Dominion Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, Virginia 23060
Web Address: www.dom.com

Dominion®

March 5, 2009

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Serial No.:	09-084
NLOS/WDC	R0
Docket No.:	50-423
License No.:	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING A SPENT FUEL POOL CRITICALITY LICENSE AMENDMENT REQUEST

Dominion Nuclear Connecticut, Inc. (DNC) submitted a stretch power uprate (SPU) license amendment request (LAR) for Millstone Power Station Unit 3 (MPS3) in letters dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A). The SPU LAR included a revised spent fuel pool (SFP) criticality analysis with proposed changes in technical specification (TS) requirements. DNC separated the MPS3 SFP TS change request from the MPS3 SPU request via letter dated March 5, 2008 (Serial No. 07-0450D).

In a letter dated August 8, 2008, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the SFP TS. DNC responded to RAI questions 1 through 19 in a letter dated September 30, 2008 (Serial No. 08-0511A). Subsequently, in a letter dated February 2, 2009, the NRC requested additional information be submitted by March 6, 2009. The responses to RAI questions 20, 22, 23, and 25 are provided in the attachment to this letter.

In a February 25, 2009 telecon between Mr. W. Bartron of DNC and Mr. H. Chernoff of the NRC, it was agreed the responses to RAI questions 21 and 24 would be submitted by March 24, 2009.

Attachment 1 contains the responses to RAI questions 20, 22, 23, and 25, provided by Westinghouse Electric Company, LLC.

The information provided by this letter does not affect the conclusions of the significant hazards consideration discussion in the December 13, 2007 DNC letter (Serial No. 07-0450C).

Should you have any questions in regard to this submittal, please contact Mrs. Wanda Craft at 804-273-4687.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

Commitments made in this letter:
1. Provide the responses to RAI questions 21 and 24 by March 24, 2009.

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President - Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 5th day of March, 2009.

My Commission Expires: 4/30/13.

Notary Public

GINGER L. ALLGOOD
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

Attachment:

1. Attachment 1: NEU-09-11, Attachment 1 Westinghouse Response to Request for Additional Information (RAI) Regarding the Spent Fuel Pool Criticality Amendment Request

cc: U.S. Nuclear Regulatory Commission
Region I
Regional Administrator
475 Allendale Road
King of Prussia, PA 19406-1415

Ms. C. J. Sanders
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop O8-B3
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Millstone Power Station

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

ATTACHMENT 1

RESPONSE TO RAI QUESTIONS 20, 22, 23, and 25

FOR THE SPENT FUEL POOL LICENSE AMENDMENT REQUEST

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

NEU-09-11, Attachment 1 Westinghouse Response to Request for Additional Information (RAI) Regarding the Millstone 3 Spent Fuel Pool Criticality Amendment Request

Question 20

In the response to RAI 1 regarding the axial burnup distribution modeling, it is stated that no racks were modeled. High density water was modeled in the space for the racks, grid spacers, etc. Demonstrate that this is a conservative modeling parameter for all fuel burnup levels that could be stored in the spent fuel pool.

Response:

The response to RAI 1 regarding axial burnup modeling used results from a study intended to be generic in applicability by neglecting fuel storage racks. The study would therefore not be influenced by the design of any particular rack design or material. A new, more complete and Millstone Unit 3 specific analysis has been completed to address this question. This study was performed using PARAGON and SCALE 5.1 and included the Region II and III rack models as described in Section 4.1 of WCAP-16721. These models include the stainless steel present in the storage cell, and fixed poison wrapper. The BORAL material is also modeled for the Region II racks. Justification for the use of PARAGON and SCALE 5.1 is provided in the response to RAI 23.

Limiting axial burnup profiles have been identified as discussed in the response to RAI 3 for Millstone Unit 3 spent fuel. A single most limiting profile has been identified for each of the following categories: fuel assemblies with no blankets (unblanketed), fuel assemblies with natural enrichment blankets, and fuel assemblies with 2.6 w/o ^{235}U blankets. Each of these three profiles was collapsed to 4-zone and 7-zone distributions for depletion calculations. The depletion calculations were performed using the power, temperature, soluble boron, and fuel density assumptions documented in WCAP-16721. Calculations were performed for fuel assemblies with axial blankets in both Region II and Region III with non-blanket region enrichments of 3, 4, and 5 w/o ^{235}U for burnups ranging from 10 to 60 GWd/MTU. Unblanketed fuel assemblies were only considered at 3 w/o ^{235}U for burnups ranging from 10 to 50 GWd/MTU. This selection was made because the highest enrichment of unblanketed assemblies used at Millstone Unit 3 is 2.9 w/o and the discharge burnups for the unblanketed fuel are bounded by 50 GWd/MTU.

The tables given below present the results of these calculations. The reactivity difference presented is the k_{eff} from the 7-zone model subtracted from the k_{eff} of the 4-zone model. A positive reactivity difference therefore indicates the 4-zone model to be conservative relative to the 7-zone model. As can be seen in the tables, the 4-zone model is conservative for all conditions analyzed. In most cases the reactivity difference is statistically significant at a 2-sigma level. The data presented in the tables represent the entire range of enrichment and burnup for fuel assemblies that are or might be stored in the Millstone Unit 3 spent fuel pool. These results demonstrate that the presence of storage rack structure does not change the conclusion that the 4-zone model is conservative relative to the 7-zone model for axial burnup shapes relevant to Millstone Unit 3.

In addition, since the stainless steel rack structure did not affect the conclusion that the 4-zone model is conservative relative to the 7-zone model for axial burnup shapes, it is reasonable to conclude that this also would be true for grid spacers. This is because the stainless steel rack structure is a much stronger neutron absorber than the zircalloy spacers, and the volume of stainless steel in the rack structure is significantly larger than the volume of zircalloy spacer material.

Reactivity Difference Between 4-zone and 7-zone Models in Region II Fuel Assemblies with No Blankets

Assembly Average Burnup (MWd/MTU)	3 w/o ^{235}U Fuel	
	Δk_{eff}	RSS 1-σ Uncertainties
10,000	0.00360	0.00048
15,000	0.00253	0.00047
20,000	0.00262	0.00047
25,000	0.00222	0.00050
30,000	0.00208	0.00047
40,000	0.00098	0.00046
50,000	0.00040	0.00045

Reactivity Difference Between 4-zone and 7-zone Models in Region III Fuel Assemblies with No Blankets

Assembly Average Burnup (MWd/MTU)	3 w/o ^{235}U Fuel	
	Δk_{eff}	RSS 1-σ Uncertainties
10,000	0.00352	0.00041
15,000	0.00274	0.00042
20,000	0.00233	0.00050
25,000	0.00151	0.00045
30,000	0.00129	0.00046
40,000	0.00184	0.00051
50,000	0.00098	0.00040

Reactivity Difference Between 4-zone and 7-zone Models in Region II Fuel Assemblies with Natural Enrichment Blankets

Burnup (MWd/MTU)	3 w/o ^{235}U Fuel		4 w/o ^{235}U Fuel		5 w/o ^{235}U Fuel	
	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ
10,000	0.00648	0.00046	0.00585	0.00050	0.00581	0.00048
15,000	0.00763	0.00043	0.00804	0.00049	0.00828	0.00047
20,000	0.00688	0.00042	0.00884	0.00045	0.00902	0.00047
25,000	0.00605	0.00045	0.00786	0.00045	0.00903	0.00047
30,000	0.00381	0.00045	0.00608	0.00046	0.00894	0.00047
40,000	0.00343	0.00043	0.00421	0.00045	0.00634	0.00048
50,000	0.00207	0.00042	0.00297	0.00042	0.00439	0.00048
60,000	0.00077	0.00040	0.00090	0.00045	0.00265	0.00049

Reactivity Difference Between 4-zone and 7-zone Models in Region III Fuel Assemblies with Natural Enrichment Blankets

Burnup	3 w/o ^{235}U Fuel		4 w/o ^{235}U Fuel		5 w/o ^{235}U Fuel	
(MWd/MTU)	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ
10,000	0.00536	0.00042	0.00594	0.00045	0.00618	0.00045
15,000	0.00739	0.00041	0.00849	0.00044	0.00706	0.00046
20,000	0.00772	0.00041	0.00830	0.00044	0.00899	0.00042
25,000	0.00654	0.00042	0.00855	0.00043	0.00961	0.00045
30,000	0.00465	0.00045	0.00744	0.00053	0.00866	0.00044
40,000	0.00426	0.00042	0.00470	0.00048	0.00551	0.00048
50,000	0.00226	0.00041	0.00288	0.00042	0.00451	0.00052
60,000	0.00064	0.00040	0.00205	0.00044	0.00277	0.00043

Reactivity Difference Between 4-zone and 7-zone Models in Region II Fuel Assemblies with 2.6 w/o Enrichment Blankets

Burnup	3 w/o ^{235}U Fuel		4 w/o ^{235}U Fuel		5 w/o ^{235}U Fuel	
(MWd/MTU)	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ
10,000	0.00459	0.00047	0.00458	0.00048	0.00365	0.00050
15,000	0.00474	0.00046	0.00468	0.00049	0.00533	0.00049
20,000	0.00231	0.00049	0.00574	0.00045	0.00646	0.00046
25,000	0.00249	0.00047	0.00348	0.00045	0.00683	0.00044
30,000	0.00208	0.00046	0.00366	0.00049	0.00604	0.00045
40,000	0.00102	0.00043	0.00294	0.00045	0.00489	0.00047
50,000	0.00064	0.00041	0.00118	0.00047	0.00240	0.00047
60,000	0.00092	0.00045	0.00129	0.00047	0.00262	0.00045

Reactivity Difference Between 4-zone and 7-zone Models in Region III Fuel Assemblies with 2.6 w/o Enrichment Blankets

Burnup	3 w/o ^{235}U Fuel		4 w/o ^{235}U Fuel		5 w/o ^{235}U Fuel	
(MWd/MTU)	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ	Δk_{eff}	RSS 1-σ
10,000	0.00379	0.00045	0.00304	0.00043	0.00379	0.00047
15,000	0.00413	0.00041	0.00580	0.00042	0.00499	0.00044
20,000	0.00366	0.00043	0.00539	0.00042	0.00640	0.00042
25,000	0.00247	0.00043	0.00474	0.00045	0.00591	0.00044
30,000	0.00157	0.00044	0.00377	0.00045	0.00612	0.00045
40,000	0.00043	0.00042	0.00288	0.00047	0.00379	0.00043
50,000	0.00136	0.00040	0.00156	0.00041	0.00232	0.00051
60,000	0.00030	0.00040	0.00057	0.00045	0.00187	0.00042

Question 22

In the response to RAI 1 regarding the axial burnup distribution modeling, it is stated that neither the four or seven node case is more reactive. MPS3, therefore, concludes both models are sufficient. Additionally, in the comparison on page 3 of the RAI response, only two high burn-up levels were used and the Δk_{eff} results show a 0.00171 difference. Given the reserved analytical margin is 0.001, justify why this difference is acceptable. Provide additional justification for a four-node model, or perform your analysis with a more conservative model (e.g., more zones).

Response:

A more complete study comparing the reactivity of 4-zone and 7-zone burnup profiles was presented in the response to RAI 20. This study includes an enrichment range of 3 – 5 w/o ^{235}U and a burnup range of 10 to at least 50 GWd/MTU, and considers all three fuel assembly axial designs present in the spent fuel pool at Millstone Unit 3. As discussed in the response to RAI 20, the results of this study demonstrate that the 4-zone model is a conservative representation relative to the 7-zone model.

Question 23

In the response to RAIs 5 and 6, it is stated that the PARAGON code and SCALE Version 5.1 were used for computational efficiency, rather than PHOENIX-P and SCALE Version 4.4. Provide a justification for the use of these codes based on technical sufficiency.

Response:

The analysis reported in WCAP-16721 uses the PHOENIX-P lattice code for generated depleted isotopic number densities and SCALE Version 4.4 for three-dimensional Monte Carlo calculations to determine neutron multiplication factors (k_{eff}) in the spent fuel pool environment. SCALE Version 4.4 uses the BONAMI and NITAWL modules for cross section processing and the KENO V.a module for transport calculations. Reference 1 explicitly allows the use of NITAWL-KENO V.a and PHOENIX-P for spent fuel pool criticality safety calculations. Further justification for the use of these codes is provided here.

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 Version 4.4 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 approved in Reference 1 are used in both versions of SCALE, so it is concluded that SCALE version 5.1 is an appropriate code for use in these calculations.

The PARAGON lattice code was approved for use in reactor physics calculations by the NRC in Reference 2. PARAGON is intended for use as a stand alone 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". It is therefore concluded that PARAGON is an acceptable code for the generation of depleted isotopic number densities.

Question 25

Section 4.5.2 of WCAP 16721-P, "Millstone Unit 3 Spent Fuel Pool Criticality Analysis," dated March 2007, states that three scenarios were analyzed in detail for the spent fuel assembly mishandling and dropped assembly events. Demonstrate the three scenarios analyzed bounds all mishandling and dropped assembly events. Additionally, the MPS3 Final Safety Analysis Report states that "SFP bulk water temperature corresponding to the maximum reactivity in the normal operating temperature range is used in the criticality analysis. The normal operating bulk water temperature range used in the criticality analysis is 32 °F to 160 °F, which bounds the actual normal operating temperature range." In WCAP 16721, it is stated that 68 of was used for the criticality analysis and that is the most reactive temperature for Regions I and II. Demonstrate this is the most reactive temperature for Regions I and II.

Response:

A wide range of postulated fuel assembly drop and misload scenarios were analyzed with a preliminary spent fuel pool model in support of the analysis reported in WCAP-16721. The results from this preliminary spent fuel pool model were used to screen which scenarios produced the limiting reactivity insertion events. The three scenarios selected from the preliminary model results were then further analyzed with the final spent fuel pool models, and these are the three scenarios listed in WCAP-16721-P Section 4.5.2 and Table 4-12.

Preliminary Spent Fuel Pool Model Analysis

This preliminary spent fuel pool model differs from the final model due to slight differences in the allowable fuel enrichments modeled in the spent fuel pool racks, and also due to different spacing between Region II and III of the spent fuel pool. The small fuel enrichment differences between the preliminary and final models will not impact the relative reactivity insertions of the postulated misload/dropped assembly events, so the preliminary screening conclusions remain valid. Also, the actual minimum spacing between Region II and III racks, as used in the final models and documented in WCAP-16721-P (Table 3-2), is larger than that used in the preliminary screening model. Thus the preliminary screening model will produce larger (more conservative) reactivity insertions, and the preliminary screening conclusions remain valid.

Fuel Misloading Scenarios

As shown in the attached Table 1, 12 fuel misloading scenarios were analyzed using the preliminary models. The k_{eff} change for each scenario is relative to the k_{eff} of a base case for each region (base case k_{eff} not shown). A single 5 w/o U-235 enriched fresh fuel assembly, which is the maximum allowed fuel assembly reactivity, was postulated to be misloaded in various locations of the Region I, II and III racks.

As the attached Table 1 shows, the maximum reactivity insertion for any of the 12 misloading events is the "Misloaded fuel assembly in Center of Region III" scenario, which is substantially larger than the results of the other cases shown. Region III is comprised of racks with no credited fixed neutron poison, so the reactivity insertion is greater than in Regions I or II. The scenarios involving center locations of the rack also eliminates any mitigation from radial neutron leakage. In fact, it can be seen that for all three regions, the maximum reactivity misload occurs in the center of each Region.

Only a single fuel assembly misload is postulated, since the double contingency principle states that the analysis need not consider two unlikely independent concurrent events to ensure protection against a criticality accident.

Thus, the fuel misloading scenario for a "Misloaded fuel assembly in Center of Region III " will be further analyzed, and is one of the three scenarios discussed in WCAP-16721 Section 4.5.2.

Dropped Fuel Assembly Scenarios

Only 1 fuel assembly may be moved at a time in the spent fuel pool. If a fuel assembly is being moved, and a fuel drop event occurs, there are three possible outcomes for the fuel assembly that is dropped:
(1) it can fall and come to rest on top of the fuel storage racks
(2) it can fall directly into an empty fuel storage cell, or
(3) it can fall between storage Regions

Note that the fuel assembly cannot drop between fuel storage racks within a given storage Region, because the spacing is too small between racks of the same Region to allow it.

(1) For the case of a fuel assembly dropping on top of the spent fuel pool storage racks, the physical separation between the fuel assemblies seated in the spent fuel pool storage racks and the assembly lying on top of the racks is sufficient to neutronically decouple the dropped fuel assembly from the stored fuel assemblies. In other words, dropping the fuel assembly on top of the storage racks does not produce a positive reactivity insertion.

(2) If the fuel assembly falls directly into an empty storage location, this is treated from a reactivity viewpoint in the same way as if it was a misloaded fuel assembly. The reactivity effects as previously discussed in the fuel misloading discussion apply, and no additional analysis is necessary.

(3) For the case of a dropped fuel assembly falling in between fuel storage Regions, five cases are analyzed as shown in the attached Table 1. Refer to WCAP-16721 Figure 3-1 for an approximate representation of where Regions I, II and III are located relative to each other.

- Should a dropped fuel assembly occur between Regions I and II, two possible limiting cases are analyzed with a full spent fuel pool model. One is a case where the space between Regions I and II is just large enough to allow the dropped fuel assembly to fit between the two regions. The second case analyzed is a dropped fuel assembly that fits directly into a corner where Region I and II racks meet each other, with the dropped fuel assembly pushing against both the Region I and II racks. The case of a dropped fuel assembly between Regions I and II is more limiting since it produces a larger k_{eff} change, and this scenario will receive further analysis.
- Should a dropped fuel assembly occur between Region II and III, two possible limiting cases are analyzed with a full spent fuel pool model. One is a case where the dropped assembly is between the Region II and III racks. The second case analyzed is a dropped fuel assembly that fits directly into a corner where Region II and III racks meet each other, with the dropped fuel assembly pushing against both the Region II and III racks. The case of a dropped fuel assembly between Regions II

and III is more limiting, since it produces a larger k_{eff} change, and this scenario will receive further analysis.

- Should a dropped fuel assembly occur between Region I and III, only one case was evaluated. Only one case was evaluated because Region 1 and Region III have a large gap between them, with no common corners between the two regions. As noted in WCAP-16721 Table 3-1, there is more than a 12 inch gap between these 2 regions. Given the large gap between these two regions, and the resulting k_{eff} of this case being less than the dropped fuel assembly between Regions II and III, it was not necessary to carry this scenario for any further analysis.

Final Spent Fuel Pool Analysis

The three limiting dropped fuel assembly and misloaded fuel assembly scenarios discussed above were reanalyzed with the final spent fuel pool model. The three scenarios are:

- Misloaded fuel assembly in Center of Region III
- Dropped fuel assembly Between Regions I and II
- Dropped fuel assembly Between Regions II and III

This final spent fuel pool model differs from the preliminary model only slightly in that the enrichments modeled in the spent fuel pool are the final enrichments for each region, and the actual minimum space between Regions II and III of the spent fuel pool is considered. This preliminary screening model also did not account for the actual minimum spacing between Region II and III racks. The actual minimum spacing (WCAP-16721 Table 3-2) between Region II and III racks used in the final models, is larger than that used in the preliminary screening model and thus the preliminary screening model will produce larger reactivity insertions.

The resulting reactivity values and required boron concentrations for the three limiting cases, as determined by the final spent fuel pool models, are shown in Table 2, which is directly from WCAP-16721 Table 4-12. As can be seen, the k_{eff} change values for each of the three cases in the final analysis did change from the preliminary analysis, due to the enrichment changes in all regions of the spent fuel pool to reflect final analyzed enrichment values, and the spacing changes between Regions II and III.

Conclusions for Dropped/Misloaded Fuel Events

The results from the three limiting postulated fuel assembly misload and fuel assembly drop scenarios are given in Table 2 below, as taken from WCAP-16721. These results show that a fresh 5 w/o U-235 enriched fuel assembly misloaded into an interior location in Region III is the worst single scenario, both in terms of reactivity change and amount of soluble boron needed (402 ppm) to compensate for the possible reactivity insertion.

The proposed Technical Specification changes do not alter existing Millstone Unit 3 Technical Specification 3.9.1.2 which will continue to require 800 ppm of soluble boron at all times fuel is present in the spent fuel pool. The basis of TS 3.9.1.2 is solely to provide protection for the possible fuel misload or fuel drop events that could cause reactivity insertions. Therefore, there is a wide margin of safety present in the 800 ppm of soluble boron in the spent fuel pool as required by Technical Specifications, and the maximum value of 402 ppm reported in WCAP-16721.

Table 1 Results from Initial Evaluation of Postulated Fuel Mishandling and Dropped Fuel Assembly Scenarios

Scenario	$k_{eff} \pm \sigma$	Relative Change to Base Case (Δk_{eff})
Misloaded fuel assembly in Region I Near Region II	0.92115 ± 0.00030	0.00058
Misloaded fuel assembly in Region I Near Region III	0.92097 ± 0.00030	0.00040
Misloaded fuel assembly in Region I Near SFP Corner	0.92009 ± 0.00032	-0.00048
Misloaded fuel assembly in Center of Region I	0.92230 ± 0.00035	0.00173
Misloaded fuel assembly in Region II Near Region III	0.93198 ± 0.00032	0.01141
Misloaded fuel assembly in Region II Near SFP Corner	0.92049 ± 0.00032	-0.00008
Misloaded fuel assembly in Region II Near Region I	0.93028 ± 0.00033	0.00971
Misloaded fuel assembly in Center of Region II	0.95922 ± 0.00029	0.03865
Misloaded fuel assembly in Region III Near SFP Corner	0.95966 ± 0.00029	0.03909
Misloaded fuel assembly in Region III Near Region I	0.92055 ± 0.00022	-0.00002
Misloaded fuel assembly in Region III Near Region II	0.96783 ± 0.00029	0.04726
Misloaded fuel assembly in Center of Region III	**1.00257 ± 0.00029**	**0.08200**
Dropped fuel assembly in Corner Between Regions I and II	0.95471 ± 0.00026	0.03414
Dropped fuel assembly Between Regions I and II	**0.96048 ± 0.00028**	**0.03991**
Dropped fuel assembly in Corner Between Regions II and III	0.97581 ± 0.00025	0.05524
Dropped fuel assembly Between Regions II and III	**0.98380 ± 0.00028**	**0.06323**
Dropped fuel assembly Between Regions I and III	0.96240 ± 0.00027	0.04183

Note - scenarios shown in bold were limiting cases analyzed and reported in WCAP-16721

Table 2 Results from Final Evaluation of Postulated Fuel Mishandling and Dropped Fuel Assembly Scenarios, from WCAP-16721-P section 4.5.2

Scenario	Relative Change to Base Case (Δk_{eff})	Required Soluble Boron Concentration (ppm)
Misloaded fuel assembly in Center of Region III	**0.07519**	**402**
Dropped fuel assembly Between Regions I and II	**0.03021**	**325**
Dropped fuel assembly Between Regions II and III	**0.01733**	**170**

Regions I and II Spent Fuel Pool Temperature for Maximum Reactivity

The nominal spent fuel pool temperature considered in all Regions I and II calculations in the spent fuel pool environment for WCAP-16721 is 68 °F. A bias to account for operation at a more reactive temperature in the operating range was developed and applied in the sum of biases and uncertainties. This bias was determined to be zero in the analysis for Regions I and II, indicating that the nominal conditions of 68 °F and water density of 1 g/cm^3 were the most reactive for the analyses shown in WCAP-16721.

Subsequently, it was discovered in response to RAI 8a that a small temperature bias did exist if the bulk pool temperature were lowered to 32 °F. The specific temperature bias values reported in the response to RAI 8a, are 0.00080 Δk_{eff} for Region I and 0.00075 Δk_{eff} for Region II. These temperature bias values include the Δk_{eff} between 68 °F and 32 °F, as well as the Monte Carlo (1 sigma) uncertainty for each case. Additional reactivity data for Regions I and II at spent fuel pool temperatures ranging from 32 °F to 160 °F were also provided in the response to RAI 8a, but the 32 °F spent fuel pool temperature was the most reactive temperature for both Regions I and II.

As documented in the response to RAI 6e and 8a, the reactivity margin inherent in the radial neutron leakage from the rack modules, in excess of 0.00150 Δk_{eff}, provides more than enough reactivity margin to compensate for the less than 0.00100 Δk_{eff} increase due to the temperature bias between 68 °F and 32 °F.

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