Dominion Nuclear Connecticut, Inc.
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Dominion®

September 30, 2008

Withhold Attachment 1 from Public Disclosure Under 2.390(a)(4)

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2378

Serial No.:	08-0511A
NLOS/GAW	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 STORAGE 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). Subsequently, the Nuclear Regulatory Commission (NRC) sent DNC a request for additional information (RAI) via letter dated August 8, 2008. The response to the RAI questions are provided in the attachments to this letter.

Attachment 1 provides the proprietary responses to RAI questions 1 through 17, provided by Westinghouse Electric Company, LLC which DNC is requesting to be withheld from Public Disclosure in Accordance with 10 CFR 2.390.

Attachment 2 contains the non-proprietary version of the responses to RAI questions 1 through 17.

Attachment 3 provides an affidavit, signed by Westinghouse Electric Company LLC, the owner of the information, attesting to the proprietary information contained in Attachment 1. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information, provided in Attachment 1, which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference Westinghouse letter CAW-08-2478 and should be addressed to Mr. J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

A001
NRR

Attachment 4 provides the non-proprietary responses to RAI questions 18 and 19 provided by DNC.

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 Mr. Geoffrey Wertz at 804-273-3572.

Sincerely,

Leslie N. Hartz
Vice President – Nuclear Support Services

Commitments made in this letter: None

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Support Services of Dominion Nuclear Connecticut, Inc. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 30th day of September, 2008.

My Commission Expires: 4/30/09.

Notary Public

GINGER LYNN ALLIGOOD
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2009

Attachments:

1. Attachment 1: NEU-08-05, Rev 1, Attachment 1 (Proprietary) Westinghouse Response to Request for Additional Information (RAI) Regarding the Spent Fuel Pool Criticality Amendment Request
2. Attachment 2: NEU-08-05, Rev 1, Attachment 2 (Non-Proprietary) Westinghouse Response to Request for Additional Information (RAI) Regarding the Spent Fuel Pool Criticality Amendment Request
3. Attachment 3: Westinghouse letter, CAW-08-2478, "APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE"
4. Attachment 4: Non-Proprietary Response to RAI Questions 18 and 19

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 2

NON-PROPRIETARY RESPONSE TO RAI QUESTIONS 1 – 17

FOR THE SPENT FUEL POOL LICENSE AMENDMENT REQUEST

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

NEU-08-50, Rev 1, Attachment 2 (Non-Proprietary) Westinghouse Response to Request for Additional Information (RAI) Regarding the Spent Fuel Pool Criticality Amendment Request

Question 1

WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, indicates benchmark analyzes where performed to justify the axial nodalization used in the criticality analysis. Provide the description and results of those benchmarks analyzes.

Response:

The benchmark comparison performed to demonstrate the acceptability of the 4-zone model was a comparison to a 7-zone model. Both the 4- and 7-zone models have three fine mesh point at the top of the fuel assembly. The 7-zone model also has three symmetric fine mesh point at the bottom of the fuel assembly. Both models are based on Profile 5 from Reference 5. The top and bottom ends of the assemblies are of particular importance to the overall assembly reactivity in the spent fuel pool environment as these regions have lower depletion and therefore higher reactivity than the central portion of the assembly. Since the overall reactivity is dominated by the top end of the fuel assembly, where the burnup gradient is accurately captured in the 4-zone model, the 4-zone model is an adequate representation of the discharged fuel assembly.

The calculations were performed in SCALE 4 using KENO V.a, and represent only fuel assemblies surrounded by full density water. No racks are modeled, so the conclusions of this benchmark are generically applicable to all types of fuel storage racks. The study depleted 5.0 weight percent (w/o) fuel in both mesh point structures to 55,000 mega watt day per metric ton uranium (MWd/MTU) and 65,000 MWd/MTU to detect any differences that occur at high burnups. Further calculations were performed considering 10 years of ^{241}Pu decay. These results, as well as the differences and root-sum-squared uncertainties, are provided in the table below.

These results indicate that the 4-zone model is comparable in accuracy to the 7-zone model. All results agree to within less than $2 \cdot (\sigma^2_{4\text{-zone}} + \sigma^2_{7\text{-zone}})^{0.5}$, which provides greater than 95% confidence that the two models are of equivalent reactivity. It should also be noted that neither model is more reactive in all four cases. Furthermore, these results indicate that the bottom fuel assembly burnup gradient does not contribute significantly to overall reactivity in the spent fuel pool environment. Therefore, it is concluded that the 4-zone model can be employed with confidence for use in the burnup credit calculations.

Comparison of 7-zone and 4-zone Models

Condition	7-Zone Model		4-Zone Model		Difference	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
55,000 BU 0 yr Decay	1.12037	0.00058	1.11901	0.00064	0.00136	0.00086
55,000 BU 10 yr Decay	1.09285	0.00061	1.09217	0.00064	0.00068	0.00088
65,000 BU 0 yr Decay	1.07643	0.00063	1.07774	0.00062	-0.00131	0.00088
65,000 BU 10 yr Decay	1.04648	0.00061	1.04477	0.00072	0.00171	0.00094

k_{eff}: Effective multiplication factor

σ: Standard deviation

RSS σ: Root-sum-square standard deviation

Question 2

WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, states, "Input to this analysis is based on a limiting axial burnup profile data provided in the Department of Energy [DOE] Topical Report, as documented in Reference 12. The burnup profile in the DOE topical report is based on a database of 3169 axial burnup profiles for pressurized water reactor [PWR] fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data." However, the DOE Topical Report, (Reference 8 herein) does not have a limiting axial burnup profile. Rather, burnup is divided into 12 groups with each interval having a limiting axial burnup profile. For ease of use, those 12 groups are compressed into three intervals. The axial burnup profile indicated in Figure 2-1 is indicated by the DOE Topical Report and NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," (Reference 6) as being conservative for burnups greater than or equal to 30 giga watt day per metric ton uranium (GWD/MTU), but non-conservative below 30 GWD/MTU. WCAP-16721 has used this burnup profile for burnups of 5, 15, and 25 GWD/MTU. Provide a justification for the use of this axial burnup profile below 30 GWD/MTU.

Response:

The responses to RAIs 2 and 3 provide a justification for the axial burnup profiles used in WCAP-16721-P. The majority of the technical justification presented is provided in response to RAI 3. The majority of the information is presented there because the uniform profile tends to be the limiting profile between 10 and 30 GWd/MTU burnup. Comparisons are only presented for Regions II and III because the burnup limit in Region I is less than 10 GWd/MTU, and therefore only the uniform shape need be considered.

The analysis in WCAP-17621-P uses two burnup profiles: Profile 5 from Reference 5 and a uniform burnup profile. At each burnup step, the infinite array reactivity determined in the spent fuel pool rack environment is determined based on the isotopic number densities generated from *both* burnup profiles. The higher reactivity is then selected at each burnup step. The table below shows the profile which generates higher reactivity for burnup steps from 10,000 MWd/MTU to 60,000 MWd/MTU for Regions II and III. Note that the distributed burnup profile (Profile 5 from Reference 5) is only limiting for burnups less than 30,000 MWd/MTU for 3.0 w/o cases at 25,000 MWd/MTU. The k_{eff} calculated at this burnup is less than 0.88 in Region II and less than 0.93 in Region III. Further justification of the application of a uniform burnup profile between 10,000 and 30,000 MWd/MTU is provided in response to RAI 3.

Limiting Burnup Profile for Regions II and III and All Burnup Steps

Burnup (MWd/MTU)	Region II			Region III		
	3.0 w/o	4.0 w/o	5.0 w/o	3.0 w/o	4.0 w/o	5.0 w/o
5,000	Uniform	Uniform	Uniform	Uniform	Uniform	Uniform
15,000	Uniform	Uniform	Uniform	Uniform	Uniform	Uniform
25,000	Distributed	Uniform	Uniform	Distributed	Uniform	Uniform
35,000	Distributed	Distributed	Distributed	Distributed	Distributed	Distributed
45,000	Distributed	Distributed	Distributed	Distributed	Distributed	Distributed
55,000	Distributed	Distributed	Distributed	Distributed	Distributed	Distributed

Question 3

WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, states, "A key aspect of the burnup credit methodology employed in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the depleted fuel assemblies. This effect can be important in the analysis of the fuel assembly characteristics when the majority of spent fuel assemblies stored in the MPS3 spent fuel pool have a discharge burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative. Therefore, it is necessary to consider both uniform and axially distributed burnup profiles, and the more conservative representation will be utilized to determine fuel assembly storage requirements." Subsequent statements indicate that only a uniform axial burnup profile was used for the "Region I 4-out-of-4" storage configuration whereas a uniform axial burnup profile was used in the Region II and Region III storage configurations to compare with the results of the axially distributed profile mentioned above. These statements indicate WCAP-16721 has used a uniform axial burnup profile for burnups of 5, 15, 25, 35, 45, and 55 GWD/MTU. A uniform axial burnup profile is generally accepted as conservative at low burnup. There is a transition point at which the uniform axial profile becomes non-conservative. As burnup increases beyond this transition point, the uniform axial burnup profile becomes ever more non-conservative. However, exactly where a uniform axial burnup profile transitions from conservative to non-conservative is not generically known. There is no evidence in the LAR that this transition point was established for MPS3 fuel. Based on literature familiar to the NRC staff, a uniform axial burnup should be considered in the following manner: conservative for burnup (BU) < 10 GWD/MTU, non-conservative for BU > 20 GWD/MTU, indeterminate for BU between those values. Provide a justification for the use of the uniform axial burnup profile above 10 GWD/MTU.

Response:

As discussed in the response to RAI 2, two axial burnup profiles were considered in the analysis presented in WCAP-16721-P. The more reactive infinite array multiplication factor is reported in Table 4-9 for Region II, or Table 4-10 for Region III. Only the uniform profile is considered for Region I as the burnup limits are all below 10,000 MWd/MTU. The point at which the conservative burnup profile changes from uniform to distributed is apparent in the table in the response to RAI 2, and is dependent on the initial enrichment of the fuel assembly being considered. For 3.0 w/o fuel, the transition point occurs between 15,000 and 25,000 MWd/MTU, and for 4.0 w/o and 5.0 w/o fuel between 25,000 and 35,000 MWd/MTU.

The justification for the use of the uniform burnup profile is based on a review of all burnup shapes at Millstone Power Station Unit 3 (MPS3) from cycles 1 – 13. This database represents more than 200 unique axial burnup profiles calculated by the Westinghouse core design nodal code, ANC. Three different axial assembly designs have been used during MPS3 operation: fuel with no axial blankets, fuel with natural enrichment axial blankets, and fuel with slightly enriched (2.6 w/o) axial blankets. Each of these axial designs is treated separately in this justification.

Unblanketed Fuel

The first four regions of fuel used at MPS3, that is fuel fed into cycles 1 and 2, used no axial blankets. Since unblanketed fuel is no longer used in MPS3 reload core designs, the focus of the justification for these fuel assemblies is assessing the safety of storage based on as-built and as-operated conditions. With this in mind, only discharge burnup shapes are considered for these assemblies. Furthermore, since the axial end effect will drive reactivity beyond the burnup at which the uniform burnup profile is limiting, low burnup in the top two nodes will indicate a more reactive profile. The shapes with the lowest nodes 1 and 2 burnup are considered most limiting. The most limiting unblanketed fuel assembly burnup profiles from cycles 1 through 3 are listed in the tables shown below. It can be seen from the table that the shape in location 1,4 is the most limiting profile as it has the lowest burnup in the first two nodes.

For computational efficiency, these fuel depletions and KENO calculations were performed using the PARAGON code and SCALE Version 5.1. These computer codes perform the same functions as the PHOENIX-P and SCALE Version 4.4 codes used in WCAP-16721-P. The 97.5% of theoretical density case was depleted again so that the reactivity differences could be determined using consistent code versions.

A series of depletion calculations was performed to assess the infinite array reactivity based on the limiting axial burnup profile identified from cycle 1. The discharged burnup of the limiting assembly, B58, is 20,668 MWd/MTU. Credit is taken, as mentioned above, for the as-built pellet stack density and non-uprated operating conditions. The pellet stack density used is 93.83% of theoretical density. This is the product of the as-built density and the volume of UO_2 in the pellet region, thus accounting for dishing and chamfering. The calculated infinite array k_{eff} for this burnup shape at 20,668 MWd/MTU is 0.89962 ± 0.00036.

The uniform burnup profile was used to deplete fuel to 20,668 MWd/MTU for comparison to the k_{eff} calculated above. The isotopics generated were only considered in Region II as fuel stored in Region III that was never used in a cycle operated at 3650 MWt is not covered in WCAP-16721-P. Region I is also neglected given the extremely low burnup requirement. The Region II infinite array k_{eff} with the uniform burnup profile depleted to 20,668 MWd/MTU is 0.89986 ± 0.00032.

The Region II uniform burnup profile k_{eff} is slightly higher than that calculated with the most reactive discharged burnup profile from unblanketed fuel. The depletion calculations were performed at a constant soluble boron concentration of 1000 ppm. The cycle 1 average soluble boron concentration was approximately 650 ppm. The increased boron concentration adds additional conservatism to the reactivity determination for the cycle 1 discharged assembly. It should also be noted that no credit is taken for ^{241}Pu decay in this comparison. The limiting assembly, B58, was discharged from the core over 20 years ago, but the maximum amount of decay credit claimed in Region II is 10 years. This decay credit would be worth approximately 1.7% Δk_{eff}. The additional margin provided by this decay credit is sufficient to provide confidence that the uniform burnup profile will bound the reactivity of all unblanketed fuel assemblies used at MPS3.

Discharged Burnup Shapes from Cycle 1

Node Midpoint (inches)	Node Average Relative Burnup						
	1,4*	2,3*	1,2*	4,6*	3,7*	5,5*	1,7*
141	0.391	0.391	0.392	0.407	0.410	0.410	0.411
135	0.635	0.636	0.637	0.648	0.653	0.647	0.654
129	0.825	0.826	0.827	0.830	0.834	0.826	0.835
123	0.945	0.946	0.947	0.943	0.946	0.938	0.946
117	1.018	1.019	1.020	1.012	1.014	1.007	1.014
111	1.061	1.062	1.062	1.053	1.054	1.048	1.054
105	1.086	1.086	1.086	1.078	1.078	1.073	1.077
99	1.100	1.101	1.101	1.092	1.092	1.088	1.092
93	1.110	1.110	1.110	1.102	1.101	1.098	1.101
87	1.116	1.116	1.116	1.109	1.108	1.106	1.108
81	1.122	1.122	1.121	1.116	1.114	1.113	1.114
75	1.127	1.127	1.127	1.122	1.120	1.120	1.120
69	1.133	1.133	1.132	1.128	1.127	1.127	1.126
63	1.140	1.139	1.139	1.135	1.134	1.135	1.133
57	1.147	1.146	1.145	1.143	1.141	1.143	1.141
51	1.154	1.153	1.152	1.150	1.148	1.151	1.148
45	1.160	1.159	1.159	1.157	1.155	1.158	1.154
39	1.163	1.162	1.161	1.160	1.158	1.162	1.158
33	1.158	1.157	1.156	1.155	1.153	1.158	1.153
27	1.136	1.136	1.135	1.135	1.133	1.138	1.133
21	1.082	1.082	1.082	1.084	1.083	1.089	1.083
15	0.971	0.971	0.971	0.979	0.979	0.986	0.980
9	0.765	0.765	0.765	0.782	0.783	0.791	0.784
3	0.456	0.456	0.456	0.479	0.480	0.489	0.481

* Quarter core location.

Discharged Burnup Shapes from Cycle 2

Node Midpoint (inches)	Node Average Relative Burnup						
	6,7	4,8	2,4	3,5	3,3	1,3	5,5
141	0.430	0.431	0.470	0.471	0.471	0.472	0.472
135	0.680	0.682	0.715	0.716	0.717	0.717	0.719
129	0.860	0.862	0.883	0.885	0.886	0.886	0.885
123	0.966	0.967	0.975	0.978	0.978	0.978	0.976
117	1.026	1.027	1.025	1.027	1.027	1.027	1.026
111	1.060	1.061	1.051	1.054	1.054	1.054	1.053
105	1.080	1.080	1.066	1.069	1.068	1.068	1.067
99	1.091	1.091	1.075	1.077	1.077	1.077	1.076
93	1.098	1.098	1.082	1.083	1.083	1.083	1.082
87	1.104	1.103	1.087	1.088	1.088	1.088	1.087
81	1.108	1.107	1.092	1.093	1.092	1.092	1.092
75	1.113	1.112	1.097	1.097	1.097	1.097	1.096
69	1.118	1.117	1.102	1.102	1.102	1.102	1.101
63	1.123	1.122	1.108	1.107	1.107	1.107	1.107
57	1.129	1.128	1.114	1.112	1.113	1.112	1.112
51	1.135	1.134	1.120	1.118	1.118	1.118	1.118
45	1.140	1.139	1.125	1.123	1.124	1.123	1.124
39	1.143	1.142	1.129	1.127	1.127	1.127	1.127
33	1.140	1.139	1.129	1.126	1.127	1.126	1.127
27	1.123	1.122	1.118	1.115	1.116	1.115	1.116
21	1.078	1.078	1.083	1.081	1.081	1.081	1.081
15	0.980	0.981	1.002	1.000	1.000	1.000	1.000
9	0.790	0.791	0.825	0.823	0.823	0.823	0.827
3	0.484	0.486	0.526	0.526	0.525	0.526	0.529

Discharged Burnup Shapes from Cycle 3

Node Midpoint (inches)	Node Average Relative Burnup		
	1,1	3,5	2,8
141	0.449	0.453	0.461
135	0.704	0.706	0.724
129	0.890	0.888	0.906
123	0.984	0.983	0.997
117	1.033	1.033	1.040
111	1.060	1.061	1.062
105	1.075	1.077	1.073
99	1.084	1.086	1.080
93	1.090	1.092	1.085
87	1.095	1.097	1.090
81	1.099	1.101	1.094
75	1.104	1.105	1.098
69	1.108	1.109	1.102
63	1.113	1.114	1.107
57	1.118	1.118	1.111
51	1.123	1.123	1.116
45	1.127	1.127	1.120
39	1.130	1.129	1.123
33	1.128	1.126	1.122
27	1.116	1.113	1.112
21	1.082	1.076	1.079
15	0.997	0.990	0.995
9	0.799	0.798	0.804
3	0.490	0.494	0.496

Natural Blanket Assemblies

Fuel assemblies with axial blankets of natural enrichment, that is approximately 0.74 w/o ^{235}U, were used at MPS3 from region 5 through region 8. These assemblies were used starting in cycle 3 and continued in use until cycle 9. All natural blanket assembly axial burnup profiles were reviewed, and the most limiting shape for each cycle is provided in the table below, regardless of whether it is a discharged burnup shape or not. This means that some of the profiles may have seen additional exposure, which would have flattened the axial burnup distribution compared to the end of the previous cycle. This is a conservative approach. The most reactive shape comes from cycle 6, in quarter core location 1,7 (assembly H23 and its symmetric partners H17, H18, and H28). This profile is chosen because it has the lowest burnup in the top two nodes.

A series of depletion calculations was performed using the cycle 6 axial burnup profile, including depletion of the blanket region. The natural enrichment blanket was modeled as 1.0 w/o ^{235}U to increase its reactivity, which is conservative. The natural enrichment blanket was only considered at the top of the fuel assembly, which is also conservative. The central 11 feet of the assembly were modeled as 5.0 w/o as this bounds the enrichment that was used in assemblies with natural blankets. It also increases the overall reactivity of the assembly, thus making this approach a conservative representation of the reactivity of these assemblies. A 4-zone axial representation was used with the same axial nodes as those used in WCAP-16721-P to simplify the analysis. The depletion calculations were also performed at uprated conditions for power and temperature to add additional conservatism. KENO calculations were performed every 5,000 MWd/MTU from 10,000 to 30,000 MWd/MTU, inclusive, for comparison with the uniform burnup profile used in WCAP-16721-P. These KENO calculations were performed in both Region II and Region III. The results are provided for each region in the tables below and demonstrate at least 0.00459 Δk_{eff} of margin relative to the uniform depletion shape used in WCAP-16721-P.

Natural Blanket Burnup Profiles from Cycles 3 - 9

Node Midpoint (inches)	Node Average Relative Burnup						
	3	4	5	6	7	8	9
141	0.166	0.144	0.153	0.118	0.209	0.227	0.250
135	0.687	0.648	0.603	0.542	0.695	0.713	0.709
129	0.894	0.886	0.837	0.746	0.893	0.906	0.901
123	1.009	1.019	0.971	0.892	0.995	1.003	1.001
117	1.065	1.085	1.039	0.990	1.045	1.049	1.049
111	1.095	1.117	1.074	1.058	1.070	1.072	1.072
105	1.112	1.131	1.092	1.105	1.082	1.083	1.084
99	1.122	1.139	1.102	1.139	1.090	1.090	1.090
93	1.129	1.143	1.111	1.164	1.095	1.094	1.095
87	1.135	1.146	1.119	1.183	1.100	1.098	1.099
81	1.140	1.150	1.128	1.197	1.104	1.101	1.103
75	1.145	1.153	1.137	1.208	1.108	1.105	1.107
69	1.150	1.157	1.144	1.217	1.113	1.109	1.111
63	1.155	1.160	1.152	1.223	1.118	1.113	1.115
57	1.160	1.164	1.159	1.225	1.123	1.117	1.119
51	1.164	1.169	1.167	1.224	1.128	1.122	1.123
45	1.167	1.172	1.174	1.217	1.133	1.126	1.127
39	1.169	1.174	1.179	1.200	1.137	1.129	1.130
33	1.164	1.170	1.177	1.170	1.138	1.129	1.129
27	1.147	1.151	1.158	1.117	1.128	1.119	1.119
21	1.101	1.096	1.104	1.026	1.094	1.087	1.085
15	0.990	0.968	0.971	0.875	1.003	1.001	0.995
9	0.766	0.713	0.707	0.641	0.797	0.801	0.791
3	0.178	0.151	0.172	0.131	0.260	0.271	0.265

Comparison of Reactivity Results for Region II

Burnup (MWd/MTU)	Uniform		Cycle 6 Shape		Difference	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.11438	0.00039	1.10935	0.00039	0.00503	0.00055
15,000	1.08050	0.00035	1.07329	0.00034	0.00721	0.00049
20,000	1.04820	0.00033	1.04029	0.00033	0.00791	0.00047
25,000	1.01757	0.00036	1.00986	0.00031	0.00771	0.00048
30,000	0.98738	0.00032	0.98279	0.00034	0.00459	0.00047

Comparison of Reactivity Results for Region III

Burnup (MWd/MTU)	Uniform		Cycle 6 Shape		Difference	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.16481	0.00032	1.15925	0.00031	0.00556	0.00045
15,000	1.13105	0.00035	1.12375	0.00030	0.00730	0.00046
20,000	1.09908	0.00032	1.09052	0.00031	0.00856	0.00045
25,000	1.06827	0.00032	1.05958	0.00029	0.00869	0.00043
30,000	1.03770	0.00029	1.03196	0.00033	0.00574	0.00044

Enriched Blanket Assemblies

Fuel assemblies with axial blankets of 2.6 w/o enrichment were used at MPS3 since region 9. These assemblies were used starting in cycle 7. All enriched blanket assembly axial burnup profiles were reviewed, and the most limiting shape for each cycle is provided in the table below, regardless of whether it is a discharged burnup shape or not. This means that some of the profiles may have seen additional exposure, which would have flattened the axial burnup distribution compared to the end of the previous cycle. This is a conservative approach. The most reactive shape comes from cycle 12, in quarter core location 6, 6 (assembly P40 and its symmetric partners P37, P38, and P39). This shape is most reactive, as with the previous fuel assembly designs, because it has the lowest burnup in the top two nodes.

A series of depletion calculations was performed using the cycle 12 axial burnup profile, including depletion of the blanket region. The blanket was modeled as 2.6 w/o ^{235}U, and only considered at the top of the fuel assembly. The central 11 feet of the assembly were modeled as 5.0 w/o as this bounds the enrichment that was used in assemblies with enriched blankets. It also increases the overall reactivity of the assembly, thus making this approach a conservative representation of the reactivity of these assemblies. A 4-zone axial representation was used with the same axial nodes as those used in WCAP-16721-P to simplify the analysis. The depletion calculations were also performed at uprated conditions for power and temperature to add additional conservatism. KENO calculations were performed every 5,000 MWd/MTU from 10,000 to 30,000 MWd/MTU, inclusive, for comparison with the uniform burnup profile used in WCAP-16721-P. These KENO calculations were performed in both Region II and Region III. The results are provided for each region in the tables below and demonstrate at least 0.00317 Δk_{eff} of margin relative to the uniform depletion shape used in WCAP-16721-P.

Enriched Blanket Burnup Profiles from Cycles 7 – 13

Node Midpoint (inches)	Node Average Relative Burnup						
	7	8	9	10	11	12	13
141	0.364	0.382	0.373	0.391	0.389	0.381	0.401
135	0.723	0.754	0.716	0.706	0.702	0.698	0.705
129	0.898	0.928	0.890	0.869	0.868	0.867	0.865
123	0.993	1.009	0.979	0.962	0.964	0.964	0.956
117	1.036	1.044	1.023	1.010	1.014	1.015	1.005
111	1.055	1.059	1.045	1.036	1.040	1.042	1.032
105	1.063	1.066	1.058	1.051	1.056	1.058	1.047
99	1.068	1.070	1.066	1.061	1.066	1.068	1.058
93	1.072	1.072	1.073	1.070	1.073	1.076	1.067
87	1.076	1.075	1.078	1.077	1.080	1.083	1.074
81	1.080	1.077	1.083	1.084	1.086	1.089	1.082
75	1.084	1.080	1.088	1.091	1.093	1.096	1.089
69	1.089	1.083	1.094	1.099	1.100	1.102	1.097
63	1.093	1.086	1.099	1.106	1.106	1.109	1.104
57	1.098	1.089	1.105	1.114	1.114	1.116	1.112
51	1.103	1.091	1.110	1.122	1.121	1.123	1.12
45	1.109	1.095	1.116	1.129	1.128	1.129	1.128
39	1.114	1.097	1.120	1.135	1.133	1.134	1.134
33	1.117	1.098	1.120	1.138	1.134	1.134	1.137
27	1.113	1.093	1.113	1.130	1.124	1.123	1.129
21	1.086	1.070	1.085	1.097	1.090	1.087	1.098
15	1.005	1.001	1.009	1.011	1.003	0.998	1.015
9	0.830	0.836	0.834	0.828	0.819	0.812	0.836
3	0.468	0.492	0.466	0.436	0.433	0.425	0.449

Comparison of Reactivity Results for Region II

Burnup (MWd/MTU)	Uniform		Cycle 12 Shape		Difference	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.11438	0.00039	1.11121	0.00034	0.00317	0.00052
15,000	1.08050	0.00035	1.07535	0.00033	0.00515	0.00048
20,000	1.04820	0.00033	1.04255	0.00031	0.00565	0.00045
25,000	1.01757	0.00036	1.01162	0.00030	0.00595	0.00047
30,000	0.98738	0.00032	0.98170	0.00032	0.00568	0.00045

Comparison of Reactivity Results for Region III

Burnup (MWd/MTU)	Uniform		Cycle 12 Shape		Difference	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.16481	0.00032	1.16078	0.00033	0.00403	0.00046
15,000	1.13105	0.00035	1.12611	0.00032	0.00494	0.00047
20,000	1.09908	0.00032	1.09340	0.00031	0.00568	0.00045
25,000	1.06827	0.00032	1.06177	0.00031	0.00650	0.00045
30,000	1.03770	0.00029	1.03143	0.00031	0.00627	0.00042

Question 4

WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, states, "Table 3-5 lists the fuel and moderator temperatures employed in the depletion calculations for the assembly-average burnup model and each node of the .... axial burnup model. These values are based on conservative temperature profiles for MPS3 at uprated conditions. The use of uprated conditions for depletion calculations - with increased power, moderator temperatures and fuel temperatures - lead to increased reactivity determinations at any given burnup relative to fuel irradiated in the core prior to the uprate." Table 3-5 indicates the core exit temperature used in the analysis is approximately 628° Fahrenheit (F). MPS3 SPU LAR, (Reference 1), Attachment 5, Table 2.8.3-1 lists the post-SPU nominal core inlet temperature as 556.4°F and the average temperature rise in the core as 71.6°F. This makes the nominal core exit temperature 628°F. Therefore, the temperature used in the analysis is a nominal value rather than a conservative value. MPS3 SPU LAR, (Reference 1), Attachment 5, Table 2.8.3-1 lists the pre-SPU nominal core exit temperature would be approximately 623.5°F. So while the post-SPU nominal core exit temperature exceeds the pre-SPU core exit temperature, it is not clear whether or not it bounds the pre-SPU maximum core exit temperature. Provide a justification for the use of the nominal core moderator and fuel temperatures in the depletion calculations.

Response:

WCAP-16721-P incorporates bounding values for both core average temperature and maximum core outlet temperature in all depletion calculations. WCAP-16721-P uses 594.5 °F for the core average temperature during uniform burnup fuel depletions, and 628 °F as the maximum core outlet temperature in calculations that use an axial temperature/burnup distribution. These are bounding analysis values for the SPU and are specified in MPS3 SPU LAR Attachment 5 Table 2.8.3-1. These values will bound actual plant operating conditions, as discussed below.

Table 2.8.3-1 provides, in the column labeled "SPU Analysis Value", the limiting values used in the SPU analysis. For example, Table 2.8.3-1 lists the SPU core average temperature of 594.5 °F and a corresponding vessel average temperature of 589.5 °F. The difference between the vessel average temperature and the core average temperature is due to bypass flow. Nominal plant operating temperatures are expected to be about 2.5 °F below these values, and a limit will be placed in plant procedures to limit the vessel average temperature to 589.5 °F for the SPU, which corresponds to a core average temperature of 594.5 °F. Vessel average temperatures must be used in plant limits, since core average temperature is not a directly measurable quantity.

Also in Table 2.8.3-1, the core outlet temperature is shown to be 628 °F as the "SPU Analysis Value". This value is based on a RCS flow of 363,200 gpm, which is the minimum allowed TS RCS flow for the SPU, and a core average temperature of 594.5 °F. Actual plant operating RCS flow will be substantially higher than the minimum. Since the core average temperature rise of 71.6 °F shown in Table 2.8.3-1 is based on the minimum RCS TS flow, the actual operating core temperature rise will be less than 71.6 °F. Since the actual operating plant core average temperature will be lower than 594.5 °F, as discussed above, and the actual core operating temperature rise will be less than 71.6 °F, then the maximum operating core outlet temperature must be less than 628 °F.

Based on the above, WCAP-16721-P uses values for both core average temperature (594.5 °F) and maximum core outlet temperature (628 °F) in its calculations that will bound plant operating conditions.

The use of the word "nominal" in Table 2.8.3-1 refers only to the nominal value used in the analysis and does not mean the plant actual operating nominal temperature.

Question 5

NRC staff guidance is to use the most reactive fuel (Reference 7). NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR [light water reactor] Fuel," (Reference 8) provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on criticality analysis in storage and transportation casks, the basic principals with respect to the depletion analysis apply generically to SFPs, since the phenomena occur in the reactor as the fuel is being used. The results have some translation to SFP criticality analyses, especially when the discussion includes the effect in an infinite lattice analysis, similar to that performed for SFP analyzes. The basic premise is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. Other than moderator/fuel temperature and soluble boron, none of the other core operating parameters are discussed in WCAP- 16721. Provide a discussion of these other core operating parameters and their impact on the MPS3 SFP criticality analysis.

Response:

NUREG/CR-6665, Reference 4, identifies both specific power and operating history effects as weakly correlated to increased reactivity for discharged fuel assemblies. The maximum impact noted in Reference 4 is approximately 0.00200 Δk_{eff}. This result is caused by reduced power operation near the end of assembly depletion. The depletion calculations supporting the analysis presented in WCAP-16721-P do not include part power operation. Instead, the soluble boron concentration is maintained at a constant value above the cycle average value for the entire depletion. The spectral hardening from the presence of boron, especially at the end of the cycle when the concentration is several hundred ppm above physical values, provides additional margin to account for this potential impact. The use of additional margin is the approach suggested in Reference 4 for accounting for the potential for operating history effects.

A series of calculations was performed to investigate the effect of burnable absorbers on spent fuel reactivity. MPS3 fuel management does not use fixed burnable absorbers, so the investigation involved the use of ZrB_2 IFBA only (i.e., an annular coating on the fuel pellet). The largest number of IFBA rods typically used in a fuel assembly is 156. This number was used for this investigation as it maximizes the spectral hardening effects of the presence of IFBA. A typical 156 IFBA pattern was used both in fuel depletion calculations and in KENO spent fuel models. A nominal 1.5X IFBA loading was used as it is representative of uprated MPS3 operation, and also maximizes the amount of BA in the assembly. The 4-zone distributed burnup model was used with the same relative burnups as documented in WCAP-16721-P. The depletion calculations were performed at MPS3 uprated conditions for power level and moderator temperatures (consistent with the conditions in WCAP-16721-P).

For computational efficiency, these fuel depletions and KENO calculations were performed using the PARAGON code and SCALE Version 5.1. These computer codes perform the same functions as the PHOENIX-P and SCALE Version 4.4 codes used in WCAP-16721-P. The 97.5% of theoretical density redepletion used in the response to RAI 6 parts a and b was used again so that the reactivity differences could be determined.

Two sets of KENO calculations were performed in both Region II and Region III. The first contains the depleted fuel with the remaining IFBA material conservatively omitted from the assemblies in the infinite array spent fuel pool model. The second models the residual IFBA explicitly in the KENO infinite array spent fuel pool model. The results of these calculations are provided in the tables below. They indicate that the presence of IFBA will cause spectral hardening and increase discharged reactivity by approximately 0.00100 to 0.00350 Δk_{eff} units. The inclusion of the remaining IFBA, however, more than compensates for this effect. The results indicate that neglecting IFBA in fuel assembly depletion is a conservative practice.

Results from Region II Calculations Neglecting IFBA in Spent Fuel Pool Model

Burnup (MWd/MTU)	No IFBA		IFBA Depletion (^{10}B omitted)		k_{eff} Difference (No IFBA – IFBA)	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.11194	0.00033	1.11365	0.00035	-0.00171	0.00048
15,000	1.07704	0.00034	1.07996	0.00034	-0.00292	0.00048
20,000	1.04485	0.00031	1.04718	0.00033	-0.00233	0.00045
25,000	1.01410	0.00035	1.01683	0.00032	-0.00273	0.00047
30,000	0.98667	0.00033	0.99009	0.00034	-0.00342	0.00047
40,000	0.93986	0.00034	0.94165	0.00031	-0.00179	0.00046
50,000	0.89750	0.00032	0.90095	0.00038	-0.00345	0.00050
60,000	0.86135	0.00033	0.86358	0.00038	-0.00223	0.00050

Results from Region II Calculations Containing IFBA in Spent Fuel Pool Model

Burnup (MWd/MTU)	No IFBA		IFBA Depletion (^{10}B included)		k_{eff} Difference (No IFBA – IFBA)	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.11194	0.00033	1.04742	0.00033	0.06452	0.00047
15,000	1.07704	0.00034	1.04467	0.00034	0.03237	0.00048
20,000	1.04485	0.00031	1.02848	0.00032	0.01637	0.00045
25,000	1.01410	0.00035	1.00559	0.00032	0.00851	0.00047
30,000	0.98667	0.00033	0.97990	0.00032	0.00677	0.00046
40,000	0.93986	0.00034	0.93449	0.00034	0.00537	0.00048
50,000	0.89750	0.00032	0.89497	0.00031	0.00253	0.00045
60,000	0.86135	0.00033	0.86011	0.00035	0.00124	0.00048

Results from Region III Calculations Neglecting IFBA in Spent Fuel Pool Model

Burnup (MWd/MTU)	No IFBA		IFBA Depletion (^{10}B omitted)		k_{eff} Difference (No IFBA – IFBA)	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.16213	0.00030	1.16298	0.00031	-0.00085	0.00043
15,000	1.12679	0.00031	1.12945	0.00029	-0.00266	0.00042
20,000	1.09474	0.00034	1.09677	0.00031	-0.00203	0.00046
25,000	1.06429	0.00029	1.06719	0.00031	-0.00290	0.00042
30,000	1.03608	0.00030	1.03873	0.00030	-0.00265	0.00042
40,000	0.98758	0.00032	0.99065	0.00031	-0.00307	0.00045
50,000	0.94443	0.00033	0.94752	0.00028	-0.00309	0.00043
60,000	0.90580	0.00031	0.90842	0.00029	-0.00262	0.00042

Results from Region III Calculations Containing IFBA in Spent Fuel Pool Model

Burnup (MWd/MTU)	No IFBA		IFBA Depletion (^{10}B included)		k_{eff} Difference (No IFBA – IFBA)	
	k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
10,000	1.16213	0.00030	1.09340	0.00033	0.06873	0.00045
15,000	1.12679	0.00031	1.09276	0.00034	0.03403	0.00046
20,000	1.09474	0.00034	1.07801	0.00031	0.01673	0.00046
25,000	1.06429	0.00029	1.05482	0.00032	0.00947	0.00043
30,000	1.03608	0.00030	1.02934	0.00031	0.00674	0.00043
40,000	0.98758	0.00032	0.98268	0.00031	0.00490	0.00045
50,000	0.94443	0.00033	0.94192	0.00033	0.00251	0.00047
60,000	0.90580	0.00031	0.90461	0.00030	0.00119	0.00043

Question 6

WCAP-16721 Section 2.4, Methodology Assumptions, lists some of the assumptions used in the criticality analysis. One assumption states, "The design basis limit for k_{eff} is conservatively reduced from 0.95 to 0.949 for this analysis." However, maintaining k_{eff} less than or equal to 0.95 at all times is not a design basis limit, it is a regulatory limit. Therefore, the analysis is only reserving 0.001 delta (Δ) k_{eff} analytical margin to the MPS3 licensing basis limit. Any identified non-conservatism or potential non-conservatism that can not be offset by an explicitly known conservatism will erode the reserved analytical margin. A significant erosion of the reserved analytical margin will preclude a reasonable assurance determination by the NRC staff. Please provide the following information regarding the assumptions listed in Section 2.4.

a) **Nominal and tolerances for fuel stack density. Identify the conservatism associated with using a fuel stack density of 10.686 gm/cm^3. Identify how this conservatism carries through to depleted fuel.**

b) **Nominal and tolerances for pellet dishing and chamfering. Identify the conservatism associated with modeling pellets as full right circular cylinders. Identify how this conservatism carries through to depleted fuel.**

c) **Justification for not modeling fuel assembly grids.**

d) **The reactivity worth of not modeling uranium-234 (^{234}U) and uranium-236 (^{236}U) in fresh fuel. Identify how this conservatism carries through to depleted fuel.**

e) **Alternatively, the licensee may quantify conservatisms elsewhere in the analysis.**

Response:

Parts a) and b)

A series of calculations were performed to quantify the amount of reactivity margin created by the conservative assumption made in WCAP-16721-P that the fuel pellet density for all pellets is 10.686 gm/cm^3. This density corresponds to 97.5% of nominal theoretical density. These calculations consider all four configurations in all three regions of the pool. In Region I, because of the low burnup requirements, only fresh fuel is considered. In Regions II and III both fresh and depleted fuel were used in calculations. As discussed below in more detail, a significant amount of reactivity margin is present relative to the nominal condition.

The nominal fuel stack density is [$]^{a,c}$ of theoretical density per Reference 1. The full theoretical density basis is 10.96 gm/cm^3 as stated in Reference 2. The nominal dishing and chamfering reduces the fissile volume in the pellet by an additional [$]^{a,c}$, from Reference 3.

Region I

Only fresh cases were explicitly considered in Region I. The conservatism associated with the 97.5% of theoretical density assumption was determined by using KENO to calculate the k_{eff} at [$]^{a,c}$ of theoretical density and [$]^{a,c}$ of theoretical density. These results were each subtracted from the base case (97.5% of theoretical density), with the results presented in the table below. A second table is provided which demonstrates that between 0.00300 and almost 0.00500 Δk_{eff} units of conservatism exist in both the "4-out-of-4" and "3-out-of-4" configurations at fresh conditions. The results for Regions II and III provide significant confidence that this conservatism is not eroded with depletion, especially considering the small amount of depletion credited in Region I.

Results for Different Pellet Densities in Region I

Configuration	97.5% TD		[$]^{a,c}$		[$]^{a,c}$	
	k_{eff}	σ	k_{eff}	σ	k_{eff}	σ
4-out-of-4	0.92613	0.00043	0.92229	0.00041	0.92119	0.00039
3-out-of-4	0.91880	0.00042	0.91557	0.00044	0.91454	0.00043

Conservatism for Different Pellet Densities in Region I

Configuration	[$]^{a,c}$		[$]^{a,c}$	
	Δk_{eff}	RSS σ	Δk_{eff}	RSS σ
4-out-of-4	0.00384	0.00059	0.00494	0.00058
3-out-of-4	0.00323	0.00061	0.00426	0.00060

Regions II and III

For Regions II and III, fresh and depleted cases were considered. The fresh cases were performed in the same manner as those for Region I. For depleted cases, fuel was depleted in the same manner as was done in the original WCAP-16721-P analysis using the uniform and distributed burnup profiles. The higher k_{eff} from the two cases was selected for comparison. Initial enrichments of 3.0, 4.0, and 5.0 w/o ^{235}U were considered in both regions.

For computational efficiency, these fuel depletions and KENO calculations were performed using the PARAGON code and SCALE Version 5.1. These computer codes perform the same functions as the PHOENIX-P and SCALE Version 4.4 codes used in WCAP-16721-P. The 97.5% of theoretical density case was depleted again so that the reactivity differences could be determined.

Three tables are provided below with the results of these calculations. The first two tables provide the k_{eff} results from both Region II and Region III. The third table provides the difference in calculated k_{eff} for each case based on subtracting the reduced theoretical density k_{eff} from the 97.5% of theoretical density case. The root-sum-squared one sigma uncertainties are also provided.

As can be seen in the final table, a significant amount of conservatism is identified in each case. At low burnups, dominated by the uniform burnup distribution, the identified conservatism ranges from approximately 0.00100 Δk_{eff} to more than 0.00600 Δk_{eff}. At high burnups, limited by the distributed burnup profile, the conservatism increases and ranges from 1% Δk_{eff} to over 1.9% Δk_{eff}.

Results for Region II for Different Theoretical Densities

Enrichment (w/o ^{235}U)	Burnup (MWd/MTU)	97.5% TD		[ ]^{a,c}		[ ]^{a,c}	
		k_{eff}	σ	k_{eff}	σ	k_{eff}	σ
1.79	0	0.93020	0.00032	0.92645	0.00033	0.92464	0.00034
3.0	10,000	0.98811	0.00031	0.98487	0.00031	0.98341	0.00033
	15,000	0.94907	0.00031	0.94608	0.00031	0.94435	0.00030
	20,000	0.91324	0.00031	0.90943	0.00031	0.90792	0.00030
	25,000	0.87988	0.00044	0.87537	0.00031	0.87287	0.00029
	30,000	0.85439	0.00039	0.84280	0.00030	0.84085	0.00030
	40,000	0.80966	0.00032	0.79433	0.00032	0.79177	0.00038
	50,000	0.77233	0.00033	0.75557	0.00026	0.75356	0.00032
	60,000	0.74102	0.00027	0.72462	0.00031	0.72190	0.00029
4.0	10,000	1.06127	0.00037	1.05948	0.00032	1.05800	0.00034
	15,000	1.02450	0.00035	1.02247	0.00031	1.02180	0.00034
	20,000	0.98994	0.00032	0.98720	0.00033	0.98631	0.00033
	25,000	0.95674	0.00031	0.95367	0.00031	0.95294	0.00031
	30,000	0.92751	0.00033	0.92158	0.00031	0.91985	0.00032
	40,000	0.88022	0.00032	0.86521	0.00034	0.86446	0.00031
	50,000	0.83919	0.00031	0.82332	0.00037	0.82135	0.00035
	60,000	0.80332	0.00036	0.78713	0.00030	0.78446	0.00031
5.0	10,000	1.11438	0.00039	1.11329	0.00039	1.11250	0.00035
	15,000	1.08050	0.00035	1.07907	0.00035	1.07864	0.00034
	20,000	1.04820	0.00033	1.04688	0.00033	1.04651	0.00033
	25,000	1.01757	0.00036	1.01598	0.00031	1.01488	0.00032
	30,000	0.98738	0.00032	0.98540	0.00032	0.98417	0.00033
	40,000	0.93986	0.00034	0.92612	0.00035	0.92483	0.00029
	50,000	0.89750	0.00032	0.88295	0.00037	0.88082	0.00032
	60,000	0.86135	0.00033	0.84489	0.00031	0.84362	0.00031

Results for Region III for Different Theoretical Densities

Enrichment (w/o ^{235}U)	Burnup (MWd/MTU)	97.5% TD		[ ]^{a,c}		[ ]^{a,c}	
		k_{eff}	σ	k_{eff}	σ	k_{eff}	σ
1.47	0	0.92158	0.00031	0.91733	0.00029	0.91525	0.00028
3.0	10,000	1.03931	0.00031	1.03600	0.00029	1.03514	0.00030
	15,000	0.99954	0.00028	0.99632	0.00030	0.99539	0.00029
	20,000	0.96234	0.00030	0.95867	0.00030	0.95713	0.00027
	25,000	0.92759	0.00031	0.92360	0.00027	0.92107	0.00027
	30,000	0.90040	0.00032	0.88850	0.00030	0.88707	0.00027
	40,000	0.85354	0.00028	0.83930	0.00034	0.83736	0.00029
	50,000	0.81427	0.00026	0.79917	0.00033	0.79666	0.00029
	60,000	0.78131	0.00030	0.76589	0.00028	0.76378	0.00027
4.0	10,000	1.11209	0.00032	1.11028	0.00029	1.10961	0.00031
	15,000	1.07617	0.00033	1.07296	0.00033	1.07250	0.00029
	20,000	1.04026	0.00030	1.03800	0.00030	1.03688	0.00030
	25,000	1.00679	0.00031	1.00434	0.00030	1.00302	0.00029
	30,000	0.97619	0.00030	0.97084	0.00027	0.96971	0.00028
	40,000	0.92738	0.00028	0.91340	0.00029	0.91229	0.00030
	50,000	0.88361	0.00031	0.87009	0.00032	0.86867	0.00036
	60,000	0.84677	0.00029	0.83190	0.00030	0.83005	0.00030
5.0	10,000	1.16481	0.00032	1.16350	0.00030	1.16278	0.00031
	15,000	1.13105	0.00035	1.12996	0.00030	1.12882	0.00032
	20,000	1.09908	0.00032	1.09772	0.00030	1.09787	0.00031
	25,000	1.06827	0.00032	1.06597	0.00029	1.06493	0.00033
	30,000	1.03770	0.00029	1.03609	0.00029	1.03513	0.00032
	40,000	0.98758	0.00032	0.97603	0.00030	0.97465	0.00029
	50,000	0.94443	0.00033	0.93092	0.00030	0.92955	0.00031
	60,000	0.90580	0.00031	0.89164	0.00032	0.88990	0.00029

Conservatism Identified from Different Theoretical Densities

Enrichment (w/o ^{235}U)	Burnup (MWd/MTU)	Region II				Region III			
		[ ]^{a,c}		[ ]^{a,c}		[ ]^{a,c}		[ ]^{a,c}	
		Δk_{eff}	RSS σ	Δk_{eff}	RSS σ	Δk_{eff}	RSS σ	Δk_{eff}	RSS σ
Max Fresh	0	0.00375	0.00046	0.00556	0.00047	0.00425	0.00042	0.00633	0.00042
3.0	10,000	0.00324	0.00044	0.00470	0.00045	0.00331	0.00042	0.00417	0.00043
	15,000	0.00299	0.00044	0.00472	0.00043	0.00322	0.00041	0.00415	0.00040
	20,000	0.00381	0.00044	0.00532	0.00043	0.00367	0.00042	0.00521	0.00040
	25,000	0.00451	0.00054	0.00701	0.00053	0.00399	0.00041	0.00652	0.00041
	30,000	0.01159	0.00049	0.01354	0.00049	0.01190	0.00044	0.01333	0.00042
	40,000	0.01533	0.00045	0.01789	0.00050	0.01424	0.00044	0.01618	0.00040
	50,000	0.01676	0.00042	0.01877	0.00046	0.01510	0.00042	0.01761	0.00039
	60,000	0.01640	0.00041	0.01912	0.00040	0.01542	0.00041	0.01753	0.00040
4.0	10,000	0.00179	0.00049	0.00327	0.00050	0.00181	0.00043	0.00248	0.00045
	15,000	0.00203	0.00047	0.00270	0.00049	0.00321	0.00047	0.00367	0.00044
	20,000	0.00274	0.00046	0.00363	0.00046	0.00226	0.00042	0.00338	0.00042
	25,000	0.00307	0.00044	0.00380	0.00044	0.00245	0.00043	0.00377	0.00042
	30,000	0.00593	0.00045	0.00766	0.00046	0.00535	0.00040	0.00648	0.00041
	40,000	0.01501	0.00047	0.01576	0.00045	0.01398	0.00040	0.01509	0.00041
	50,000	0.01587	0.00048	0.01784	0.00047	0.01352	0.00045	0.01494	0.00048
	60,000	0.01619	0.00047	0.01886	0.00048	0.01487	0.00042	0.01672	0.00042
5.0	10,000	0.00109	0.00055	0.00188	0.00052	0.00131	0.00044	0.00203	0.00045
	15,000	0.00143	0.00049	0.00186	0.00049	0.00109	0.00046	0.00223	0.00047
	20,000	0.00132	0.00047	0.00169	0.00047	0.00136	0.00044	0.00121	0.00045
	25,000	0.00159	0.00048	0.00269	0.00048	0.00230	0.00043	0.00334	0.00046
	30,000	0.00198	0.00045	0.00321	0.00046	0.00161	0.00041	0.00257	0.00043
	40,000	0.01374	0.00049	0.01503	0.00045	0.01155	0.00044	0.01293	0.00043
	50,000	0.01455	0.00049	0.01668	0.00045	0.01351	0.00045	0.01488	0.00045
	60,000	0.01646	0.00045	0.01773	0.00045	0.01416	0.00045	0.01590	0.00042

Part c)

A series of calculations was performed to investigate the conservatism of not modeling fuel assembly grids in the analysis presented in WCAP-16721-P. In each region, calculations are performed using models with and without grids modeled over a range of soluble boron concentrations of 0 – 500 ppm. This range covers the credited boron concentrations in the WCAP-16721-P analysis. In each region, the maximum credited fresh enrichment and a depleted 5.0 w/o case are considered. This approach allows for the differences in soluble boron worth between fresh and depleted fuel. The results provided below demonstrate that not modeling grids is conservative for all regions considered in WCAP-16721-P in most conditions. The data indicate that for the maximum fresh enrichment in Region III, a slight non-conservatism of less than 0.00040 Δk_{eff} may occur. It should be noted, however, that the calculated k_{eff} in these conditions is more than 15% Δk_{eff} below the 0.95 requirement. This potential non-conservatism can also be offset with conservatism in the technical specification (TS) required soluble boron concentration. WCAP-16721-P identifies a required boron concentration of 402 ppm, but the TS requires 800 ppm. The reactivity impact of this additional soluble boron in Region III, where the non-conservatism was identified, is approximately 5.5% Δk_{eff}, which is considerably more than the identified non-conservatism. The maximum non-conservatism identified, 0.00038 Δk_{eff}, is worth less than 2 ppm. The additional 398 ppm is sufficient to account for any potential non-conservatism resulting from neglecting grids in the neutronic models used in WCAP-16721-P.

Conservatism Calculated for Grid Omission in Region I

Boron Conc. (ppm)	Enrichment (w/o ^{235}U)	No Grids		With Grids		Conservatism	
		k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
0	3.85	0.93212	0.00028	0.93163	0.00027	0.00049	0.00039
100		0.92034	0.00027	0.91859	0.00027	0.00175	0.00038
200		0.90831	0.00021	0.90800	0.00021	0.00031	0.00030
300		0.89584	0.00020	0.89497	0.00020	0.00087	0.00028
400		0.88526	0.00020	0.88475	0.00020	0.00051	0.00028
500		0.87429	0.00020	0.87281	0.00019	0.00148	0.00028
0	5.0	0.88421	0.00019	0.88370	0.00018	0.00051	0.00026
100		0.87460	0.00018	0.87279	0.00018	0.00181	0.00025
200		0.86494	0.00019	0.86452	0.00018	0.00042	0.00026
300		0.85587	0.00018	0.85531	0.00018	0.00056	0.00025
400		0.84685	0.00017	0.84669	0.00018	0.00016	0.00025
500		0.83825	0.00019	0.83792	0.00018	0.00033	0.00026

Conservatism Calculated for Grid Omission in Region II

Boron Conc. (ppm)	Enrichment (w/o ^{235}U)	No Grids		With Grids		Conservatism	
		k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
0	1.81	0.93340	0.00016	0.93327	0.00015	0.00013	0.00022
100		0.91485	0.00031	0.91407	0.00033	0.00078	0.00045
200		0.89743	0.00034	0.89571	0.00032	0.00172	0.00047
300		0.88007	0.00032	0.87939	0.00029	0.00068	0.00043
400		0.86381	0.00031	0.86248	0.00030	0.00133	0.00043
500		0.84814	0.00030	0.84767	0.00032	0.00047	0.00044
0	5.0	0.89373	0.00029	0.89264	0.00029	0.00109	0.00041
100		0.88414	0.00028	0.88311	0.00030	0.00103	0.00041
200		0.87436	0.00031	0.87344	0.00029	0.00092	0.00042
300		0.86538	0.00027	0.86445	0.00028	0.00093	0.00039
400		0.85578	0.00029	0.85512	0.00028	0.00066	0.00040
500		0.84705	0.00027	0.84654	0.00027	0.00051	0.00038

Conservatism Calculated for Grid Omission in Region III

Boron Conc. (ppm)	Enrichment (w/o ^{235}U)	No Grids		With Grids		Conservatism	
		k_{eff}	σ	k_{eff}	σ	Δk_{eff}	RSS σ
0	1.45	0.91802	0.00009	0.91787	0.00010	0.00015	0.00013
100		0.88464	0.00009	0.88412	0.00009	0.00052	0.00013
200		0.85401	0.00009	0.85400	0.00010	0.00001	0.00013
300		0.82624	0.00009	0.82613	0.00010	0.00011	0.00013
400		0.80055	0.00009	0.80093	0.00009	-0.00038	0.00013
500		0.77724	0.00009	0.77749	0.00009	-0.00025	0.00013
0	5.0	0.81895	0.00008	0.81856	0.00008	0.00039	0.00011
100		0.79875	0.00008	0.79841	0.00008	0.00034	0.00011
200		0.78069	0.00008	0.78038	0.00008	0.00031	0.00011
300		0.76383	0.00008	0.76370	0.00008	0.00013	0.00011
400		0.74845	0.00008	0.74835	0.00008	0.00010	0.00011
500		0.73399	0.00008	0.73360	0.00008	0.00039	0.00011

Part d)

A series of calculations were performed to demonstrate the conservatism of neglecting ^{234}U and ^{236}U in fresh fuel. This conservatism is not carried into depleted fuel. Typical values for ^{234}U and ^{236}U content were obtained from recent as-built reports for MPS3 fuel. The last two reload regions were reviewed, and approximate averages of 0.05 w/o ^{234}U and 0.02 w/o ^{236}U were obtained. The models were modified by reducing the ^{238}U weight percent by the appropriate amount for modeling ^{234}U or ^{236}U as enrichment in ^{235}U is controlled. The effects of ^{234}U and ^{236}U were investigated separately. In all three regions, significant conservatism is demonstrated by neglecting ^{234}U, but no statistically significant reactivity effects are attributable to ^{236}U. The results are provided below.

Conservatism Determined for Neglecting ^{234}U and ^{236}U

Region	Enrichment	Case	k_{eff}	σ	Δk_{eff}	RSS σ
1	3.85	Base	0.93261	0.00043		
		0.05 w/o ^{234}U	0.93011	0.00040	0.00250	0.00059
		0.02 w/o ^{236}U	0.93176	0.00040	0.00085	0.00059
2	1.81	Base	0.93325	0.00036		
		0.05 w/o ^{234}U	0.93054	0.00031	0.00271	0.00048
		0.02 w/o ^{236}U	0.93371	0.00036	-0.00046	0.00051
3	1.45	Base	0.91809	0.00027		
		0.05 w/o ^{234}U	0.91420	0.00029	0.00389	0.00040
		0.02 w/o ^{236}U	0.91710	0.00030	0.00099	0.00040

Part e)

Given the conservatism identified for most conditions identified in Parts a – d, it is viewed that additional offsetting conservatism is not required for these areas. Significant conservatism is identified for all regions in parts a and b, relating to fuel stack density. The modeling of fuel assemblies without grids is demonstrated to be conservative in part c, with minor exceptions. The potential non-conservatism in Region III is significantly lower in magnitude than the conservatism identified in parts a and b. The practice of neglecting ^{234}U and ^{236}U in fresh fuel was demonstrated to be conservative in part d. Overall, conservatism was identified for all regions in the MPS3 spent fuel pool in responses to parts a – d.

An additional conservatism that can be credited is radial leakage from the spent fuel rack modules. A series of calculations was run which determined the reactivity of the finite array of each region individually in the entire spent fuel pool model. The fuel and storage racks for the other regions are omitted in each case. This allows for the determination of the radial leakage credit as the difference between the target k_{eff} determined using the infinite array model and the region specific spent fuel pool model. For conservatism, the KENO uncertainty for the region-specific full pool calculation is added to the calculated k_{eff} to reduce the radial leakage credit. These results are provided in the table below. It should be noted that more than 0.00150 Δk_{eff} of additional conservatism exist for each region. This conservatism will be used in the response to RAI 8.

Radial Leakage Credit for Each Region

Region	k_{eff}	σ	target k_{eff}	Conservatism
I	0.93078	0.00013	0.93279	0.00188
II	0.93166	0.00010	0.93339	0.00163
III	0.91569	0.00009	0.91852	0.00274

Many of the key components of the conservatism inherent in the analysis methodology used in WCAP-16721-P have been addressed in the response to this RAI. The conservatism of the depletion uncertainty, which is discussed in the response to RAI 12 should also be noted. The depletion uncertainty used in WCAP-16721-P is conservative by approximately more than 1%Δk_{eff} compared to the measured discrepancies between measured and predicted isotopic number densities. A more conservative estimate of the depletion uncertainty still demonstrates more than 0.9% Δk_{eff} conservatism. The conservatism demonstrated through neglecting IFBA effects, discussed in response to RAI 5, and through a bounding theoretical density assumption, discussed in response to RAI 6 parts a and b, is also significant. While the conservatism provided by residual IFBA is large at low burnup, the conservatism provided by the theoretical density assumption is large at high burnups. These two sources of conservatism dovetail to guarantee a significant amount of margin throughout the entire life cycle of the fuel assembly. The burnup profiles used in WCAP-16721-P, discussed in response to RAIs 2 and 3, provide a significant amount of reactivity margin for all blanketed fuel designs. The combination of profiles provides between 0.3% and 0.9% Δk_{eff} conservatism compared to bounding profiles from MPS3 operation. Taken together, these conservatism modeling features provide more than 2% Δk_{eff} quantified reactivity margin in the analysis presented in WCAP-16721-P beyond the 0.00100 Δk_{eff} of administrative margin.

Question 7

WCAP-16721 Section 3.5, Fuel Assembly Design Parameters, states, "The design parameters of the Westinghouse 17x17 STD [standard], V5H [vantage 5H], RFA [robust fuel assembly] and NGF [next generation fuel] fuel assembly types are summarized in Table 3-4 and Figure 3-3. Note that for the purposes of this analysis, the RFA and NGF dimensions are identical and will be treated as one fuel type. Simulations were performed for each storage configuration in this analysis to determine the fuel assembly combinations that produce the highest reactivity."

a) **WCAP-16498-P, "17x17 Next Generation Fuel (17x17 NGF) Reference Core Report," (Reference 9) has not yet been reviewed and approved by the NRC staff. Therefore NGF should not be considered as part of the NRC staff's review of this LAR. What are the licensee's plans to address the final approved version of WCAP-16498-P affects on this analysis?**

b) **Provide a description and results of the simulations used to determine the fuel assembly combinations that produce the highest reactivity.**

Response:

a) The MP3 SPU LAR submittal (Attachment 5 section 2.8.1.2) references the use of RFA/RFA-2 design fuel for the SPU. Similarly the NRC SER addresses the use of RFA/RFA-2 design fuel in the SPU. NRC review for this LAR was not requested for NGF fuel for the SPU. WCAP-16721-P included NGF fuel for completeness since the NGF fuel parameters of interest for the SPU criticality analysis are the same as RFA/RFA-2 fuel. DNC therefore concurs that NGF fuel need not be considered for the SPU.

b) The analyses provided in WCAP-16721-P conservatively neglect the presence of spacer grids in the spent fuel assemblies. For more details on this treatment, see the response to question 6c. The important neutronic parameters that remain in the model are fuel rod size and pitch, and are identical for all designs presented. Small variations exist among the different assembly types in the guide tube and instrument tube diameters and thicknesses. Since these tubes are made of cladding material, they are nearly transparent to neutrons and the tubes do not occupy enough volume to significantly affect neutron moderation. A series of calculations is presented in the table below which demonstrates that the reactivity of the "Standard" and "V5H" assembly types are statistically insignificantly different in Regions I and II. The "RFA" fuel type is slightly less reactive than the other two. In Region III the reactivity of all three fuel assembly types are statistically identical. The "Standard" fuel assembly was used for all analyses in WCAP-16721-P. It should be noted that the NGF product has the same dimensions as the RFA product and therefore, strictly from a criticality safety perspective, is conservatively bounded by the use of the "Standard" assembly type.

Comparison of Calculated k_{eff} Values for Various Assembly Designs in All Regions

Enrichment	3.72 w/o ^{235}U		1.81 w/o ^{235}U		1.47 w/o ^{235}U	
	Region I		Region II		Region III	
Assembly	k_{eff}	σ	k_{eff}	σ	k_{eff}	σ
Standard	0.92613	0.00043	0.93325	0.00036	0.92158	0.00031
V5H	0.92624	0.00042	0.93373	0.00031	0.92137	0.00030
RFA	0.92503	0.00042	0.93279	0.00034	0.92142	0.00034

Question 8

WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "Applicable biases factored into this evaluation are: 1) the methodology bias deduced from the validation analyses of pertinent critical experiments; and 2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 32°F to 160°F." However, WCAP-16721 Section 2.1.1, The SCALE Code, states, "The SCALE version 4.4 version of the 238-group ENDF/B-V neutron cross section library is also utilized in this analysis. However, this library is only utilized for off-nominal temperature simulations (greater than 68 °F). The 238-group library is a general purpose library that is applicable at all temperatures. The 44-group library was collapsed using a representative spectrum from a 17x17 PWR assembly at 68°F, so any deviations from these conditions should be considered as potentially moving outside the basis of applicability for this specialized library. In addition, these calculations are only considered in a relative sense, to establish the reactivity changes due to temperature deviations. Since there is no need to quantify the absolute magnitude of the reactivity at these conditions, a comprehensive validation analysis is not performed for the 238-group neutron cross section library."

a) **Provide a description of how the effect of temperatures below 68°F was evaluated.**
b) **Provide a description and results of the simulations used to determine the presence or absence of a temperature bias.**

Response:

Part a)

The operating temperature range for the MPS3 spent fuel pool ranges up to 160 °F. The nominal condition modeled is at 68 °F with a water density of 1 g/cm^3. The purpose of the temperature bias is to capture reactivity impacts caused by any allowed variation in temperature and therefore moderation. Explicit cases are run at 50 °F and the water density at 14.7 psi at that temperature (1.00 g/cm^3). These cases are of particular importance when a decreasing reactivity trend is noted with increasing temperature. The calculated k_{eff} at 50 °F and 1.00 g/cm^3 is compared to the base case of 68 °F and 1.00 g/cm^3. In response to this RAI, the temperature range was expanded to 32 °F to ensure that the entire range of possible spent fuel pool temperatures has been explicitly considered.

In Regions I and II, the change in reactivity is less than the root-sum-squared uncertainty from the 50 °F and 68 °F cases, so the difference is statistically insignificant and can therefore be neglected. Further justification of this definition of statistical significance is provided in response to RAI 9.

The expanded temperature range causes small but statistically significant reactivity increases when considering the 32 °F case. These cases are included in the tables below. The worst case temperature bias, including the one sigma Monte Carlo uncertainties from each case, is 0.00080 Δk_{eff} in Region I and 0.00075 Δk_{eff} in Region II. As discussed in the response to RAI 6 part e, the radial leakage in the spent fuel pool is sufficient to account for this bias. The available margin in Region I is 0.00188 Δk_{eff} and in Region II is 0.00163 Δk_{eff}.

In Region III, calculations are performed below 68 °F, but are not needed because of the positive reactivity trend with increasing temperature.

Further discussion of the overall temperature bias methodology is provided in the response to part b.

Part b)

The temperature bias considered in WCAP-16721-P is intended to add any reactivity increase that can be caused by water temperature and density variations within the normal operating temperature range in the MPS3 spent fuel pool. The inclusion of a temperature bias for a region is contingent upon demonstrating that such a reactivity increase can occur. The nominal case for all models in the WCAP-16721-P analysis is at 68 °F and 1.00 g/cm^3 density water.

In each region a series of calculations is performed assuming nominal fuel assembly and storage rack dimensions while varying the bulk water temperature and appropriately adjusting the water density. This series of calculations covers the entire operating range from 50 °F to 160 °F, and was expanded in Part a) to 32 °F. All fuel assemblies are assumed to be depleted to just above the minimum burnup requirement for 5 w/o ^{235}U fuel in each region as the harder neutron spectrum of the depleted fuel will be more sensitive to the moderation changes. A range of enrichments were considered in each region.

The response of regions I and II to a moderation change is similar because both feature fixed BORAL neutron poison panels. The efficacy of these fixed neutron poisons is reduced by the decreased moderation at higher temperatures, and thus lower water densities. The tables provided below show a decreasing trend of reactivity with temperature. The 50 °F case does not present a statistically significant increase in reactivity for any enrichment in either Region I or Region II, although as discussed in the response to Part a) a small positive bias is noted at 32 °F.

The response of region III to a moderation change is markedly different because of the (assumed) lack of a fixed neutron poison and a significantly larger storage cell pitch. In this case, the storage configuration is over moderated and thus moderation decreases caused by reduced water density at elevated pool temperatures create a positive reactivity effect. The table below provides the calculated k_{eff} values evaluated to quantify this effect. The temperature which corresponds to the lowest and highest reactivities is used in this determination. It should be noted that this condition is for 4.05 w/o ^{235}U fuel depleted to 55,000 MWd/MTU of burnup. The 4.05 w/o ^{235}U case was selected as it was the most positive of the three. The limiting values from 3.05 w/o ^{235}U and 5.05 w/o ^{235}U are also provided for completeness. It should also be noted that the Monte Carlo computational uncertainties are treated conservatively as with individual uncertainty components to further increase the conservatism of this bias, as shown in the equation below.

$$Temperature\ Bias = k_{eff}^{160} - k_{eff}^{\min} + \sigma^{160} + \sigma^{\min}$$

Where,

k_{eff}^{160} is the calculated k_{eff} value at 160 °F,

$k_{eff}^{\min}$ is the calculated k_{eff} value for minimum reactivity case, either 50 °F or 68 °F and 1.00 g/cm^3 H_2O density,

σ^{160} is the Monte Carlo uncertainty associated with the 160 °F case,

σ^{min} is the Monte Carlo uncertainty associated with the minimum reactivity case, either 50 °F or 68 °F and 1.00 g/cm^3 H_2O density.

Temperature Effect Data for Region I

Temp (°F)	Enrichment (w/o ^{235}U)	k_{eff}	σ	BE[1] Δk_{eff}	RSS σ	ρ H_2O (g/cm^3)
68	4.0	0.77708	0.00016			1.00
32	4.0	0.77753	0.00019	0.00045	0.00025	1.00
50	4.0	0.77708	0.00035	0.00000	0.00038	1.00
68	4.0	0.77653	0.00034	-0.00055	0.00038	0.99832
80	4.0	0.77629	0.00036	-0.00079	0.00039	0.99672
100	4.0	0.77436	0.00033	-0.00272	0.00037	0.99313
120	4.0	0.77295	0.00033	-0.00413	0.00037	0.98857
140	4.0	0.77066	0.00034	-0.00642	0.00038	0.98318
160	4.0	0.76887	0.00034	-0.00821	0.00038	0.97705
68	5.0	0.82301	0.00018			1.00
32	5.0	0.82341	0.00019	0.00040	0.00026	1.00
50	5.0	0.82261	0.00037	-0.00040	0.00041	1.00
68	5.0	0.82297	0.00038	-0.00004	0.00042	0.99832
80	5.0	0.82227	0.00036	-0.00074	0.00040	0.99672
100	5.0	0.82097	0.00034	-0.00204	0.00038	0.99313
120	5.0	0.81970	0.00040	-0.00331	0.00044	0.98857
140	5.0	0.81786	0.00034	-0.00515	0.00038	0.98318
160	5.0	0.81645	0.00035	-0.00656	0.00039	0.97705

[1] The best estimate Δk_{eff} presented in the table is the difference between calculated k_{eff} values without the inclusion of the Monte Carlo uncertainties.

Temperature Effect Data for Region II

Temp (°F)	Enrichment (w/o ^{235}U)	k_{eff}	σ	BE[2] Δk_{eff}	RSS σ	ρ H_2O (g/cm^3)
68	3.0	0.76184	0.00013			1.00
32	3.0	0.76220	0.00013	0.00036	0.00018	1.00
50	3.0	0.76187	0.00014	0.00003	0.00019	1.00
68	3.0	0.76192	0.00013	0.00008	0.00018	0.99832
80	3.0	0.76157	0.00013	-0.00027	0.00018	0.99672
100	3.0	0.76114	0.00014	-0.00070	0.00019	0.99313
120	3.0	0.76083	0.00014	-0.00101	0.00019	0.98857
140	3.0	0.76035	0.00014	-0.00149	0.00019	0.98318
160	3.0	0.75989	0.00013	-0.00195	0.00018	0.97705
68	4.0	0.82993	0.00014			1.00
32	4.0	0.83026	0.00014	0.00033	0.00020	1.00
50	4.0	0.82969	0.00014	-0.00024	0.00020	1.00
68	4.0	0.82965	0.00014	-0.00028	0.00020	0.99832
80	4.0	0.82934	0.00014	-0.00059	0.00020	0.99672
100	4.0	0.82876	0.00013	-0.00117	0.00019	0.99313
120	4.0	0.82761	0.00014	-0.00232	0.00020	0.98857
140	4.0	0.82692	0.00014	-0.00301	0.00020	0.98318
160	4.0	0.82597	0.00013	-0.00396	0.00019	0.97705
68	5.0	0.89305	0.00014			1.00
32	5.0	0.89352	0.00014	0.00047	0.00020	1.00
50	5.0	0.89298	0.00014	-0.00007	0.00020	1.00
68	5.0	0.89287	0.00015	-0.00018	0.00021	0.99832
80	5.0	0.89217	0.00014	-0.00088	0.00020	0.99672
100	5.0	0.89081	0.00015	-0.00224	0.00021	0.99313
120	5.0	0.88964	0.00014	-0.00341	0.00020	0.98857
140	5.0	0.88832	0.00015	-0.00473	0.00021	0.98318
160	5.0	0.88701	0.00015	-0.00604	0.00021	0.97705

[2] The best estimate Δk_{eff} presented in the table is the difference between calculated k_{eff} values without the inclusion of the Monte Carlo uncertainties.

Temperature Bias Data for Region III

Temp (°F)	Enrichment (w/o ^{235}U)	k_{eff}	σ	Temperature Bias
50	3.05	0.74875	0.00024	[redacted]
160	3.05	0.75889	0.00025	0.01063
68	4.05	0.80827	0.00027	[redacted]
160	4.05	0.82025	0.00024	0.01249
68	5.05	0.87048	0.00029	[redacted]
160	5.05	0.88042	0.00028	0.01051

Question 9

WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "For item c., the fuel rod manufacturing tolerance for the reference design fuel assembly consists of the following components; an increase in pellet diameter [.....], a decrease in fuel cladding thickness [.....] and an increase in fuel enrichment of [.....]." However, neither the Region I 3-out-of-4 nor the Region I 4-out-of-4 storage configurations include the pellet diameter as an uncertainty. Provide a quantitative justification for why the pellet diameter was not included in those uncertainty determinations.

Response:

The fuel pellet diameter tolerance was considered for Region I in both the "3-Out-of-4" and "4-Out-of-4" loading configurations, but its effect was not included in WCAP-16721-P because it is statistically insignificant. The results of the fuel pellet diameter tolerance are provided below. These results indicate that for both storage configurations, the best estimate reactivity impact (BE Δk_{eff}) is negative. This tolerance is only considered in the positive direction as the only possible positive reactivity impact caused by changing the pellet diameter is an increase in the amount of fissile material.

In this context, the minimum level of statistical significance is the root-sum-square (RSS) of the Monte Carlo uncertainties - this significance is appropriate as it represents the uncertainty on the difference of the two calculated k_{eff} values. As detailed in Case 3 of Section 6.2 of Reference 1, this uncertainty is statistically insignificant and therefore neglected in the final determination of uncertainties.

<table>
<tr><td>Condition</td><td>k_{eff}</td><td>σ</td><td>BE Δk_{eff}</td><td>RSS σ</td></tr>
<tr><td colspan="5">3-Out-of-4</td></tr>
<tr><td>Nominal</td><td>0.91880</td><td>0.00042</td><td rowspan="2">-0.00036</td><td rowspan="2">0.00052</td></tr>
<tr><td>Max Pellet</td><td>0.91844</td><td>0.00030</td></tr>
<tr><td colspan="5">4-Out-of-4</td></tr>
<tr><td>Nominal</td><td>0.92613</td><td>0.00043</td><td rowspan="2">-0.00040</td><td rowspan="2">0.00061</td></tr>
<tr><td>Max Pellet</td><td>0.92573</td><td>0.00043</td></tr>
</table>

Question 10

WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "For item d., the following component tolerances are varied to their outer bounds: the stainless steel canister inner dimension and thickness, the storage cell center-to-center spacing, the BORAL poison loading and the Boraflex channel wrapper thickness."

a) **Provide a quantitative justification for why uncertainties for the BORAL panel width, thickness and wrapper material thickness are not included the Region I and Region II storage configurations.**
b) **Provide a quantitative justification for why uncertainties for the "...stainless steel canister inner dimension and thickness, the storage cell center-to-center spacing...," are not included for Region II.**
c) **Provide a quantitative justification for why an uncertainty for the ".....stainless steel canister inner dimension ....," is not included for Region III.**

Response:

Part a)

The BORAL panels in the Region I and Region II models have a width of 7.425 inches. This width is the minimum allowed for the panels, as shown in Tables 3-1 and 3-2 in WCAP-16721-P. The potential manufacturing variation is bounded and included in the models as a bias, so no uncertainty is required.

The exact thickness and thickness tolerance of the BORAL panel were not considered in the analysis documented in WCAP-16721-P largely because they are not controlling dimensions for the poison loading. The areal density of ^{10}B is specified and controlled. An uncertainty is included in both Region I and Region II (see "Decrease in BORAL Loading" in Tables 4-1, 4-2, and 4-3) to account for the minimum areal density as specified in Tables 3-1 and 3-2 in WCAP-16721-P. A panel with less thickness would have a larger volume fraction of B_4C to produce the required areal density.

One case was run in the "Region I 4-out-of-4" storage configuration to investigate the impact of modeling a thinner BORAL panel. The panel was reduced in thickness from 0.107 inches to approximately 0.096 inches, while maintaining the areal density. The panel was centered in the BORAL channel with water modeled in the gaps between the panel and the storage cell wall and the wrapper. The results, provided in the table below, demonstrate that the modeling technique used in the base model reported in WCAP-16721-P is conservative. Note that the BORAL panel can not be increased in thickness due to resulting interference with the wrapper material.

No tolerance has been provided for the thickness of the wrapper material. An estimate of the tolerance on the wrapper thickness is ±0.005 inches. This estimate is determined by conservatively increasing the Region III wrapper thickness tolerance by 0.001 inches. Calculations were performed in Regions I and II and demonstrate that this tolerance does not cause a statistically significant increase in reactivity. The results are provided in the table below.

Results of Reduced Thickness BORAL Panel Case

Case	k_{eff}	σ	Δk_{eff}	RSS σ
Base	0.92613	0.00043		
Reduced Panel Thickness	0.92440	0.00043	-0.00173	0.00061

Results for Wrapper Thickness Sensitivity in Regions I and II

Case	k_{eff}	σ
Region I Base	0.92613	0.00043
Region I Increased Wrapper	0.92665	0.00040
Region I Decreased Wrapper	0.92517	0.00038
Region II Base	0.93059	0.00033
Region II Increased Wrapper	0.93042	0.00035
Region II Decreased Wrapper	0.93050	0.00033

<u>Parts b and c)</u>

As part of the original assessment of uncertainties in the MPS3 spent fuel pool, calculations were performed to assess the impact of tolerances on the storage cell inner dimension and wall thickness and the storage location pitch in Region II and the storage cell inner dimension in Region III. The results of these calculations determined that no positive reactivity impact was attributable to these three parameters. The parameters were varied in both directions in all four cases. The results of these calculations are provided in the tables below, along with the base case for comparison.

Tolerance Calculation Results for Region II

Case	k_{eff}	σ
Base	0.93059	0.00033
Minimum Inner Dimension	0.93055	0.00032
Maximum Inner Dimension	0.92958	0.00031
Minimum Wall Thickness	0.93059	0.00032
Maximum Wall Thickness	0.93009	0.00031
Minimum Pitch	0.93040	0.00033
Maximum Pitch	0.92980	0.00032

Tolerance Calculation Results for Region III

Case	k_{eff}	σ
Base	0.92158	0.00031
Minimum Inner Dimension	0.92158	0.00030
Maximum Inner Dimension	0.92119	0.00030

Question 11

WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "In the case of the tolerance due to positioning of the fuel assembly in the storage cells (item e.), all nominal calculations are carried out with fuel assemblies centered in the storage cells. Simulations are run to investigate the effect of off-center position of the fuel assemblies for each of the fuel assembly storage configurations. These cases positioned the assemblies as close as possible in four adjacent storage cells and at intermediate positions in between. For the Region I and Region II racks containing BORAL, the centered or intermediate positions may yield the highest reactivity due to the proximity of the BORAL fixed neutron poison panels and any increased reactivity at intermediate positions is accounted for as an uncertainty. The highest reactivity condition for the Region III racks is determined to occur when the assemblies are as close as allowable within the storage cells." However, the details of those simulations are never discussed and only Region III has an uncertainty attributed to Off-Center Assembly Positioning.

a) **Provide a description and results of the simulations used to determine the fuel assembly off-center combinations that produce the highest reactivity for each storage configuration.**

Response:

In all four storage configurations analyzed in WCAP-16721-P, a series of calculations were performed to determine the effects of assembly off-center positioning. In a series of steps, the assembly or assemblies in the model are incrementally moved toward the corner of the storage cell in such a manner as to bring 4 assemblies together. For models comprised of a single storage cell, the reflective boundary conditions mimic the effect of 4 assemblies coming closer together. These calculations are performed over a range of positions to investigate the reactivity of the assemblies while centered, moved to the boundary of the storage cell, and several intermediate locations. The point at which the fuel assembly array impacts the adjacent storage cell wall or poison wrapper is the point of maximum displacement. The intermediate position cases are especially important in the Region I and II racks with fixed poisons attached to the cell walls. As the assemblies move away from the center of the storage cell they move closer together, increasing reactivity. This affect is countered by the fact that the assemblies are moving closer to the fixed poisons. This is demonstrated by the fact that the off-center positioning uncertainty in Region II is caused by a case with less than half of the total possible eccentricity. The off-center positioning uncertainty is then calculated in the same manner as other uncertainties - including the Monte Carlo uncertainty from each calculation. This uncertainty term is neglected only if the reactivity trend is clearly decreasing with increasing assembly displacement.

A table is provided below containing all the off-center position cases considered for each configuration. The values that were used in WCAP-16721-P are highlighted for convenience. It should be noted that both Regions II and III include an uncertainty for off-center assembly positioning.

Off-Center Positioning Cases for the "Region I 3-out-of-4" Storage Configuration

Displacement (cm)	k_{eff}	σ
0	0.92091	0.00044
0.2	0.92075	0.00017
0.3	0.92074	0.00017
0.4	0.91966	0.00017
0.467	0.91978	0.00017

Off-Center Positioning Cases for the "Region I 4-out-of-4" Storage Configuration

Displacement (cm)	k_{eff}	σ
0	0.92613	0.00043
0.05	0.92596	0.00016
0.10	0.92553	0.00016
0.15	0.92575	0.00016
0.20	0.92549	0.00016
0.25	0.92569	0.00017
0.30	0.92555	0.00016
0.35	0.92522	0.00017
0.40	0.92499	0.00016
0.467	0:92466	0.00017

Off-Center Positioning Cases for Region II

Displacement (cm)	k_{eff}	σ
0	0.93059	0.00033
0.10	0.93094	0.00032
0.20	0.93148	0.00033
0.30	0.92973	0.00030
0.40	0.92998	0.00032
0.467	0.93080	0.00031

Off-Center Positioning Cases for Region III

Displacement (cm)	k_{eff}	σ
0	0.92158	0.00031
0.10	0.92212	0.00031
0.20	0.92228	0.00029
0.30	0.92277	0.00030
0.403	0.92385	0.00028

Question 12

WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "For item f., a depletion determination uncertainty [...] is included in the final summation of biases and uncertainties. Since the depletion determination uncertainty is dependent on the magnitude of the burnup credited in the analysis, it is determined iteratively at each initial enrichment considered in a storage configuration." This method of determining the burnup uncertainty is a deviation from NRC staff guidance contained in Reference 7. The NRC staff estimates that the method used to determine the burnup uncertainty may result in a smaller uncertainty than the method contained in the NRC staff guidance. However, as this analysis actually treats the burnup uncertainty as a bias, the alternate method may be acceptable. To make that determination, the NRC staff requests the following information:

a) **The basis for the WCAP-16721-P method for determining the burnup uncertainty.**
b) **The zero (0) burnup reactivity for each initial enrichment considered in a storage configuration.**

Response:

Part a)

The uncertainty of 1.0% Δk_{eff} per 30,000 MWd/MTU burnup is based on a comparison of isotopics predictions to measured quantities.

Measured isotopic number densities are available from Yankee Core 5. This core was fueled with Zircaloy-clad UO_2 fuel, so it is considered an applicable data set to support MPS3 conclusions. These data are also used in the qualification of the PHOENIX-P/ANC system, as documented in Reference 3. A comparison of measured to predicted isotopic number densities for several key isotopes is presented in the table below.

PHOENIX-P's ability to predict isotopics must be converted into terms of reactivity (specifically, Δk_{eff}) for the purpose of a reactivity uncertainty, so these differences are applied to the isotopic predictions in the original analysis from WCAP-16721-P. A series of calculations was performed in Regions II and III to investigate the reactivity impact of the isotopic perturbations listed below. In Region II the perturbations are based on 5.0 w/o fuel with 45,000 MWd/MTU burnup. In Region III the calculations are based on 5.0 w/o fuel with 55,000 MWd/MTU burnup. These steps were selected as they are above or near the required minimum burnup in these regions. The 4-zone distributed burnup model is used for both sets of calculations.

Two different perturbations are performed in each region. The first perturbation, termed "conservative", modifies the number densities of each listed isotope in the conservative direction (increased for isotopes with odd atomic weights, decreased for even atomic weights); the second perturbation, termed "as-measured", modifies the number densities in the direction indicated by the table. The results are provided in tables below, along with the depletion uncertainty that was included in each region for comparison.

These results demonstrate that in all cases the WCAP-16721-P uncertainty is significantly larger (~0.01 Δk_{eff}) than that based on the conservative treatment of differences between measured and predicted isotopic number densities. Therefore, it is determined that the depletion uncertainty in WCAP-16721-P is sufficient to bound the "reactivity uncertainty due to uncertainty in fuel depletion" that is described in Reference 2.

PHOENIX-P Isotopic Number Density Predictions a,c

Results of Perturbation Calculations in Region II

Case	k_{eff}	σ	Uncertainty Value
Base	0.91198	0.00034	
Conservative	0.91613	0.00032	0.00478
As-Measured	0.91174	0.00032	< 0
WCAP Value			0.01411

Results of Perturbation Calculations in Region III

Case	k_{eff}	σ	Uncertainty Value
Base	0.90713	0.00030	
Conservative	0.91227	0.00029	0.00573
As-Measured	0.90740	0.00030	0.00087
WCAP Value			0.01852

Part b)

The calculated k_{eff} values for the zero burnup enrichment case for each storage configuration in which burnup is credited is provided in the tables below.

Zero Burnup k_{eff} Values for "Region I 4-out-of-4" Storage Configuration

Enrichment (^{235}U w/o)	k_{eff}	σ
3.85	0.93261	0.00043
4.00	0.93984	0.00044
5.00	0.97967	0.00043

Zero Burnup k_{eff} Values for Region II

Enrichment (^{235}U w/o)	k_{eff}	σ
1.81	0.93188	0.00036
3.00	1.07873	0.00036
4.00	1.14984	0.00038
5.00	1.20038	0.00040

Zero Burnup k_{eff} Values for Region III

Enrichment (^{235}U w/o)	k_{eff}	σ
1.45	0.91809	0.00027
3.00	1.13375	0.00033
4.00	1.20235	0.00033
5.00	1.25023	0.00036

Question 13

It is unclear from WCAP-16721-P Table 4-5, Table 4-6, and Table 4-7, whether or not the Enrichment Uncertainty contains the KENO V.a case uncertainties for those storage configurations as it does for the "Region I 3-out-of-4" storage configuration as indicated by Table 4-1. Confirm that the KENO V.a case uncertainties are included in all Enrichment Uncertainties.

Response:

The KENO V.a case uncertainties are included in all enrichment tolerance determinations consistent with other reactivity tolerance determinations. That is, the enrichment uncertainty is determined according to the following equation:

$$\Delta k_{eff} = (k_{unc} + \sigma_{unc,\ mc}) - (k_{nom} - \sigma_{nom,\ mc})$$

Where,

Δk_{eff} is the magnitude of the enrichment uncertainty,
k_{unc} is the multiplication factor of the perturbed system,
$\sigma_{unc,\ mc}$ is the monte carlo standard deviation from the perturbed calculation,
k_{nom} is the multiplication factor of the nominal system,
$\sigma_{nom,\ mc}$ is the monte carlo standard deviation from the nominal calculation.

To further illustrate this point, the tables below provide the KENO V.a $k_{eff} \pm \sigma$ values used to determine the enrichment uncertainties provided in WCAP-16721-P Tables 4-5 through 4-7. As the use of the above equation yields enrichment uncertainties identical to WCAP-16721-P, the tables demonstrate that in each instance, a conservative practice of including the stochastic uncertainty from each case in the enrichment uncertainty was performed.

Region I (WCAP-16721-P Table 4-5)

Enrichment (^{235}U w/o)	k_{eff}	σ	Enrichment Uncertainty (Δk_{eff})
3.85	0.93261	0.00043	
3.90	0.93550	0.00040	0.00372
4.00	0.93984	0.00044	
4.05	0.94194	0.00040	0.00294
5.00	0.97967	0.00043	
5.05	0.98138	0.00045	0.00259

Region II (WCAP-16721-P Table 4-6)

Enrichment (^{235}U w/o)	k_{eff}	σ	Enrichment Uncertainty (Δk_{eff})
1.80	0.93188	0.00036	
1.85	0.94071	0.00035	0.00954
3.00	1.07873	0.00036	
3.05	1.08311	0.00034	0.00508
4.00	1.14984	0.00038	
4.05	1.15278	0.00038	0.00370
5.00	1.20038	0.00040	
5.05	1.20257	0.00040	0.00299

Region III (WCAP-16721-P Table 4-7)

Enrichment (^{235}U w/o)	k_{eff}	σ	Enrichment Uncertainty (Δk_{eff})
1.45	0.91809	0.00027	
1.50	0.92913	0.00027	0.01158
3.00	1.13375	0.00033	
3.05	1.13785	0.00037	0.00480
4.00	1.20235	0.00033	
4.05	1.20605	0.00035	0.00438
5.00	1.25023	0.00036	
5.05	1.25289	0.00032	0.00334

Question 14

The Depletion Uncertainty for 3.0 without Initial Enrichment in WCAP-16721-P Table 4-6, "Region II" Storage Configuration Total Biases and Uncertainties Results, appears to be incorrect. Confirm the actual depletion uncertainty for 3.0 weight percent initial enrichment for Region II.

Response:

The depletion uncertainty for 3.0 w/o initial enrichment listed in Table 4-6 of WCAP-16721-P is 0.00910 Δk_{eff} units. The burnup limit for 3.0 w/o initial enrichment for Region II is 16,891 MWd/MTU. If a depletion uncertainty of 0.01 Δk_{eff} per 30,000 MWd/MTU is considered, the depletion uncertainty would be determined to be 0.00563 Δk_{eff} units. The value reported in Table 4-6 and used for determination of the burnup limits in Region II for 3.0 w/o provides additional conservatism of 0.00347 Δk_{eff} units. If the less conservative value had been included in the determination of the sum of biases and uncertainties, the target k_{eff} would have increased 0.00175 Δk_{eff} units. No credit is taken for this conservative treatment of the depletion uncertainty.

Question 15

The Depletion Uncertainty for 3.0 without Initial Enrichment in WCAP-16721-P Table 4-7, "Region III" Storage Configuration Total Biases and Uncertainties Results, appears to be incorrect. Confirm the actual depletion uncertainty for 3.0 weight percent initial enrichment for Region III.

Response:

The depletion uncertainty for 3.0 w/o initial enrichment listed in Table 4-7 of WCAP-16721-P is 0.01341 Δk_{eff} units. The burnup limit for 3.0 w/o initial enrichment for Region III is 25,516 MWd/MTU. If a depletion uncertainty of 0.01 Δk_{eff} per 30,000 MWd/MTU is considered, the depletion uncertainty would be determined to be 0.00851 Δk_{eff} units. The value reported in Table 4-7 and used for determination of the burnup limits in Region III for 3.0 w/o provides additional conservatism of 0.00490 Δk_{eff} units. If the less conservative value had been included in the determination of the sum of biases and uncertainties, the target k_{eff} would have increased 0.00300 Δk_{eff} units . No credit is taken for this conservative treatment of the depletion uncertainty.

Question 16

The keff values in WCAP-16721-P Table 4-8, Table 4-9, and Table 4-10 storage configurations all have KENO V.a case uncertainties associated with them. However, those KENO V.a case uncertainties are not used in determining the Burnup Uncertainty or the burnup necessary to meet the target k_{eff} values. This is non-conservative. Therefore, since the burnup necessary to meet the target k_{eff} values is used directly in establishing the burnup/enrichment relationships in Section 5, this non-conservatism acts as a bias. Given the small amount of reserved analytical margin, the licensee should identify offsetting conservatism in the analysis.

Response:

The methodology uncertainty of 0.00709 Δk_{eff} is included in the sum of biases and uncertainties and includes a term which accounts for the uncertainty from each Monte Carlo calculation. The formulation for this uncertainty is:

$$\sigma_{meth} = M\sqrt{(\sigma_m)^2 + (\max(\sigma_{KENO}))^2}$$

Where:

- σ_{meth} is the methodology uncertainty,
- M is the one-sided 95/95 multiplier based on the degrees of freedom from the benchmark validation, 2.22 for the experimental benchmarks considered in WCAP-16721-P,
- σ_m is the mean calculational variance from the benchmark validation, 0.00316 for the experimental benchmarks considered in WCAP-16721-P,
- $max(\sigma_{KENO})$ is the maximum individual KENO case uncertainty.

The value used for σ_{KENO} in all three regions in the MPS3 spent fuel pool is 0.00045. This is the highest value from all individual case uncertainties. The inclusion of the Monte Carlo uncertainty term in the methodology uncertainty explicitly includes its effect in the sum of biases and uncertainties, and therefore the determination of the target k_{eff}. After σ_{KENO} is root-sum-squared with σ_m, the combined uncertainty is increased by the one-sided 95/95 multiplier. Therefore, the target k_{eff} is decreased sufficiently to accommodate the uncertainty in the Monte Carlo results; any nominal KENO-calculated k_{eff} that is less than the target k_{eff} is acceptable for storage.

Any increase in the sum of uncertainties attributable to an uncertainty in the Monte Carlo calculations can be determined. As discussed above, the highest Monte Carlo uncertainty in any of the calculations supporting the analysis reported in WCAP-16721-P is 0.00045 Δk_{eff}. A review of the data presented in the response to RAI 6 parts a and b shows that the reactivity of the fuel assembly decreases by 0.000003 to 0.000008 Δk_{eff} per MWd/MTU. The increase in the burnup limit to account for the additional 0.00045 Δk_{eff} can then be obtained by dividing the uncertainty by the rate of reactivity change. The resulting increase in the burnup limit ranges from 57 to 144 MWd/MTU for various configurations. If this range is then conservatively rounded to 200 MWd/MTU, the corresponding increase in the depletion uncertainty is 0.00007 Δk_{eff}. Conservatism identified in response to RAI 6 part e and the 0.00100 Δk_{eff} of administrative margin are both sufficient to account for this small reactivity impact. While such a reactivity uncertainty is included in the original analysis, it has been quantified here to demonstrate its insignificance.

Question 17

WCAP-16721 Section 4.5.2, Soluble Boron Required to Mitigate Postulated Accident Effects, does not consider the misloading of a 5.0 weight percent ^{235}U fresh fuel assembly into location required to be empty in the Region I 3-out-of-4 storage configuration. Provide a justification for not including this accident scenario; include a description of the blocking device used and the controls which govern it.

Response:

The reactivity of the Region I 3-out-of-4 storage configuration is significantly less than the other configurations considered in the MPS3 spent fuel pool because of the presence of fixed poisons and empty storage locations in the storage array. It was therefore believed to be unlikely that a misloading accident would create a limiting postulated accident scenario. To confirm this assumption, a postulated accident scenario was considered in the Region I 3-out-of-4 storage configuration. A non-accident full pool model including Region I 3-out-of-4 storage was created. This model includes the TS limiting loading of fresh 5.0 w/o fuel assemblies in all allowed storage locations in the 3-out-of-4 storage configuration. A fresh 5.0 w/o assembly was misloaded into a storage location in this configuration which should have been empty. The postulated misloading accident Δk_{eff} and the required soluble boron concentration to mitigate the accident are provided in the table below. The postulated accident Δk_{eff} and soluble boron concentration are both bounded by the results presented in Table 4-12 of WCAP-16721-P.

Postulated Accident Scenario in Region I 3-out-of-4 Storage Configuration

Case	k_{eff}	σ	Accident Δk_{eff}	Soluble Boron (ppm)
Base case	0.91994	0.00012		
Postulated Accident	0.93762	0.00027	0.01768	174

MPS3 Spent Fuel Pool Region I storage cells are allowed by existing TS 5.6.1.1a.(2) to be placed in a "3-out-of-4" storage configuration where fuel up to 5 w/o ^{235}U enrichment, regardless of fuel burnup, may be placed in 3 storage cells, provided the 4th storage cell is both empty of fuel and has a cell blocker.

Figure 1 attached shows a schematic representation of a cell blocker in this "3-out-of -4" storage configuration. In Figure 1, a cell blocker is sketched in both the inserted and removed position, with 4 storage cells shown. The cell blocker is made of stainless steel and provides a clear visual indication to the fuel handler that fuel should not be placed in this blocked location. The cell blocker weighs about 30 pounds and has no mechanical attachment to the racks. The use of a cell blocker provides an additional barrier, beyond administrative fuel movement controls, to ensure no fuel is stored in that location.

MPS3 existing TS 3.9.14 and associated TS Figure 3.9-2 provide the regulatory controls concerning the Region I "3-out-of-4" storage configuration to ensure compliance to the criticality analysis. These TS provide the loading plan schematic and region interface requirements. The TS also provide surveillance requirements should a cell blocker be installed or removed.

Procedural controls to implement the existing TS requirements for Region I "3-out-of-4" storage are in Millstone procedures EN 31001 and SP 31022 as described next.

Millstone procedure EN 31001 "Supplemental SNM Inventory and Control", Attachment 2, documents the five Region I spent fuel storage rack modules and the current cell blocker locations. A copy of this Attachment 2 is provided as Figure 2. As shown on Attachment 2, there are currently 35 cell blockers located in two Region I spent fuel rack modules. These two Region I storage rack modules therefore constitute the "3-out-of-4" Region I storage.

WCAP-16721-P (Figure 3-1 on page 27 of 86) shows the entire spent fuel pool layout. This arrangement of cell blockers in Region I complies with the region interface requirements shown in TS Figure 3.9-2.

Millstone procedure SP 31022, "Spent Fuel Pool Criticality Requirements" implements TS 3.9-14. SP 31022 requires that the storage pattern of Region I be verified, prior to installing or removing a blocking device. Consistent with TS 3.9.14, SP 31022 requires that all fuel be removed from the storage pattern prior to removing or installing a blocking device. A surveillance form is completed to document TS surveillance requirement 4.9.14.

The SPU has not resulted in any physical changes to the spent fuel pool racks or to the cell blocker devices. All TS concerning Region I storage racks and the cell blockers are unaltered by the SPU. The Millstone specific procedural controls relating to the cell blocking devices are also not altered by the SPU.

Figure 1 Cell Blocker from Region I "3-out-of-4" Storage Configuration

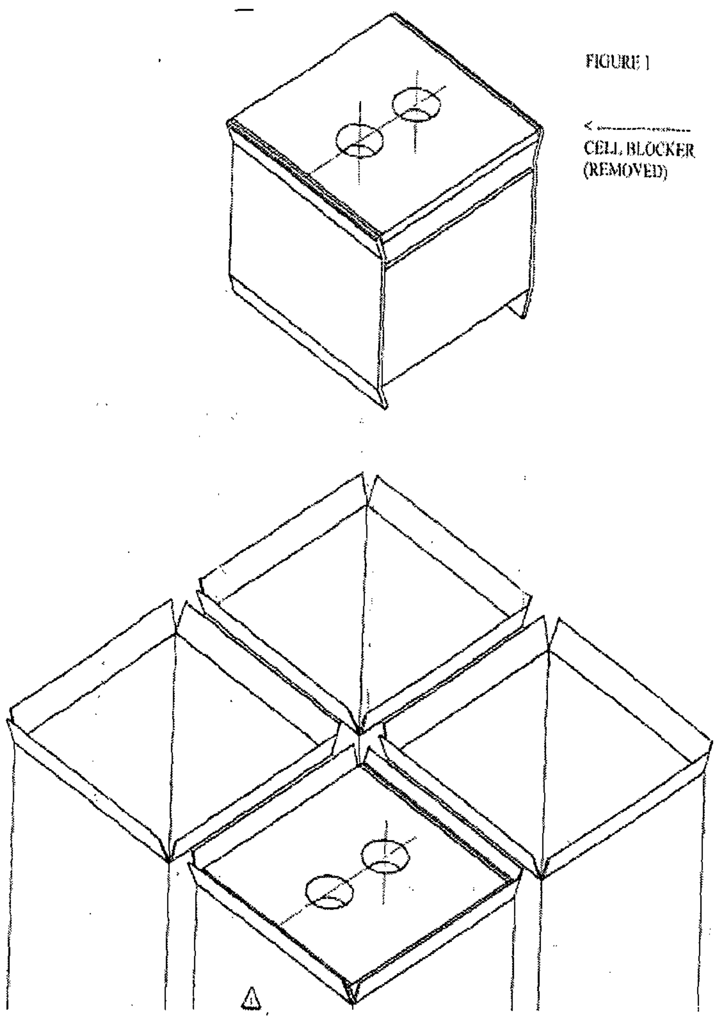
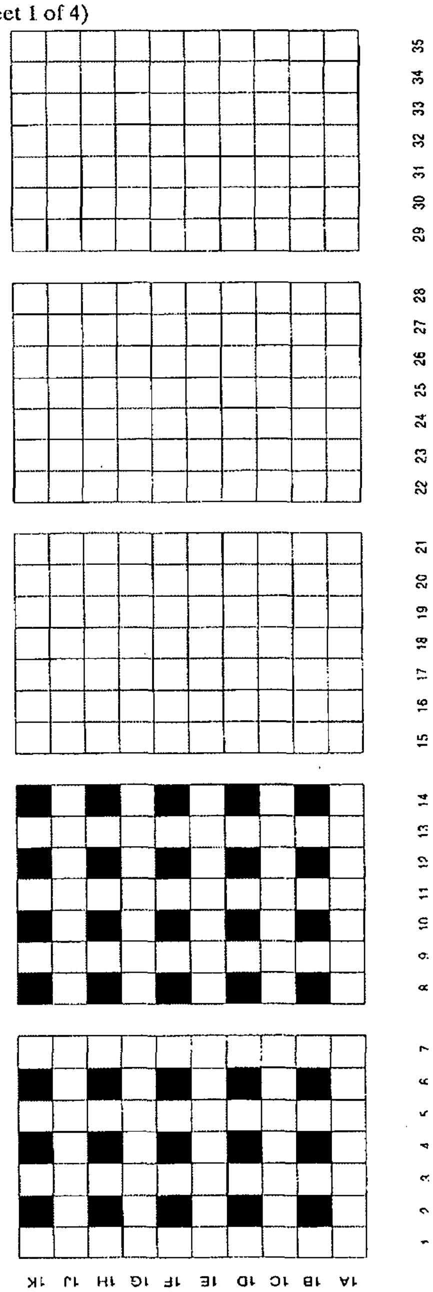


Figure 2, Procedure EN 31001, Attachment 2

Attachment 2
Spent Fuel Pool
(Sheet 1 of 4)



Signature:________________ Date:

Reviewed By______________Date:

■ Cell Blocking Device. Complete requirements in SP31022 prior to moving.

Comments

Level of Use
Reference

STOP THINK ACT REVIEW

References:

1. L.G. Blackwood, ICSBEP Guide to the Expression of Uncertainties, NEA/NSC/DOC(95)03, September 2006.
2. Laurence Kopp (USNRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998.
3. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June, 1988.
4. Parks, C. V. et.al., "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, February, 2000.
5. J.C. Wagner et. al., "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801, March, 2003.

NRC Question References (References cited in NRC questions):

1. Dominion Nuclear Connecticut, Inc., letter 07-0450, Gerald T. Bischof, Vice President Nuclear Engineering, to USNRC document control desk, re: "Dominion Nuclear Connecticut, INC., Millstone Power Station Unit 3, License Amendment Request, Stretch Power Uprate," July 13,2007. (ADAMS ML072000386)
2. Dominion Nuclear Connecticut, Inc., letter 07-0450A, Gerald T. Bischof, Vice President Nuclear Engineering, to USNRC document control desk, re: "Dominion Nuclear Connecticut, INC., Millstone Power Station Unit 3, License Amendment Request, Stretch Power Uprate - Supplemental Information," July 13, 2007. (ADAMS ML072000281)
3. Dominion Nuclear Connecticut, Inc., letter 07-0450D, Gerald T. Bischof, Vice President Nuclear Engineering, to USNRC document control desk, re: "Dominion Nuclear Connecticut, INC., Millstone Power Station Unit 3, License Amendment Request, Stretch Power Uprate - Supplemental Information," March 5, 2008. (ADAMS ML080660108).
4. U.S. NRC letter to Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, UNIT NO. 3 - Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant hazards Consideration (TAC NO. MA51 37)," dated November 28, 2000. (ADAMS ML003771974)
5. DOE/RW-0472, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages," Revision 2, September 1998.
6. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis."
7. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998. (ADAMS ML003728001)
8. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel." (ADAMS ML003688150)

9. WCAP-16498-P, "17x17 Next Generation Fuel (17x17 NGF) Reference Core Report," March 2008. (ADAMS ML081010602)

ATTACHMENT 3

WESTINGHOUSE ELECTRIC COMPANY LLC

AUTHORIZATION LETTER TO WITHHOLD INFORMATION FROM

PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

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

Direct tel: (412) 374-4643
Direct fax: (412) 374-3846
e-mail: greshaja@westinghouse.com

Our ref: CAW-08-2478

September 17, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: NEU-08-50, Revision 1, Proprietary, "Westinghouse Response to Request for Additional Information (RAI) Regarding the Spent Fuel Pool Criticality Amendment Request" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2478 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Dominion Nuclear Connecticut.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2478 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

cc: George Bacuta (NRC OWFN 12E-1)

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 17th day of September, 2008

Sharon L. Markle

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in NEU-08-50, Revision 1, "Westinghouse Response to Request for Additional Information (RAI) Regarding the Spent Fuel Pool Criticality Amendment Request," (Proprietary), for submittal to the Commission, being transmitted by Dominion Nuclear Connecticut letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Dominion Nuclear Connecticut's request for NRC approval of the Millstone Unit 3 Spent Fuel Pool Criticality Amendment.

This information is part of that which will enable Westinghouse to:

(a) Obtain NRC approval of Spent Fuel Pool Criticality Amendment.

(b) Respond to NRC Request for Additional Information in support of the Spent Fuel Pool Criticality Amendment.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of licensing amendments.

(b) Westinghouse can sell support and defense of the use of this modal for licensing purposes.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Copyright Notice

ATTACHMENT 4

NON-PROPRIETARY RESPONSE TO RAI QUESTIONS 18 & 19

FOR THE SPENT FUEL POOL LICENSE AMENDMENT REQUEST

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Spent Fuel Pool Amendment Request

Dominion Nuclear Connecticut (DNC) Response to a Request for Additional Information (RAI)

Question 18

Describe the process used to determine that fuel assemblies have attained proper burnup for storage in the burnup dependent racks.

DNC Response

Millstone Surveillance Procedure SP 31022, "Spent Fuel Pool Criticality Requirements," controls the process of ensuring that fuel assemblies have attained proper burnup for storage in the burnup-dependent fuel storage regions.

SP 31022 requires that Quality Assured (QA) calculations be performed to document that fuel assemblies meet the requirements for storage in Region I "4-out-of-4", Region II and Region III of the MPS3 spent fuel pool. Fuel burnups using measured incore power distributions are used for this determination.

First, QA calculations are performed to document the fuel burnups from measured incore power distributions, as follows:

- The Westinghouse INCORE (or future equivalent) QA computer code is used to generate measured core power distribution maps as the fuel is depleted. The use of the INCORE computer program to generate INCORE power distributions is controlled by Millstone Surveillance Procedure SP 31003, "Incore Analysis."
- The Westinghouse TOTE (or future equivalent) QA computer code is used to generate measured individual fuel assembly burnups, using the INCORE measured core power distribution maps. The resulting measured fuel assembly burnups are documented in QA calculations.

The measured fuel assembly burnup values described above are conservatively reduced by the burnup measurement uncertainty, which is currently 4%. As documented in WCAP-16721-P, section 4.2, the burnup measurement uncertainty is documented by the plant operator (Millstone). This burnup measurement uncertainty is different than the depletion measurement uncertainty, which is the uncertainty in reactivity for a given fuel depletion accounted for in WCAP-16721-P.

Once the fuel assembly burnups from measured incore power distributions are determined, additional QA calculations are performed for the determination that the fuel assembly meets the TS requirements for storage in any burnup- dependent spent fuel pool region. Using the fuel assembly initial enrichment, measured fuel burnup and fuel decay time (if required) for a given fuel assembly, the result is then checked against the TS burnup limits for each fuel storage region to determine where the fuel assembly is qualified to be stored. If the measured fuel assembly burnup is greater than that required by a region's TS burnup limit, the fuel assembly is qualified for storage in that spent fuel pool region.

When a fuel assembly is determined to be qualified for storage in a particular burnup-dependent region, the fuel assembly serial number is entered on a procedurally controlled form which lists all fuel assemblies qualified for storage in each burnup-dependent region. Separate lists are maintained for each burnup-dependent storage region, specifically:

SP 31022 Attachment 5, "Region 1 (4-out-of-4) Qualified Fuel Assemblies"
SP 31022 Attachment 6, "Region 2 Qualified Fuel Assemblies"
SP 31022 Attachment 7, "Region 3 Qualified Fuel Assemblies"

In summary, as required by plant procedure SP 31022, QA calculations are performed to document which fuel assemblies are qualified for storage in the burnup-dependent regions of the MPS3 Spent Fuel Pool. The measured fuel assembly burnups are conservatively reduced by 4% to account for burnup measurement uncertainty. The resulting combination of fuel assembly initial enrichment, current fuel decay time and measured fuel assembly burnup for each fuel assembly are compared to the TS burnup limits for each spent fuel pool storage region. If a fuel assembly is qualified for that storage region, it is placed on a list of "Qualified Fuel Assemblies" for that storage region. These lists of "Qualified Fuel Assemblies" for each storage region become the basis for allowing a fuel assembly to be moved to that storage region.

Question 19

Describe the process used to control movement of items within the SFP.

DNC Response

All fuel assembly movements are controlled as Special Nuclear Material (SNM) under the direct supervision of qualified Reactor Engineering or Operations personnel. Procedural controls and physical equipment constraints limit fuel assembly movements in the spent fuel pool to only one fuel assembly at a time.

Fuel assembly movements into and out of the spent fuel pool are controlled in accordance with Procedures MC-5, "Special Nuclear Material Inventory and Control", EN 31001, "Supplemental SNM Inventory and Control", EN 31007, "Refueling Operations" and OPS-FH-310, "MP3 Spent Fuel Pool Operations". For any fuel assembly to be moved a fuel move sheet is required. The fuel move sheet can be documented on 2 types of forms, called the "Material Transfer Form", or the "Refueling Worklist", but the controls described below apply to both these types of fuel move sheets. A fuel move sheet requires a preparer, reviewer and approver. Both the preparer and reviewer of the fuel move sheet ensure that fuel to be moved is qualified for the spent fuel pool storage region it is to be moved to. EN 31001 requires that the fuel assembly to be moved be verified to be qualified for the spent fuel pool storage region it is to be moved to, by ensuring that fuel assembly is listed on the applicable "Qualified Fuel Assemblies" list of SP 31022 Attachments 5, 6 and 7. The above response to RAI question 18 describes the development of the "Qualified Fuel Assemblies" lists in SP 31022 Attachments 5, 6 and 7. This process ensures that the fuel move sheets are correctly filled out, such that a fuel assembly is only allowed to be placed in a spent fuel pool storage region for which the fuel assembly is qualified.

When a fuel assembly is actually moved in the spent fuel pool, procedures MC-5 and EN 31001 require two personnel to verify that the fuel movement is correctly performed, as specified on the fuel move sheet. This determination that the fuel assembly is correctly moved is based on each individual verifying that the fuel assembly is correctly removed from the specified fuel storage location and correctly placed into the specified fuel storage location, as listed on the fuel move sheet. This verification is by visual identification of a fuel assembly moving out of and moving into the specified spent fuel rack locations. Each fuel storage rack location has its own unique co-ordinate value.

In addition to the above visual verification by 2 individuals that a fuel assembly is correctly moved to and from the specified storage rack locations, verification of fuel assembly serial number is also required at certain specified times. Fuel assembly serial number verification is procedurally required at the following times: (1) when loading new fuel to the spent fuel pool, (2) during core offloads, if fuel is to be off-loaded directly from the core to Region II or III storage racks, (3) after core loading, verify the serial number of fuel in the core, and (4) during the yearly spent fuel pool SNM inventory, verify the serial number of any fuel that has been moved which did not have the serial number verified at the time of the move, since the last yearly spent fuel pool SNM inventory.

The above fuel movement controls and verification process is unchanged by the SPU. While the SPU requires a change in the Region II and III TS curves for fuel qualification, this change only affects the development of the "Qualified Fuel Assemblies" lists contained in SP 31022 Attachments 6 and 7 discussed above in RAI question 18. However, the SPU changes do not affect the controls on the fuel movement process or fuel movement verification process.