a wall thickness of 0.049 inch, and a length of 149.375 inches.

As shown in the figure below, the tube pitch is 0.937 inch. The tubes are housed in a 159.17 inches high by 8.38 inches long by 8.38 inches wide frame. The tube and frame materials are stainless steel.



Note: Units are inches.

The MCNP4a calculational model conservatively neglects the stainless steel structural material, including the tubes and the steel can that surrounds all of the tubes (the structural material would lower reactivity because it absorbs neutrons and displaces moderator), and assumes all pin locations within the fuel rod storage basket (FRSB) contain fresh fuel rods at 5.0 weight-percent (wt%) U-235 (without burnable absorbers). The model uses reflective boundary conditions outside of water (at least 12 inches surrounding on all sides). This same water reflective boundary conditions model is also applied to the other fuel assemblies used in the comparison shown in Table 4.7.20 of the Holtec report provided as Enclosures B and C of the April 9, 2009 amendment request. Table 4.7.20 therefore shows that the FRSB has a lower reactivity than any fuel assembly categorized for storage in the Spent Fuel Pool (SFP). Therefore, the FRSB can be placed in any storage cell location.

9. **While it is unlikely to prove limiting, a case with all assemblies moved away from the storage rack center should have been evaluated in Section 4.7.6. This would move the highest reactivity fuel (5 wt% fresh fuel) in four adjacent rack corners toward each other. Expand the analysis to include this additional case or justify not doing so.**

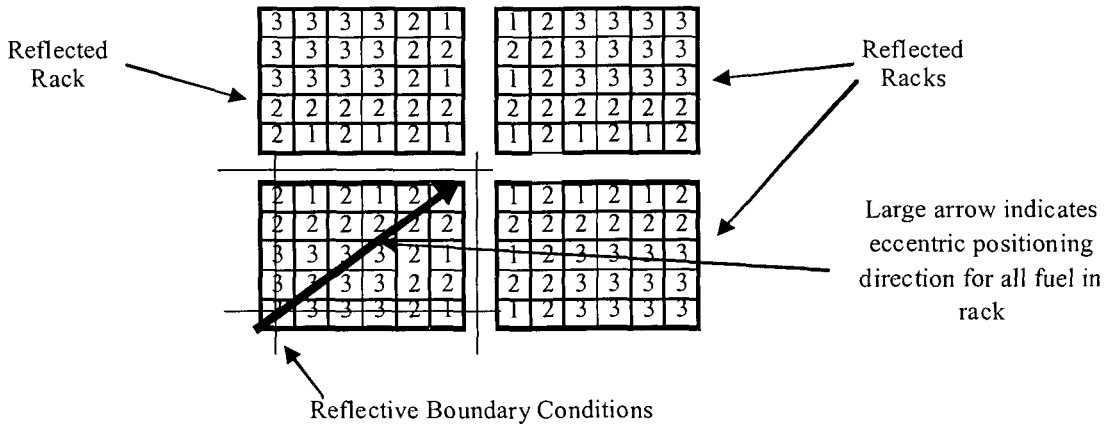
Response

This case was analyzed in the submitted analysis in Case 1. The text in Section 4.7.6 could have been more clearly worded as: "Case 1 models all the fuel assemblies eccentrically positioned away from the rack center so that there are four fresh assemblies in the four rack corners at their closest approach." A future supplement to the LAR will reflect the revised wording of Section 4.7.6.

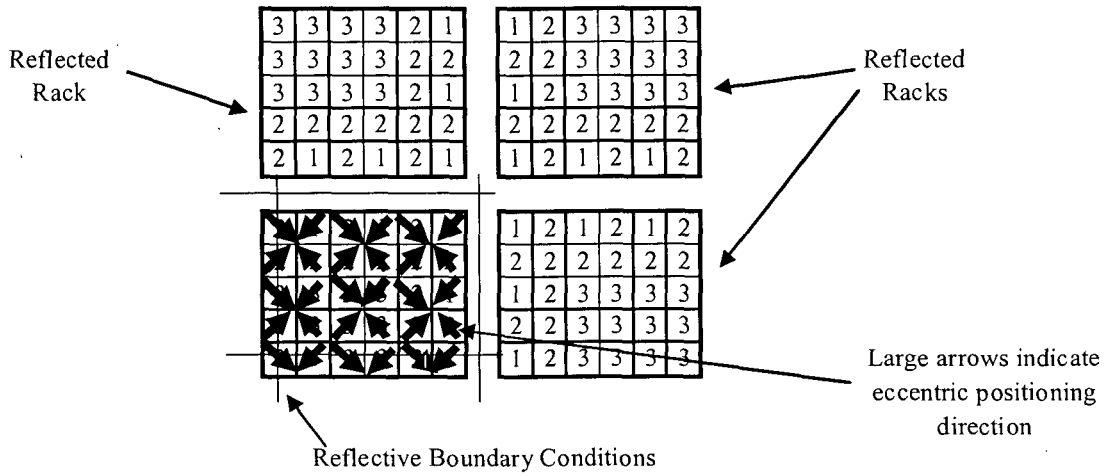
For clarification, the following *partial* figure of Eccentric Case 1 is provided, with the point of the arrow being the location having four rack corners.

Additional Figures Showing the Eccentric Fuel Positioning Cases

First Configuration, Case 1 Eccentric Positioning



Second Configuration, Case 2 Eccentric Positioning



10. Section 4.7.13.4 covers abnormal location of a fuel assembly. During the time when both old and new racks exist in the pool at the same time, there is the potential for the erroneous storage of a fuel assembly in the new racks that was intended for the old racks and for the erroneous storage of a fuel assembly in the old racks that was intended for the new racks.

Evaluate or provide justification for not including evaluation of potential abnormal conditions related to storage of fuel intended for the new racks in the old racks and to storage of fuel intended for the old racks in the new racks.

Response

The erroneous placement of fuel in the new racks that was intended for the old racks or vice versa is already considered by the misloaded cases previously performed in Section 4.7.13.4.1 of the analysis for the new racks and those performed in the current analysis of record for the old racks. The fuel that is intended for storage in the old racks or new racks cannot have an enrichment greater than the 5 weight percent assumed in the fresh fuel assembly misloaded accident considered in the analysis, due to Technical Specification limitations.

11. Section 4.7.17, "Interim Configurations," described measures taken to isolate fuel stored in old racks from fuel stored in new racks. There is some potential that this administrative control will be violated. Evaluate or provide justification for not evaluating the erroneous placement of fuel in the prohibited two empty rows between old and new racks.

Response

The erroneous placement of fuel in the prohibited two empty rows between old and new racks is bounded by the mislocated cases already performed in Section 4.7.13.4.2 of the analysis for the new racks and the analysis of record for the old racks. Although the geometry of the mislocated cases that were evaluated is not the same, the mislocated fuel assembly is directly face adjacent to two fresh fuel assemblies in the mislocated model. Additionally, the poison along those two interfaces is modeled at half thickness. Therefore, a misplacement of a fuel assembly in the prohibited two empty rows would not exceed the k_{eff} for the mislocated cases because there is neutron absorber in every cell wall surrounding the extra fuel assembly, whereas in the mislocated case the poison is only modeled at half thickness along two faces. In the LAR, Table 4.7.18 shows that the mislocated accident would require 1179 parts per million (ppm) soluble boron. The minimum Technical Specification fuel pool storage boron concentration of 2000 ppm will be much greater than the accident scenario minimum soluble boron concentration of 1179 ppm for the mislocated accident scenario to maintain k_{eff} less than 0.95, providing 821 ppm of margin (more than sufficient to accommodate any potential additional positive reactivity effect from mislocating a fuel assembly in the two empty rows). Also, the gap between the old and new racks results in more decoupling of the bulk fuel in the racks than would be provided by just the two empty rows, providing additional conservatism.

12. Table 4.5.2 describes the design basis fuel assembly. Provide the distance and tolerance from the bottom of the fuel assembly to the bottom of the active fuel. If this distance varies with fuel design, provide design-specific information.

Response

Based on Westinghouse fuel drawings, the distances from the bottom of the bottom nozzle to the bottom of the active fuel, neglecting any gap between end plug and bottom nozzle, are provided in the following table for the fuel assembly designs used at BVPS-2.

Fuel Assembly Regions	Distance (in.)	Tolerance (in.)
1-3	3.426	±0.015
4-5	2.894	±0.015
6-8	2.813	±0.011
9-17	3.193	±0.011

The minimum distance from the bottom of the bottom nozzle to the seated fuel rod end plug is 2.813 inches for fuel assembly Region 6 through 8. The distance from the top of the base plate to the bottom of the absorber panel is 2.875 inches. Assuming a worst-case stack up of tolerances, this leaves a maximum non-overlap distance of 0.198 inches (i.e. (2.813 – 0.011) – (2.875 + 0.125)) between the active fuel and the neutron absorber panel. Since Region 6 through 8 had six-inch natural blankets, this area is an area of low importance, and therefore this area has a low sensitivity to the reactivity effect of the hypothetical non-overlap.

13. In Table 4.5.2, the value used positive tolerance on fuel rod pitch appears to be crediting that the fuel rod pitch will vary randomly across a row of 17 fuel rods. Since there are only 16 pin pitches across a fuel assembly, shouldn't the tolerance have been 0.0075 divided by 16 rather than 17?

Revise the analysis to correctly incorporate the fuel rod pitch tolerance or justify not doing so.

Response

The criticality analysis will be updated to use 16 rather than 17. A future supplement to the LAR will reflect this revision to the analysis.

14. Table 4.5.3 provides the core operating parameters used for CASMO depletion calculations. Was the analysis performed to show that these parameters are conservative and bounding? Note that consideration should be given to the full range of parameters experienced by all fuel currently stored and for fuel that will be stored in the future. Consider too that parameters that lead to spectral hardening and increased plutonium production also reduce depletion of thermal neutron

absorbing fission products. It should not just be assumed that anything that hardens the spectrum is conservative.

Provide the ranges of operating parameters affecting the CASMO depletion calculations and provide better justification for bounding values selected.

Response

This request for additional information will be resolved in future correspondence.

15. **Table 4.5.4 lists the calculated burnup-dependent soluble boron concentrations for BVPS-2 Cycle 14. It is not clear from the text why using these calculated values is appropriate. More appropriate would be the measured boron letdown curve or the predicted boron letdown curve from the nuclear design report. Beyond that issue, consideration should also be given to the boron letdown curves from previous cycles. The bounding value used for the depletion analysis should be bounding for all prior cycles and should also bound future cycles.**

    Confirm that the boron letdown curve provided in Table 4.5.4 bounds past measured boron letdown curves and expected future boron letdown curves.

Response

Table 4.5.4 listed projected soluble boron concentrations for BVPS-2 Cycle 14, because at the time of the development of this criticality analysis, Cycle 14 was still in the design phase. Also, Cycle 14 was an uprate cycle to 2900 MWth from 2770 MWth, and therefore this approach was utilized to account for the uprated operating conditions.

A review of the measured and the Westinghouse Nuclear Design Report cycle-averaged soluble boron concentrations for BVPS-2 Cycles 1-14 has been performed to confirm that the value used in the analysis was bounding, as summarized in the table below. As shown in the table on the next page, the maximum measured cycle-averaged soluble boron concentration is 1013 ppm. NUREG/CR-6800 recommends using the maximum cycle-averaged soluble boron concentration. The maximum measured average soluble boron concentration (measured values for Cycles 12-14 are B-10 depletion corrected) is bounded by the 1050 ppm used for the depletion analysis (the 1050 ppm was developed using a straight average of the values from Table 4.5.4 in the LAR).

Cycle Number	Nuclear Design Report (ppm)	Measured Cycle-averaged Soluble Boron Concentration (ppm)
1	568	n/a
2	579	n/a
3	712	n/a
4	679	n/a
5	649	707
6	677	727
7	741	785
8	750	756
9	749	810
10	953	**1013**
11	916	1002
12	884	954
13	885	938
14	895	915

A process will be established prior to receipt of the next reload batch of BVPS Unit 2 fuel to ensure that the design features and operating parameters of fuel used in the future at BVPS Unit 2 are consistent with the assumptions of the criticality analysis.

16. Table 4.5.10 provides key storage cell parameters and tolerances. Provide or justify not providing information on the following rack parameters:

a. Steel density, including tolerance.

Response: The steel density used is a conservatively minimum value of 7.84 g/cc and therefore no tolerance calculation is needed.

b. Dimensions and tolerances associated with "corner angles" and "filler panels" (see figure 2.6.2), and associated with "developed cell" dimensions and tolerances. Note from Figure 2.6.4 that the exterior poison panels appear to be shorter than interior poison panels.

Response: The dimensions and tolerances of these items are not utilized in the analysis. These figures are intended to show the construction of a rack in general and therefore may differ somewhat from the actual racks. The analysis does not consider "corner angles", "filler panels" or "developed cells" explicitly; instead it considers the cell wall, poison, and

sheathing along with their tolerances. All other structural components are conservatively neglected and the consideration of the storage cell tolerances covers any tolerances related to cell formation, regardless of whether the cell was "developed" or "fabricated". The neutron absorber panels on the exterior of the racks are identical (i.e., same width, length and axial location, thickness, and boron carbide content) to those installed in the interior of the racks.

17. **Table 4.7.2 and Figure 4.7.2 both show a minimum required burnup of 0.00 GWD/MTU for fuel initially enriched to 2.0 wt%. The equation under Table 4.7.2 yields a minimum burnup of 0.52 GWD/MTU. The inconsistency between the equation under Table 4.7.2 and the data presented in Table/Figure 4.7.2 should be eliminated.**

    Revise Table 4.7.2 and Figure 4.7.2, the equation provided under Table 4.7.2 and any associated text to eliminate inconsistency in required minimum burnup.

Response

A future supplement to the LAR will eliminate such inconsistencies.

18. **Section 4.2.1, first paragraph -the last sentence states: "This approach has been validated in [4.3] by showing that the cross sections result in the same reactivity in both CASMO-4 and MCNP4a."**

    This is not "validation". This cross-code comparison utilizing the same cross sections does not tell us anything about the potential composition and cross-section errors and associated biases introduced by the modeling of lumped fission products and use of lumped fission product cross sections.

    What is the worth of the lumped fission products in the fuel storage racks? What fission products are included in the lumped fission products?

    The "5% of the reactivity decrement" suggested by the Kopp memo does not cover modeling simplifications and approximations such as use of lumped fission products.

    Eliminate or provide better justification for the use of lumped fission products. Justification for the use of the "lumped fission product"

modeling simplification should include quantification of associated bias and bias uncertainty.

Response

This request for additional information will be resolved in future correspondence.

19. **Section 4.2.1, second paragraph - The uncertainty stated in this paragraph (i.e. 0.0011) is the 95/95 confidence level uncertainty on the "mean" k_{eff} value calculated for the critical experiments modeled in the validation study. Note that this is not the 95/95 confidence on the distribution of the sample population (the k_{eff} values calculated for the critical experiments).**

    It looks like the variance of the population has not been included in the analysis. This is important because the requirement is that we have a 95% confidence with a 95% probability that any single calculation that calculates as subcritical is indeed subcritical. Thus, the sample variance about the bias is the relevant quantity, not the variance of the bias. The 95/95 uncertainty used to determine the acceptable k_{eff} values would rise from 0.0011 to 0.0082.

    Review the statistical approach used to determine the acceptable k_{eff} values and either clarify how the variance of the sample population was included, or revise the statistical approach used to include consideration of the sample population variance, or justify not doing so.

Response

A future supplement to the LAR will reflect revision of the statistical approach to include consideration of the sample population variance.

20. **The text in Section 4.7.4, "Isotopic Compositions," describes modeling approximations related to calculation and use of burned fuel compositions. Provide additional detail in this section to clearly state the conditions used in the calculation of fuel compositions as a function of initial-enrichment and fuel burnup.**

    From the text in Section 4.7.4, it appears that assembly average compositions are used rather than pin-by-pin compositions. This minimizes the impact on reactivity of removing WABA rods from an assembly. During depletion, the WABA rods depress power/burnup locally until the WABA rods are removed. Provide additional detail describing and justifying the use of modeling approximations

associated with modeling burned fuel. Where appropriate, include biases and bias uncertainties associated with the modeling simplifications and approximations.

Response

This request for additional information will be resolved in future correspondence.

21. Section 4.5.4 provided a discussion of fuel assembly reactivity control devices. Confirm that integral (e.g. IFBA) and non-integral (e.g., WABA, WDR, RCCAs, etc.) reactivity control devices are modeled during depletion and removed prior to restart in rack geometry.

Response

No burnable absorbers, integral or discrete, are modeled in the fuel during the depletion calculations that are used to determine spent fuel isotopic compositions. As a result, removal prior to restart is not necessary.

22. As described in Section 4.7 and in other places in the report, CASMO-4 was used to provide rack k_{eff} sensitivity information. Considering that the storage rack is constructed of "fabricated" cells and "developed" cells (see Fig. 2.6.2), it is not obvious what the CASMO model looked like. Did CASMO model simplifications affect the results?

Describe the CASMO model used for fuel storage rack models. Where appropriate, include biases and uncertainties associated with CASMO model simplifications.

Response

This request for additional information will be resolved in future correspondence.

23. Section 4.7.1 included a discussion justifying not modeling spacer grids. Table 4.7.5 provided some burnup-dependent results for models with and without spacer grids. The initial enrichment used was not clear from the text. Provide additional information regarding the initial enrichment(s) used for these calculations. Describe the impact of varying initial enrichments on the results.

The text in Section 4.7.1 stated, "it is conservative to neglect the fuel assembly spacer grids..." Conservative is not the right word. One might argue that the impact of the grids is negligibly small, but not conservative. The last sentence in Section 4.7.1 implies that not

modeling the spacer grids yields a conservatism of approximately 0.01 Δk. As was pointed out earlier in Section 4.7.1, the effect is overestimated in the study, and the conservatism for fresh fuel is likely significantly less than 0.01 Δk. Revise the text in Section 4.7.1 to more accurately describe the impact of not modeling spacer grids.

Response

The spacer grid sensitivity calculations were performed at 5.0 weight percent U-235 fuel. Additional calculations for 2.0 and 3.5 weight percent U-235 are shown below. As shown in the table below, the impact of these additional enrichments is consistent with the 5.0 weight percent case.

A future supplement to the LAR will revise the text in Section 4.7.1, specifically the last sentence, to read: "Therefore, it is acceptable to neglect the fuel assembly spacer grids… where the absence of modeling the grid spacers results in negligible differences in reactivity that are more than offset by the soluble boron in the pool that is not credited (i.e. approximately 800 ppm)."

CASMO-4 Calculation of the Effect of Spacer Grids and Boron Concentration on Reactivity

Enrichment wt% U-235		2			3.5			5		
Burnup (GWD/MTU)	Boron (ppm)	Reference Case Input File	Spacer Grid Case Input File	Delta-k	Reference Case Input File	Spacer Grid Case Input File	Delta-k	Reference Case Input File	Spacer Grid Case Input File	Delta-k
0	400	bvgr400-b-20	bvgr400-a-20	-0.0077	bvgr400-b-35	bvgr400-a-35	-0.0098	bvgr400-b-50	bvgr400-a-50	-0.0107
	800	bvgr800-b-20	bvgr800-a-20	-0.0049	bvgr800-b-35	bvgr800-a-35	-0.0071	bvgr800-b-50	bvgr800-a-50	-0.0083
	1000	bvgr1000-b-20	bvgr1000-a-20	-0.0037	bvgr1000-b-35	bvgr1000-a-35	-0.0060	bvgr1000-b-50	bvgr1000-a-50	-0.0072
	1200	bvgr1200-b-20	bvgr1200-a-20	-0.0027	bvgr1200-b-35	bvgr1200-a-35	-0.0050	bvgr1200-b-50	bvgr1200-a-50	-0.0063
	1400	bvgr1400-b-20	bvgr1400-a-20	-0.0018	bvgr1400-b-35	bvgr1400-a-35	-0.0041	bvgr1400-b-50	bvgr1400-a-50	-0.0054
	1600	bvgr1600-b-20	bvgr1600-a-20	-0.0010	bvgr1600-b-35	bvgr1600-a-35	-0.0032	bvgr1600-b-50	bvgr1600-a-50	-0.0046
	1800	bvgr1800-b-20	bvgr1800-a-20	-0.0003	bvgr1800-b-35	bvgr1800-a-35	-0.0024	bvgr1800-b-50	bvgr1800-a-50	-0.0038
	2000	bvgr2000-b-20	bvgr2000-a-20	0.0004	bvgr2000-b-35	bvgr2000-a-35	-0.0017	bvgr2000-b-50	bvgr2000-a-50	-0.0031
20	400	bvgr400-b-20	bvgr400-a-20	0.0000	bvgr400-b-35	bvgr400-a-35	-0.0055	bvgr400-b-50	bvgr400-a-50	-0.0085
	800	bvgr800-b-20	bvgr800-a-20	0.0024	bvgr800-b-35	bvgr800-a-35	-0.0030	bvgr800-b-50	bvgr800-a-50	-0.0062
	1000	bvgr1000-b-20	bvgr1000-a-20	0.0034	bvgr1000-b-35	bvgr1000-a-35	-0.0020	bvgr1000-b-50	bvgr1000-a-50	-0.0051
	1200	bvgr1200-b-20	bvgr1200-a-20	0.0042	bvgr1200-b-35	bvgr1200-a-35	-0.0010	bvgr1200-b-50	bvgr1200-a-50	-0.0042
	1400	bvgr1400-b-20	bvgr1400-a-20	0.0050	bvgr1400-b-35	bvgr1400-a-35	-0.0001	bvgr1400-b-50	bvgr1400-a-50	-0.0033
	1600	bvgr1600-b-20	bvgr1600-a-20	0.0056	bvgr1600-b-35	bvgr1600-a-35	0.0007	bvgr1600-b-50	bvgr1600-a-50	-0.0025
	1800	bvgr1800-b-20	bvgr1800-a-20	0.0062	bvgr1800-b-35	bvgr1800-a-35	0.0014	bvgr1800-b-50	bvgr1800-a-50	-0.0018
	2000	bvgr2000-b-20	bvgr2000-a-20	0.0067	bvgr2000-b-35	bvgr2000-a-35	0.0021	bvgr2000-b-50	bvgr2000-a-50	-0.0011
40	400				bvgr400-b-35	bvgr400-a-35	0.0010	bvgr400-b-50	bvgr400-a-50	-0.0041
	800				bvgr800-b-35	bvgr800-a-35	0.0033	bvgr800-b-50	bvgr800-a-50	-0.0017
	1000				bvgr1000-b-35	bvgr1000-a-35	0.0043	bvgr1000-b-50	bvgr1000-a-50	-0.0007
	1200				bvgr1200-b-35	bvgr1200-a-35	0.0051	bvgr1200-b-50	bvgr1200-a-50	0.0002
	1400				bvgr1400-b-35	bvgr1400-a-35	0.0059	bvgr1400-b-50	bvgr1400-a-50	0.0010
	1600				bvgr1600-b-35	bvgr1600-a-35	0.0066	bvgr1600-b-50	bvgr1600-a-50	0.0018
	1800				bvgr1800-b-35	bvgr1800-a-35	0.0072	bvgr1800-b-50	bvgr1800-a-50	0.0025
	2000				bvgr2000-b-35	bvgr2000-a-35	0.0078	bvgr2000-b-50	bvgr2000-a-50	0.0031
60	400							bvgr400-b-50	bvgr400-a-50	0.0019
	800							bvgr800-b-50	bvgr800-a-50	0.0041
	1000							bvgr1000-b-50	bvgr1000-a-50	0.0050
	1200							bvgr1200-b-50	bvgr1200-a-50	0.0059
	1400							bvgr1400-b-50	bvgr1400-a-50	0.0066
	1600							bvgr1600-b-50	bvgr1600-a-50	0.0073
	1800							bvgr1800-b-50	bvgr1800-a-50	0.0079
	2000							bvgr2000-b-50	bvgr2000-a-50	0.0085

24. Section 4.7.2 addressed the reactivity effect of fuel assembly reactivity control devices. There are three issues associated with the analysis described in this section:

a. It does not appear that depletion with control rods present was considered. Frequently, second-and third-cycle fuel is placed under control rods, some of which may be used to control reactor power. Thus a realistic fuel depletion scenario could include first cycle depletion with burnable absorbers and second cycle of depletion with partially inserted control rods. Some plants have also included part-length absorbers rods in some peripheral locations to reduce neutron flux to reactor vessel welds. [In] low-leakage loading patterns, these peripheral locations frequently hold fuel that is being used for a third cycle.

Provide justification for modeling reactivity control devices throughout fuel assembly life.

Response: BVPS-2 typically operates under normal operating conditions with control rods all out (above the top of the active fuel length). Therefore, the control rods are not inserted during normal operation for a sufficient amount of time to have a significant impact on fuel depletion. The potential impact of part-length absorber rods will be resolved in future correspondence.

b. It appears that IFBA and WABA were not modeled during depletion as being present in the same assembly. Does some mechanical feature or technical specification prevent them from being in the same assembly? Provide justification for not modeling both in the same assembly.

Response: BVPS-2 has not used and does not plan to use both WABA and IFBA in the same fuel assembly. As a result, no modeling is necessary. No mechanical features or technical specifications prevent them from being in the same assembly, however, please see the commitment that follows sub-item c. at the end of this response.

c. Some plants have used standard Pyrex glass burnable absorbers. These absorbers deplete more slowly and can result in increased plutonium generation. Confirm that standard Pyrex glass burnable absorbers have not been used at BVPS-2. If they have been used, revise the analysis to address such use.

Response: BVPS-2 has not used and does not plan to use Pyrex glass burnable absorbers.

A process will be established prior to receipt of the next reload batch of BVPS Unit 2 fuel to ensure that the design features and operating parameters of fuel used in the future at BVPS Unit 2 are consistent with the assumptions of the criticality analysis.

25. The last sentence of Section 4.7.4 is "For conservatism, the isotopic concentrations are determined at zero cooling time, without xenon and [....]." Provide a reference or justification for the assertion that use of such isotopic concentrations is conservative.

Response

Please see NUREG/CR-6781. With respect to the specific methodology used in the criticality analysis (specifically, no Xe-135 and the concentration of Np-239 is added to the amount of Pu-239), sensitivity calculations were performed for various initial enrichment and burnup combinations, and for various relevant cooling times. Additional studies were done that included Xe-135 and Np-239. For a subset of these studies, MCNP4a was not able to be used since it did not have the appropriate cross sections. Therefore, MCNP5 was used.

The results are shown in the following three tables and two figures. A comparison between the two code versions is also presented. These results indicate that the modeling assumption of zero hours cooling time without Xe-135 and adding the concentration of Np-239 to Pu-239 is conservative because the zero hours cooling time reactivity is always greater than the reactivity at any cooling time when including Xe-135 and Np-239.

MCNP4a Sensitivity Calculations for Cooling Time With Standard Methodology (No Xenon-135, Np-239 added to Pu-239)

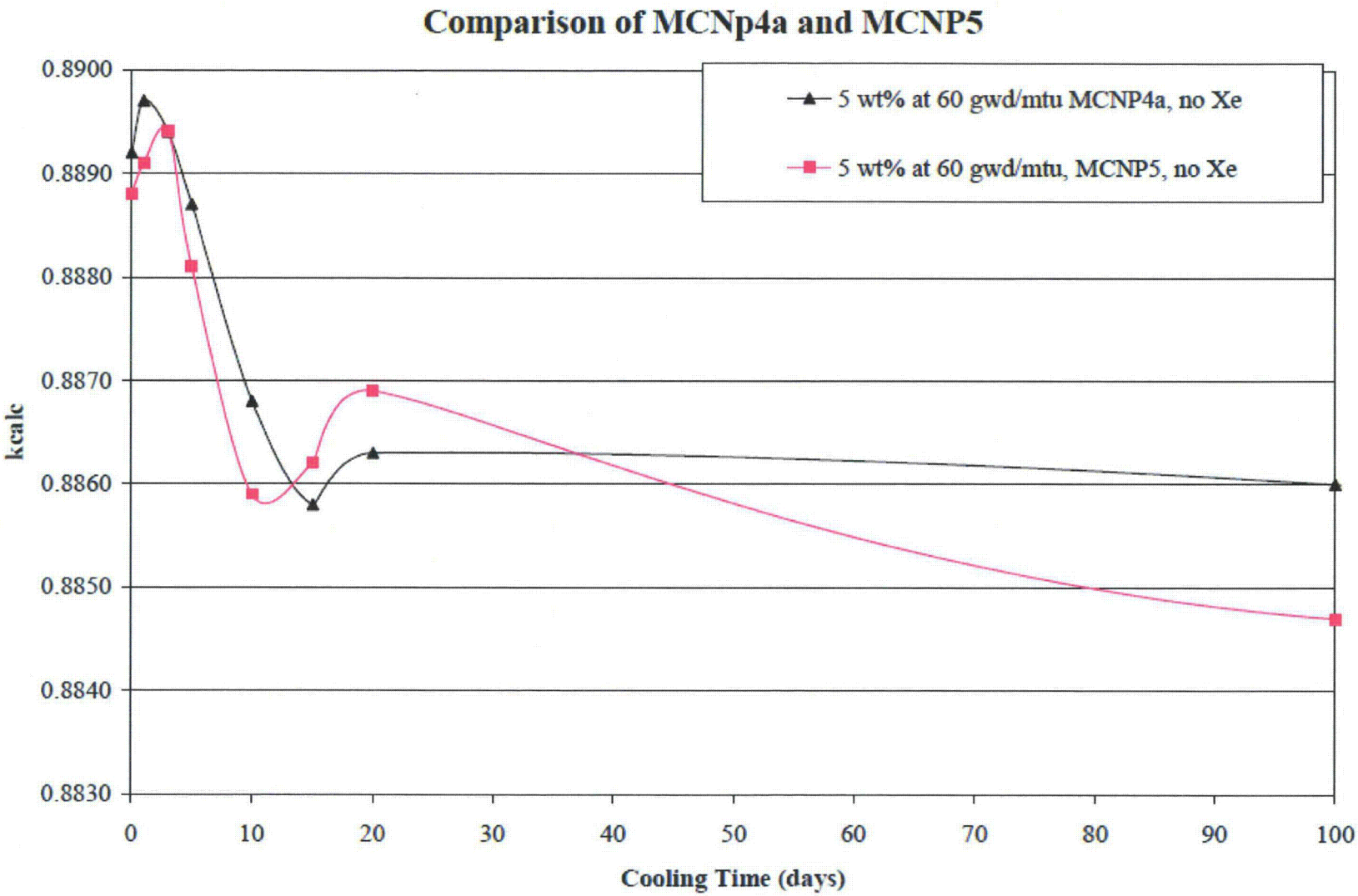
Parameter		0 days		1 day		3 days		5 days		10 days		15 days		20 days		100 days		365 days		20 years	
wt% U 235	gwd/ mtu	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc
2	5	cta205	0.9225	ctb205	0.9204	cti205	0.9206	cte205	0.9192	ctc205	0.9184	ctf205	0.9182	ctg205	0.9182	ctd205	0.9185	cth205	0.9185	ct2205	0.9127
2	20	cta2020	0.8262	ctb2020	0.8241	cti2020	0.8239	cte2020	0.8217	ctc2020	0.8212	ctf2020	0.8218	ctg2020	0.8224	ctd2020	0.8210	cth2020	0.8182	ct22020	0.7887
2	40	cta2040	0.7509	ctb2040	0.7512	cti2040	0.7489	cte2040	0.7491	ctc2040	0.7485	ctf2040	0.7476	ctg2040	0.7489	ctd2040	0.7465	cth2040	0.7431	ct22040	0.6896
2	60	cta2060	0.7097	ctb2060	0.7068	cti2060	0.7089	cte2060	0.7059	ctc2060	0.7066	ctf2060	0.7049	ctg2060	0.7065	ctd2060	0.7037	cth2060	0.6980	ct22060	0.6300
3.5	5	cta355	1.0643	ctb355	1.0632	cti355	1.0611	cte355	1.0618	ctc355	1.0616	ctf355	1.0618	ctg355	1.0609	ctd355	1.0605	cth355	1.0617	ct2355	1.0603
3.5	20	cta3520	0.9585	ctb3520	0.9567	cti3520	0.9549	cte3520	0.9543	ctc3520	0.9551	ctf3520	0.9549	ctg3520	0.9547	ctd3520	0.9546	cth3520	0.9529	ct23520	0.9359
3.5	40	cta3540	0.8678	ctb3540	0.8682	cti3540	0.8649	cte3540	0.8657	ctc3540	0.8658	ctf3540	0.8650	ctg3540	0.8646	ctd3540	0.8636	cth3540	0.8626	ct23540	0.8278
3.5	60	cta3560	0.8036	ctb3560	0.8015	cti3560	0.8014	cte3560	0.8021	ctc3560	0.7997	ctf3560	0.8012	ctg3560	0.7995	ctd3560	0.7991	cth3560	0.7958	ct23560	0.7495
5	5	cta505	1.1459	ctb505	1.1454	cti505	1.1451	cte505	1.1456	ctc505	1.1442	ctf505	1.1444	ctg505	1.1442	ctd505	1.1432	cth505	1.1435	ct2505	1.1439
5	20	cta5020	1.0466	ctb5020	1.0454	cti5020	1.0441	cte5020	1.0434	ctc5020	1.0435	ctf5020	1.0434	ctg5020	1.0436	ctd5020	1.0439	cth5020	1.0437	ct25020	1.0302
5	40	cta5040	0.9564	ctb5040	0.9527	cti5040	0.9557	cte5040	0.9560	ctc5040	0.9528	ctf5040	0.9543	ctg5040	0.9548	ctd5040	0.9525	cth5040	0.9515	ct25040	0.9273
5	60	cta5060	0.8892	ctb5060	0.8897	cti5060	0.8894	cte5060	0.8887	ctc5060	0.8868	ctf5060	0.8858	ctg5060	0.8863	ctd5060	0.8860	cth5060	0.8838	ct25060	0.8490

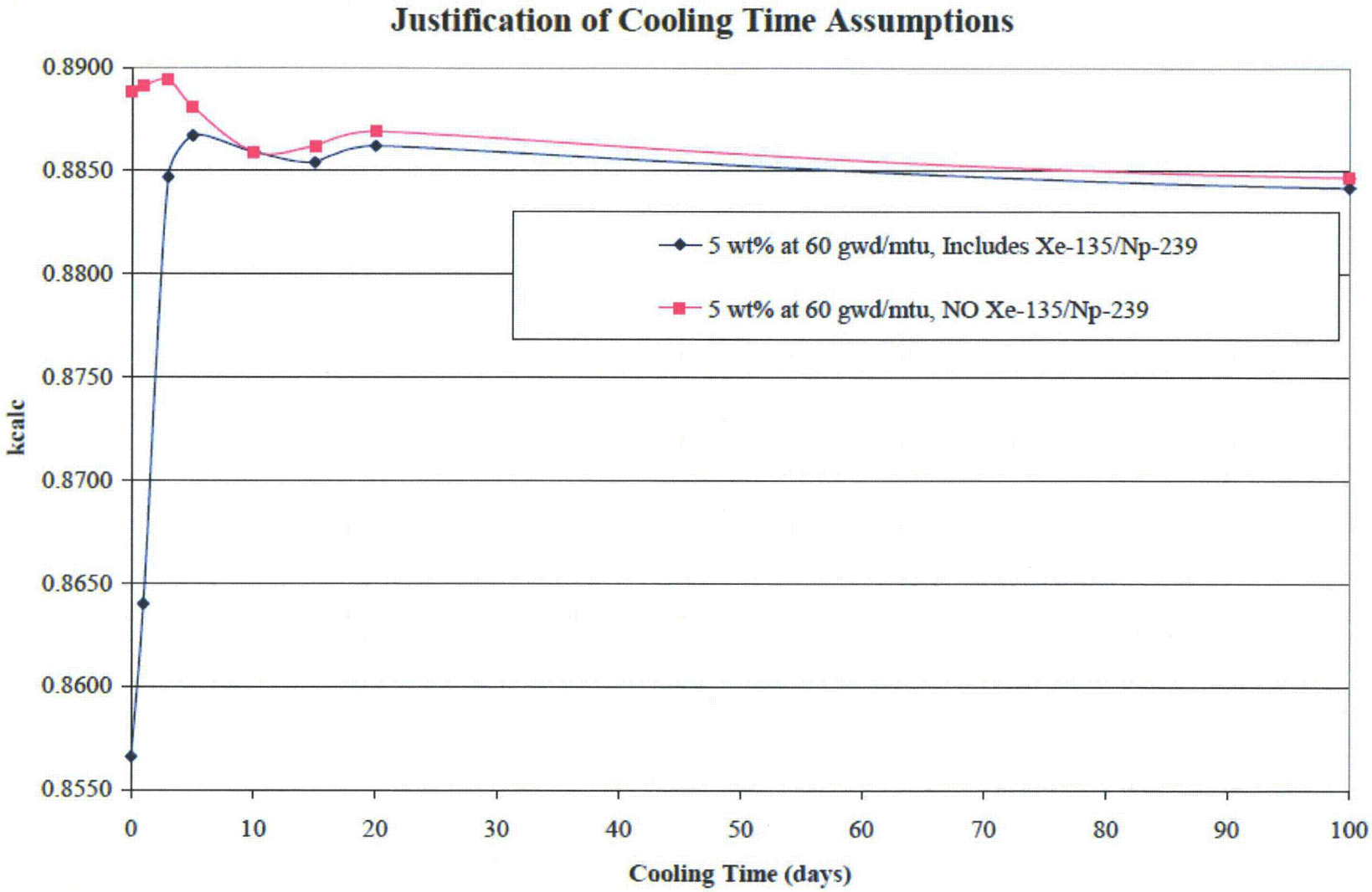
MCNP5 Sensitivity Calculations for Cooling Time With Standard Methodology (No Xenon-135, Np-239 added to Pu-239)

Parameter		0 days		1 day		3 days		5 days		10 days		15 days		20 days		100 days		365 days		20 years	
U-235	gwd/ mtu	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc
2	5	ctj205	0.9218	ctk205	0.9193	ctr205	0.9190	ctn205	0.9190	ctl205	0.9183	cto205	0.9178	ctp205	0.9177	ctm205	0.9181	ctq205	0.9168	ct3205	0.9125
2	20	ctj2020	0.8261	ctk2020	0.8251	ctr2020	0.8233	ctn2020	0.8210	ctl2020	0.8221	cto2020	0.8230	ctp2020	0.8217	ctm2020	0.8201	ctq2020	0.8193	ct32020	0.7881
2	40	ctj2040	0.7513	ctk2040	0.7486	ctr2040	0.7502	ctn2040	0.7500	ctl2040	0.7490	cto2040	0.7472	ctp2040	0.7491	ctm2040	0.7483	ctq2040	0.7427	ct32040	0.6912
2	60	ctj2060	0.7091	ctk2060	0.7080	ctr2060	0.7072	ctn2060	0.7058	ctl2060	0.7057	cto2060	0.7049	ctp2060	0.7058	ctm2060	0.7037	ctq2060	0.6994	ct32060	0.6304
3.5	5	ctj355	1.0643	ctk355	1.0623	ctr355	1.0643	ctn355	1.0609	ctl355	1.0621	cto355	1.0623	ctp355	1.0616	ctm355	1.0611	ctq355	1.0611	ct3355	1.0593
3.5	20	ctj3520	0.9582	ctk3520	0.9568	ctr3520	0.9558	ctn3520	0.9549	ctl3520	0.9546	cto3520	0.9543	ctp3520	0.9533	ctm3520	0.9546	ctq3520	0.9538	ct33520	0.9345
3.5	40	ctj3540	0.8678	ctk3540	0.8671	ctr3540	0.8654	ctn3540	0.8665	ctl3540	0.8643	cto3540	0.8646	ctp3540	0.8638	ctm3540	0.8633	ctq3540	0.8613	ct33540	0.8290
3.5	60	ctj3560	0.8037	ctk3560	0.8042	ctr3560	0.8013	ctn3560	0.7996	ctl3560	0.7996	cto3560	0.7994	ctp3560	0.8000	ctm3560	0.7999	ctq3560	0.7962	ct33560	0.7492
5	5	ctj505	1.1465	ctk505	1.1464	ctr505	1.1449	ctn505	1.1440	ctl505	1.1431	cto505	1.1431	ctp505	1.1434	ctm505	1.1450	ctq505	1.1440	ct3505	1.1445
5	20	ctj5020	1.0454	ctk5020	1.0461	ctr5020	1.0433	ctn5020	1.0438	ctl5020	1.0435	cto5020	1.0435	ctp5020	1.0446	ctm5020	1.0423	ctq5020	1.0433	ct35020	1.0308
5	40	ctj5040	0.9564	ctk5040	0.9556	ctr5040	0.9541	ctn5040	0.9537	ctl5040	0.9544	cto5040	0.9534	ctp5040	0.9544	ctm5040	0.9529	ctq5040	0.9508	ct35040	0.9281
5	60	ctj5060	0.8888	ctk5060	0.8891	ctr5060	0.8894	ctn5060	0.8881	ctl5060	0.8859	cto5060	0.8862	ctp5060	0.8869	ctm5060	0.8847	ctq5060	0.8838	ct35060	0.8485

MCNP5 Sensitivity Calculations for Cooling Time With Xenon-135 and Np-239

Parameter		0 days		1 day		3 days		5 days		10 days		15 days		20 days		100 days		365 days		20 years	
wt% U-235	gwd/ mtu	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc	input file	kcalc
2	5	cts205	0.8872	ctt205	0.8789	ct1205	0.9153	ctw205	0.9183	ctu205	0.9174	ctx205	0.9172	cty205	0.9191	ctv205	0.9173	ctz205	0.9175	ct4205	0.9121
2	20	cts2020	0.7938	ctt2020	0.7880	ct12020	0.8189	ctw2020	0.8212	ctu2020	0.8204	ctx2020	0.8208	cty2020	0.8211	ctv2020	0.8201	ctz2020	0.8166	ct42020	0.7881
2	40	cts2040	0.7228	ctt2040	0.7187	ct12040	0.7455	ctw2040	0.7482	ctu2040	0.7482	ctx2040	0.7475	cty2040	0.7473	ctv2040	0.7465	ctz2040	0.7438	ct42040	0.6925
2	60	cts2060	0.6823	ctt2060	0.6784	ct12060	0.7045	ctw2060	0.7057	ctu2060	0.7063	ctx2060	0.7045	cty2060	0.7052	ctv2060	0.7035	ctz2060	0.6984	ct42060	0.6308
3.5	5	cts355	1.0255	ctt355	1.0300	ct1355	1.0601	ctw355	1.0610	ctu355	1.0620	ctx355	1.0607	cty355	1.0600	ctv355	1.0619	ctz355	1.0607	ct4355	1.0598
3.5	20	cts3520	0.9213	ctt3520	0.9268	ct13520	0.9545	ctw3520	0.9528	ctu3520	0.9534	ctx3520	0.9533	cty3520	0.9536	ctv3520	0.9536	ctz3520	0.9533	ct43520	0.9345
3.5	40	cts3540	0.8355	ctt3540	0.8381	ct13540	0.8630	ctw3540	0.8648	ctu3540	0.8646	ctx3540	0.8658	cty3540	0.8635	ctv3540	0.8645	ctz3540	0.8619	ct43540	0.8286
3.5	60	cts3560	0.7722	ctt3560	0.7751	ct13560	0.7984	ctw3560	0.7991	ctu3560	0.7982	ctx3560	0.7996	cty3560	0.7985	ctv3560	0.7988	ctz3560	0.7960	ct43560	0.7494
5	5	cts505	1.1065	ctt505	1.1191	ct1505	1.1411	ctw505	1.1436	ctu505	1.1429	ctx505	1.1430	cty505	1.1442	ctv505	1.1435	ctz505	1.1423	ct4505	1.1438
5	20	cts5020	1.0094	ctt5020	1.0188	ct15020	1.0418	ctw5020	1.0424	ctu5020	1.0442	ctx5020	1.0427	cty5020	1.0432	ctv5020	1.0425	ctz5020	1.0431	ct45020	1.0297
5	40	cts5040	0.9213	ctt5040	0.9309	ct15040	0.9507	ctw5040	0.9525	ctu5040	0.9506	ctx5040	0.9526	cty5040	0.9543	ctv5040	0.9531	ctz5040	0.9508	ct45040	0.9291
5	60	cts5060	0.8566	ctt5060	0.8640	ct15060	0.8847	ctw5060	0.8867	ctu5060	0.8859	ctx5060	0.8854	cty5060	0.8862	ctv5060	0.8842	ctz5060	0.8826	ct45060	0.8473


Comparison of MCNp4a and MCNP5
5 wt% at 60 gwd/mtu MCNP4a, no Xe
5 wt% at 60 gwd/mtu, MCNP5, no Xe
kcalc
0.8900
0.8890
0.8880
0.8870
0.8860
0.8850
0.8840
0.8830
0
10
20
30
40
50
60
70
80
90
100
Cooling Time (days)


Justification of Cooling Time Assumptions
0.8900
0.8850
0.8800
0.8750
0.8700
0.8650
0.8600
0.8550
kcalc
0
10
20
30
40
50
60
70
80
90
100
Cooling Time (days)
5 wt% at 60 gwd/mtu, Includes Xe-135/Np-239
5 wt% at 60 gwd/mtu, NO Xe-135/Np-239

26. Section 4.7.5 addresses uncertainty in depletion calculations. There is no basis in the Kopp memo to suggest that coverage of changes in fuel geometry during irradiation was intended. Uncertainty in k_{eff} due to random fuel geometry changes should be handled as variations in the parametric ranges of the analysis and included along with the other uncertainties. Variation in k_{eff} due to anticipated geometry changes during irradiation should be handled as a bias.

Confirm that the impacts of fuel geometry changes during irradiation are properly handled.

Response

This request for additional information will be resolved in future correspondence.

27. Generic issue on calculation of uncertainties using MCNP: Where the reactivity worths of an uncertainty are calculated using MCNP, the reactivity worth should also include allowance for the Monte Carlo uncertainty in the calculation of the reactivity worth. Revise the analysis to properly incorporate k_{eff} uncertainties calculated using MCNP.

Response

A future supplement to the LAR will reflect a revision of the criticality analysis such that when two separate, independent MCNP4a calculations are compared to determine a delta k_{calc}, the uncertainty associated with each individual calculation is statistically combined and added to the k_{calc} calculation according to the following equation:

$$\text{Delta } k_{calc} = (k_{calc2} - k_{calc1}) + 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$$

28. Section 4.7.8 discussed temperature and water density effects. Clarify the source of the bias, the method by which the bias is calculated, and the justification for using CASMO to estimate the bias. Confirm that the calculational results shown in Table 4.7.15 reflect changes in both water density and temperature. Discuss how the temperature bias changes with the presence of soluble boron. If appropriate, include a revised temperature bias for soluble boron crediting.

Response

This request for additional information will be resolved in future correspondence.

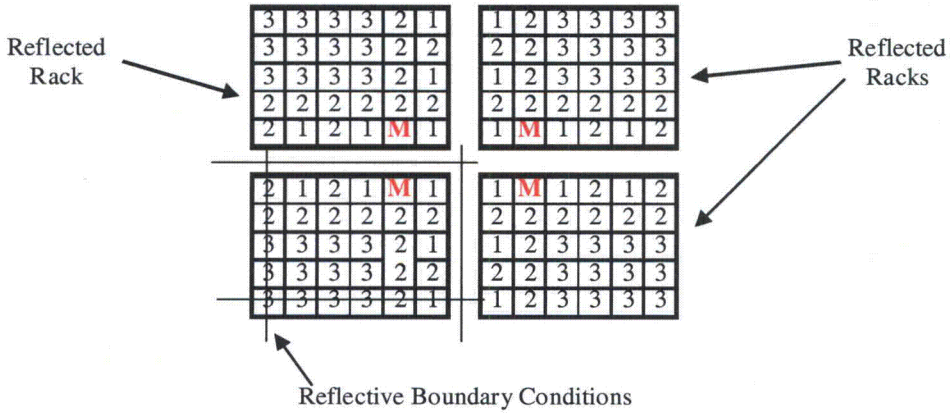
29. Section 4.7.13.4.1 described the analysis of misloaded fresh fuel assemblies. Provide the misloaded fuel assembly locations that were evaluated. If a limited set of positions were evaluated, justify use of the selected locations.

Response

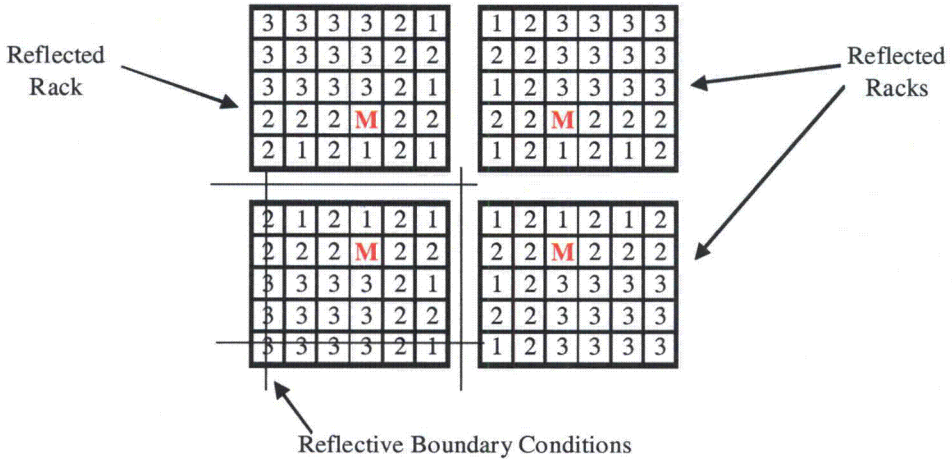
The misloaded fuel assembly cases that were analyzed are described in Table 4.7.17 and the bounding case is shown in Figure 4.7.4. Additional figures are shown below for clarification of the three cases. These cases were selected because they bound the possibilities in the storage rack. For the bounding case, the misloaded 5.0 weight percent U-235 fuel assembly is face adjacent to three other 5.0 weight percent U-235 fuel assemblies. No other location in the storage rack would allow this configuration, and therefore, as the results show, it is bounding. For the other Region 2 location, the position was selected so that the misloaded assembly was face adjacent to the Region 1 cell, again bounding any other second row location. Both of these cases bound any possible Region 3 location, so the third case is arbitrarily chosen.

Additional Figures Showing the Misloaded Cases

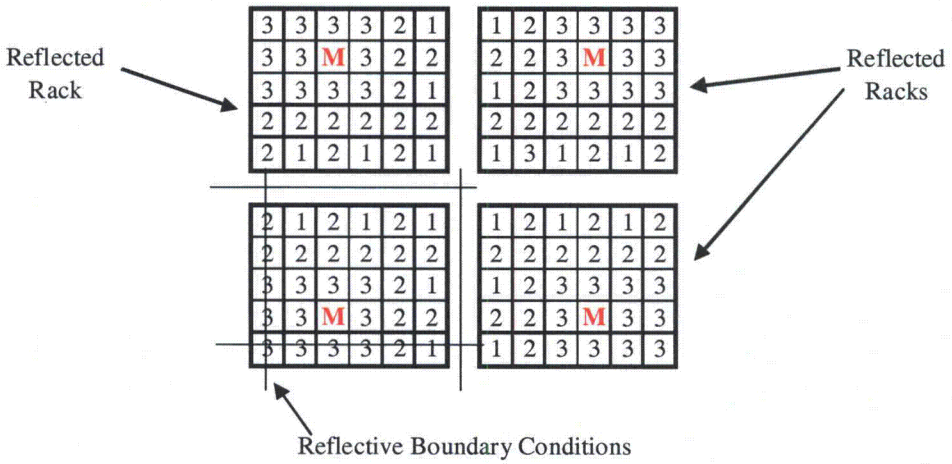
First Configuration, Misloaded in Outer Row of Region 2



Second Configuration, Misloaded in Inner Row of Region 2



Third Configuration, Misloaded in Region 3



30. Section 4.7.13.4.2, second paragraph, first sentence stated that a periodic boundary condition was used. Use of a periodic boundary condition with the model shown in Figure 4.7.3 is not appropriate. Confirm that an appropriate boundary condition was used or provide justification for use of the periodic boundary condition.

From the text in Section 4.7.13.4.2, it is not clear if any variations of the mislocated assembly model were considered. What is the impact on k_{eff} of moving the three assemblies, within their cells, closer to the mislocated assembly? What is the impact of moving the mislocated assembly a cm or two away from the storage racks? Describe what work was done to ensure that the mislocated assembly model used was the most reactive.

Response

With respect to the boundary conditions, periodic boundary conditions were used in the model because of its size. Due to the size of the model, the choice of the boundary conditions is inconsequential. Sensitivity calculations with reflective boundary conditions were performed and presented below. A comparison of these results with those in the LAR Table 4.7.4 shows that the periodic boundary conditions result in a higher soluble boron requirement. However, the criticality analysis will be updated using reflective boundary conditions. A future supplement to the LAR will reflect this update to the analysis.

Calculation of the Mislocated Fresh Fuel Assembly Accident with Reflective Boundary Conditons

Region 2 Enrichment (wt% U-235)	5.0	5.0	5.0
Region 2 Burnup (GWD/MTU)	48	48	48
Region 3 Enrichment (wt% U-235)	2.0	3.5	5.0
Region 3 Burnup (GWD/MTU)	0.00	20.00	42.00
Mislocated Case Input File, 0 ppm	ag2048	ag3548	ag5048
Mislocated Case Input File, 2500 ppm Boron	bag2048	bag3548	bag5048
Mislocated Case Reactivity, 0 ppm	1.0279	1.0280	1.0284
Mislocated Case Reactivity, 2500 ppm	0.8010	0.8021	0.8021
Target k-eff (0.945-corrections)†	0.9221	0.9221	0.9221
Soluble Boron Requirement, ppm	1166	1172	1175

With respect to the location of the mislocated fuel assembly and the placement of the fuel in the storage rack within the model, the effect of moving the assemblies within their cells or small spacing changes of the mislocated fuel assembly would be to change the fuel to moderator ratio, which may or may not increase the reactivity of the model.

However, any increase in reactivity due to small changes in the fuel to moderator ratio is more than compensated for by the following modeling conservatisms: the model only considers half of the poison; and no sheathing is considered along the outer rack edge. Additionally, the mislocated fuel assembly is face adjacent to two fresh fuel assemblies, which is not physically possible in the SFP. This approach therefore more than compensates for any effect due to small fuel to moderator ratio changes and/or boundary condition selection.

31. **Section 4.7.15 and others – The second paragraph of Section 4.7.15 provided the minimum soluble boron concentration required under normal conditions. A value of 472 parts per million (ppm) was obtained by linearly interpolating between 0 and 800 ppm. The uncertainty associated with calculation of the required boron concentration is not provided and, due to the way in which it is calculated, may be significant. The relevant requirement is from 10 CFR 50.68(b)(4).**

    For the required soluble boron concentrations determined in the report, determine the bias and bias uncertainty associated with the method used to determine the required boron concentration. Incorporate this bias and bias uncertainty into the report.

Response

The criticality analysis will be revised to remove the uncertainty associated with linear interpolation over large data points by performing an additional calculation at the interpolated soluble boron concentration. A future supplement to the LAR will reflect this revised analysis.

32. **Table 4.7.7 in the rows for "manufacturing tolerances uncertainty" and "fuel tolerances uncertainty" – Each row cites footnote, which says "These tolerance uncertainties are the maximum from all burnups and enrichments." In Table 4.7.6, this was implemented by listing the largest value for all columns. For example, 0.0037 was listed for "manufacturing tolerances uncertainty" for all enrichments and 0.0074 was listed for "fuel tolerances uncertainty" for all enrichments. Table 4.7.7 shows a different value for each enrichment. Confirm that the rest of Table 4.7.7 used the correct uncertainty values.**

Response

The "manufacturing tolerances uncertainty" and the "fuel tolerances uncertainty" values listed in Table 4.7.7 are typographical errors; however, the correct values were used in the k_{eff} calculations. A future supplement to the LAR will update Table 4.7.7 with the correct values.

33. It appears that the purpose of Tables 4.7.8 and 4.7.10 is to produce an estimate for the bias introduced by not modeling the IFBA and the IFBAs with WDRs during the depletion calculations. Since the storage racks do not take credit for the presence of IFBA, it would seem that this table should show only the impact of the presence of IFBA during the depletion calculations. All of the results should have been restarts in the rack geometry with IFBA B-10 and WDRs removed. Consequently all "Ref." values and associated k_∞ values at zero burnup should have been the same.

It also appears that the columns for IFBA with four and eight WDRs includes the IFBA B-10. This creates large negative Δk values in the last column. So this approach uses the IFBA B-10 to reduce the Δk values, thus indirectly taking credit for IFBA B-10. This is probably not appropriate without further justification. Justify crediting the residual IFBA B-10 in the calculation of the depletion bias.

The calculations were done only with fuel having initial enrichment of 3.4 wt%. Justify not performing the calculations for the range of initial enrichments and IFBA loadings permitted.

Response

This request for additional information will be resolved in future correspondence.

34. Table 4.7.11 has a problem that is similar to that observed in Table 4.7.10. The k_∞ values presented from 0 to 15 GWD/MTU appear to include the worth of the WABA rods. In Table 4.7.11, it is clearly not appropriate to include the effects of the presence of the WABA rods in the bias calculations. Table 4.7.11 should have shown the impact of depletion with WABAs present, but the k_∞ values presented should have been from restarts in rack geometry with WABAs removed.

Revise the analysis to correctly calculate the bias associated with depleting fuel with WABA present or justify not doing so. Secondly, justify not performing the calculations for the full range of initial enrichments and WABA loadings permitted.

Response

This request for additional information will be resolved in future correspondence.

35. Table 4.7.16 includes seven subsections, each of which is for one Region 3 enrichment/burnup combination. It appears that the burnup-dependent "total corrections" for Region 3 were used for every subsection of this table. For example, the first subsection was for Region 3 with 2 wt% with zero burnup and should have used a Region 3 "total corrections" value of 0.0194 for all columns in the subsection. The target k_{eff} for interpolation in the last column in the first subsection should have been calculated as 0.945 – maximum (0.0229, 0.0194) = 0.9221, interpolating to this value gives a "Soluble Boron Requirement" of 427.9 ppm. Instead, it appears that the Region 3 "total corrections" value used in the last column was 0.0238, the value for 5 wt% with 45 GWD/MTU burnup. Consequently, of the 49 "Soluble Boron Requirement" values presented in Table 4.7.16, 35 values appear to be incorrect.

Review the calculations used to prepare Table 4.7.16 and correct the table and associated text as needed. Note that some required soluble boron concentrations are repeated in other places in the report.

Response

The table does contain the mentioned lookup function error; however, the error resulted in conservative results. A future supplement to the LAR will update this information.

36. In Tables 4.7.17 and 4.7.18, it appears that the wrong "Target k_{eff}" value was used. According to the footnote on the bottom of the table, the largest total correction value for Region 2 and 3 at any burnup and enrichment should have been used as the "corrections". This value is 0.0238 from Table 4.7.7. So the target value should have been 0.945 – 0.0238 = 0.9212 not 0.9221. This caused each of the interpolated boron concentrations to be incorrect. Further, interpolating between 0 and 2500 ppm is probably not good. Linear interpolation over this large range may result in a significant underestimate of the soluble boron requirement. Review the calculations used to prepare Tables 4.7.17 and 4.7.18 and correct the tables and associated text as needed. Note that some required soluble boron concentrations are repeated in other places in the report.

Response

The footnote states that the target k_{eff} is, "Maximum of total corrections for the Region 2 and Region 3 enrichment and burnup combination." For clarification, this is the Region 2 and Region 3 enrichment and burnup combination of each specific case in Table 4.7.17 and 4.7.18 and therefore what is presented is correct. A future supplement to the LAR will edit the footnote for additional clarification.

With respect to the soluble boron calculations, please see the response to NRC RAI 31.

37. Table 4.7.21 included the maximum "soluble boron requirement" from Table 4.7.16. In response to comment No. 35 on Table 4.7.16, this value may be revised. If it is, revise Table 4.7.21 accordingly.

Response

A future supplement to the LAR will revise Table 4.7.21 to be consistent with Table 4.7.16.

38. Either in Section 4 or in Appendix A to Section 4, the following validation issues should have been addressed:

a. State the ranges of parameters that the safety analysis fits within. For example, the minimum and maximum values for ^{235}U enrichment, EALF, soluble boron concentrations, Pu content and composition, etc.

b. Describe other features present in the safety analysis models that require validation.

c. State the ranges of parameters covered by the critical experiments used in the validation study.

d. Discuss the applicability of the validation study to the safety analysis models.

e. Discuss gaps within the parametric range covered by the validation and, if necessary, additional margin adopted to cover interpolation.

f. Discuss extrapolation beyond the parameter range covered by the validation study and, if necessary, additional margin adopted to cover extrapolation.

g. Discuss validation gaps (e.g., fission product validation) and, if appropriate, additional margin adopted to cover validation gaps.

Response

A future supplement to the LAR will update the MCNP4a validation report (Appendix A in the LAR) to reflect issues a. through g. The update will include the Haut Taux de Combustion (HTC) experiments conducted by the Institut de Radioprotection et de Sûreté Nucléaire.

39. Section 4A.1, Eq. 4A.2 - As is accurately described in the text, this equation yields the *standard deviation* of *the mean.* The requirements in 10 CFR 50.68 require that we have a 95% probability with a 95% confidence level that a calculation that calculates as being subcritical actually is subcritical. The 95/95 requirement is not on the mean, but rather on a single calculation. Thus, what is needed in Section 4 is the *variance* of *the population about the mean,* not the *standard deviation* of *the mean.* The difference is an extra factor of "n" in the denominator of Eq. 4A.2. Again, it is not that Eq. 4A.2 is incorrect. Instead, the population variance about the mean needs to be calculated in Appendix A and used in Section 4.

Revise Appendix A to also provide the *variance* of *the population about the mean.* Revise Section 4 to correctly incorporate this variance.

Response

A future supplement to the LAR will update the MCNP4a validation report to provide the variance of the population about the mean, and the criticality analysis will reflect this change.

40. Sections 4A.2 and 4A.3 include discussion of comparisons between MCNP and KENO results. From Section 4A.2:

"Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias."

These two methods are not independent. They both use data derived from the same cross section measurements and, in part, from the same ENDF/B-V evaluation. Consequently, they include the same nuclear data measurement errors and evaluation errors and would be expected to respond similarly to these errors. The comparisons provided serve only to confirm that both codes respond to the same data, erroneous or not, in the same way.

The conclusions in Section 4A.2 and 4A.3 on enrichment and B-10 biases based on the KENO/MCNP comparison should not be given any credit. Revise the text in Sections 4A.2 and 4A.3 to remove use of code-to-code comparisons.

Response

A future supplement to the LAR will reflect removal of KENO from the validation report.

41. Section 4A.4.1 includes the following statement:

"The tendency toward over-prediction at close spacing means that the rack calculations may be slightly more conservative than otherwise."

This also means the critical experiment calculations are too high, which is non-conservative. The rack calculations are not "more conservative" because the computational bias also includes this trend. Revise the text to remove claims that the tendency toward over-prediction at close spacing is conservative.

Response

A future supplement to the LAR will update the MCNP4a validation report to remove these claims.

42. The analysis described in the licensee's response to the original RAI was based on "accurate measurements of critical reactor parameters" at 99 points from five cycles of BVPS-1 and 2. Provide additional details for these 99 points so that NRC staff may reach a conclusion as to the appropriateness of using this data to assess the fuel depletion uncertainty. Include as much of the following information as possible, such as: unit, cycle number, cycle burnup, % of full power, measured and predicted soluble boron concentration, control rod bank positions, % axial offset (i.e., (PT – PB)/(PT + PB)*100%; where PT is the power in the top half of the core and PB is the power in the bottom half of the core).

Response

FENOC requests that the NRC no longer consider the supplemental reactor critical data that was provided in the letter dated June 15, 2009, in support of the quantification of fuel depletion uncertainties. Therefore, this request for additional information need not be addressed.

43. The analysis relies on results from core follow calculations that were performed using CASMO-4 cross-section data. Describe the core follow calculations (e.g., codes used, input data, calibration with core measurements, etc.) and the cross-section data (e.g., 2-group macroscopic, homogenized node average, etc.) used.

Response

FENOC requests that the NRC no longer consider the supplemental reactor critical data that was provided in the letter dated June 15, 2009, in support of the quantification of

fuel depletion uncertainties. Therefore, this request for additional information need not be addressed.

44. Two methods were used to estimate the depletion uncertainty. The first method uses the difference between the highest and lowest eigenvalues to estimate uncertainty. The second method apparently uses population variance around a linear least-squares fit of calculated eigenvalues to estimate uncertainty. Neither method appears to include consideration of systematic deviation from expected values that might indicate a bias. The 5% of the reactivity decrement suggested by the Kopp memo was intended to cover both bias and bias uncertainty in fuel composition and k_{eff} calculations. Describe how the two methods presented include consideration of burned fuel composition and k_{eff} calculation biases.

Response

FENOC requests that the NRC no longer consider the supplemental reactor critical data that was provided in the letter dated June 15, 2009, in support of the quantification of fuel depletion uncertainties. Therefore, this request for additional information need not be addressed.

45. It is likely that the bulk of the 99 sets of measured critical reactor parameters are at hot-full power (HFP) conditions. At HFP, the eigenvalue is most sensitive to the fuel compositions away from the core periphery. For highly burned fuel in storage racks, the eigenvalue is most sensitive to the fuel compositions in the top 24 inches of the fuel assembly. Due to axial flux redistribution, it is not clear how the uncertainty in k_{eff} *for* a reactor at HFP is related to the uncertainty in k_{eff} *for* highly-burned fuel in spent fuel storage racks. Additionally, the reactor will include a mixture fresh, once-, twice- and, sometimes thrice-burned fuel. Since fuel is generally reused until it is highly burned, the uncertainty in k_{eff} *for* the highly-burned fuel in fuel storage racks could be significantly higher than the uncertainty for the mixture of fuel in the core. Provide justification for application of uncertainty in reactor core eigenvalues to highly-burned fuel in spent fuel storage racks.

Response

FENOC requests that the NRC no longer consider the supplemental reactor critical data that was provided in the letter dated June 15, 2009, in support of the quantification of fuel depletion uncertainties. Therefore, this request for additional information need not be addressed.

Attachment 2
L-10-082

Regulatory Commitment List
Page 1 of 1

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station Unit No. 2 in this document. Any other actions discussed in the submittal represent intended or planned actions, are described only for information, and are not Regulatory Commitments. Please notify Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 761-6071 of any questions regarding this document or associated Regulatory Commitments.

Regulatory Commitment	Due Date
1. A process will be established prior to receipt of the next reload batch of BVPS Unit 2 fuel to ensure that the design features and operating parameters of fuel used in the future at BVPS Unit 2 are consistent with the assumptions of the criticality analysis.	Prior to receipt of the next reload batch of BVPS Unit 2 fuel.