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Dominion®

March 5, 2010

U. S. Nuclear Regulatory Commission		Serial No.:	10-072A
Attention: Document Control Desk		NLOS/WDC	R0
Washington, DC 20555		Docket No.:	50-423
		License No.:	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
RESPONSE TO QUESTIONS 28 AND 30 OF REQUEST FOR ADDITIONAL INFORMATION REGARDING A SPENT FUEL POOL CRITICALITY LICENSE AMENDMENT REQUEST

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

In a letter dated August 8, 2008, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the SFP TS. DNC responded to RAI questions 1 through 19 in a letter dated September 30, 2008 (Serial No. 08-0511A). In a letter dated February 2, 2009, the NRC requested additional information. DNC responded to RAI questions 20, 22, 23, and 25 in a letter dated March 5, 2009 (Serial No. 09-084) and to RAI questions 21 and 24 in a letter dated March 23, 2009 (Serial No. 09-084A). Subsequently, in a letter dated January 26, 2010, the NRC requested additional information. DNC responded to RAI questions 26, 27 and 29 in a letter dated March 1, 2010 (Serial No. 10-072).

Attachment 1 contains the responses to RAI questions 28 and 30. Attachment 2 contains an updated markup of MPS3 TS pages affected by this SFP TS change request. These updated markups supersede the TS page markups submitted as part of the SPU license amendment request in the letters dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A). Attachment 3 contains a markup of associated Bases changes for information only. Changes to TS Bases are controlled under the provisions of 10 CFR 50.59.

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 Wanda Craft at 804-273-4687.

Sincerely,

Leslie N. Hartz
Vice President – Nuclear Support Services

Commitments made in this letter:
1. 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 5TH day of March, 2010.

My Commission Expires: May 31, 2010.

Vicki L. Hull
Notary Public

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2010

Attachments:

1. Attachment 1: Response to Request for Additional Information (RAI) Questions 28 and 30 Regarding the Spent Fuel Pool Criticality License Amendment Request
2. Attachment 2: Response to Request for Additional Information (RAI) Questions 28 and 30 Regarding the Spent Fuel Pool Criticality License Amendment Request, Updated Markup of Technical Specifications Pages
3. Attachment 3: Response to Request for Additional Information (RAI) Questions 28 and 30 Regarding the Spent Fuel Pool Criticality License Amendment Request, Markup of Technical Specifications Bases Pages For Information Only

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ATTACHMENT 1

RESPONSE TO RAI QUESTIONS 28 AND 30

REGARDING THE SPENT FUEL POOL CRITICALITY LICENSE AMENDMENT REQUEST

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

RESPONSE TO RAI QUESTIONS 28 AND 30 REGARDING THE SPENT FUEL POOL CRITICALITY LICENSE AMENDMENT REQUEST

Question 28

In DNC's response to RAI #5, DNC claims that ignoring the Integrated Fuel Burnable Absorber (IFBA) is conservative. In that response DNC claims that it is conservative to ignore the presence of IFBA when performing the depletion portion of a spent nuclear fuel criticality analysis. DNC's submittal states they performed two sets of analyses; one in which all residual IFBA was artificially removed after the depletion, and one in which all residual IFBA was retained after the depletion. DNC's submittal indicates that when the residual IFBA is artificially removed the effect of neutron spectral hardening is shown, but when the residual IFBA is left in the fuel assembly, the residual IFBA overcomes the neutron spectral hardening with a conservative result.

There is no indication in NUREG/CR-6760 that any residual IFBA was artificially removed in reaching its conclusions. The information presented in NUREG/CR-6760 indicates that any residual IFBA was left in the fuel assembly when determining the effect. DNC response is inconsistent with NUREG/CR-6760 and has not been accepted by NRC staff in the past.

There is currently insufficient information in DNC's submittal to reconcile the different conclusions. Please provide a detailed comparison of DNC's analysis and the analysis performed in NUREG/CR-6760 and justify the differences.

Response:

The analysis presented in response to request for additional information (RAI) 5 was performed using site specific information for the fuel assembly design, core operating parameters, and Integral Fuel Boron Absorber (IFBA) designs used at Millstone Power Station Unit 3 (MPS3). The calculations reported in response to RAI 5 were performed with 5.0 wt% enriched fuel with IFBA modeled over the entire axial length of the fuel. The generic analysis of 17x17 fuel presented in NUREG/CR-6760, Reference 1, considered both full-length and part-length IFBA. The conclusion presented for full length IFBA in Reference 1 agrees with the conclusion presented in RAI 5 for full-length IFBA.

The following discussion identifies the apparent differences between the MPS3's analysis and the analysis performed in NUREG/CR-6760. The results presented in response to RAI 5 and additional results below are applicable to the MPS3 SFP criticality safety analysis. The results presented here are not necessarily representative of other fuel lattices, assembly designs, IFBA designs or core operating parameters outside of MPS3.

Tables 6 and 7 contain modeling information from Section 3.3 of Reference 1. A column is added to each table to show the values used in the MPS3 analysis.

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

Parameter	NUREG/CR-6760	Response to RAI 5
Moderator Temperature (°K)	600	585.65
Fuel Temperature (°K)	1000	1029.93
Fuel Density (g/cm^3)	10.44 (UO_2)	10.69 (UO_2)
Clad Temperature (°K)	600	614.93
Clad Density (g/cm^3)	5.78 (Zr)	6.58 (Zircaloy-4)
Power Density (MW/MTU)	60	41.5
Moderator Boron Concentration (ppm)	650	1000

Most of the calculations in Reference 1 were performed assuming a uniform burnup profile. The MPS3 analysis performed in response to RAI 5 used the axial power distribution shown in Figure 3-5 of WCAP-16721-P; the fuel, moderator, and cladding temperatures varied as a function of power. The values shown in Table 6 correspond to the values at a relative power of 1.0.

The footnote to Table 1 of Reference 1 states that cases were also calculated using a power density of 30 MW/MTU and that the change in effective multiplication factor (Δk) results were not sensitive to variations in power density.

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

Parameter	NUREG/CR-6760	MPS3 Analysis
Rod Pitch (cm)	1.260	1.260
Assembly Pitch (cm)	21.5	21.5
Cladding Outside Diameter (cm)	0.8898	0.950
Cladding Inside Diameter (cm)	0.8001	0.836
Pellet Outside Diameter (cm)	0.7840	0.819
Guide/Instrument Tube Outside Diameter (cm)	1.204	1.224
Guide/Instrument Tube Inside Diameter (cm)	1.124	1.143
Array Size	17x17	17x17
Number of Fuel Rods	264	264
Number of Guide/Instrument Tubes	25	25

Reference 1 examines 17x17 assemblies containing 80, 104, and 156 IFBA rods with boron loadings of 1.57 and 2.355 mg^{10}B/inch. The MPS3 analysis uses 17x17 assemblies containing 156 IFBA rods with an IFBA loading similar to the maximum Reference 1 loading.

As mentioned above, the calculations reported in response to RAI 5 were performed with 5.0 wt% enriched fuel with IFBA modeled over the entire axial length of the fuel.

Section 3.3.5.5 of Reference 1 states that modeling a shorter IFBA stack can result in larger differences in calculated eigenvalues between cases depleted with and without IFBA present. Tables 10 and 11 of Reference 1 show the Δk effects when a non-uniform axial burnup profile is modeled with IFBA modeled over the entire axial length of the fuel and with IFBA modeled over 120 inches, centered axially in the fuel rod. The more realistic IFBA model, 120 inches centered axially, is more limiting. MPS3 uses 120 inch IFBA centered axially in the fuel rod. Therefore, the MPS3 specific study was re-performed modeling a 120 inch IFBA region centered axially in the fuel assembly, which is representative of the actual IFBA length used in the core. Results of this new study are presented in Tables 8 and 9.

Tables 8 and 9 show the difference between a model accounting for IFBA during the depletion subtracted from a model that did not account for IFBA during the depletion. Both models use the RAI 21 axial burnup profiles, isotopic number densities from PARAGON depletions, and 3D SCALE Version 4.4 SFP models to determine SFP K_{eff}. A positive value indicates that the residual IFBA is more important than the spectral hardening effect. A negative value indicates that the spectral hardening caused by the IFBA has a higher reactivity worth than the residual ^{10}B. Residual IFBA is explicitly modeled in the IFBA model. The depletion parameters used in this study are discussed in response to RAI 30 (Table 11).

Table 8 - Δk in Region 2 for 3.0 and 5.0 wt% Fuel

BU (MWd/MTU)	Decay (years)	Δk 3.0 wt% $[k_{(no\ IFBA)}-k_{(IFBA)}]$	BU (MWd/MTU)	Decay (years)	Δk 5.0 wt% $[k_{(no\ IFBA)}-k_{(IFBA)}]$
15,000	0	+0.00720	40,000	0	-0.00160
15,000	10	+0.00838	40,000	10	-0.00112
20,000	0	-0.00033	45,000	0	-0.00173
20,000	10	+0.00018	45,000	10	-0.00175

Table 9 - Δk in Region 3 for 3.0 and 5.0 wt% Fuel

BU (MWd/MTU)	Decay (years)	Δk 3.0 wt% $[k_{(no\ IFBA)}-k_{(IFBA)}]$	BU (MWd/MTU)	Decay (years)	Δk 5.0 wt% $[k_{(no\ IFBA)}-k_{(IFBA)}]$
25,000	0	-0.00298	55,000	0	-0.00194
25,000	25	-0.00192	55,000	25	-0.00127
30,000	0	-0.00343	60,000	0	-0.00221
30,000	25	-0.00235	60,000	25	-0.00172

At low burnups the residual IFBA is enough to overcome the increase in reactivity due to spectral hardening. However, once the IFBA has burned out, the WCAP-16721-P analysis is potentially non-conservative. Rather than attempt to identify off-setting conservatisms in the existing analysis, a burnup penalty is applied to the final burnup credit curves for Regions 2 and 3 as part of the response to RAI 30. The burnup limits in Region 1 are low enough that the residual IFBA overcomes the increase in reactivity due to spectral hardening. Decay times greater than zero were investigated to ensure that the effect did not increase with increasing decay times. The limiting reactivity difference occurs at zero decay time for all but one case (Region 2 at 45,000 MWd/MTU). In that case the difference is very small and statistically insignificant.

Question 30

Please verify the average core exit temperature for nominal reactor coolant system (RCS) flow. Please verify the maximum core exit temperature for the minimum TS allowed RCS flow. Please provide the maximum fuel assembly exit temperature for the minimum TS allowed RCS flow. Please provide the moderator temperature used for each case in the RAI #21 responses.

Response:

As stated in response to RAI 4, the use of the word nominal in Table 2.8.3-1 of the stretch power uprate (SPU) license amendment request (LAR) refers only to the nominal value used in the analysis, and does not refer to the plant actual operating nominal temperature. The reported inlet temperature is a nominal value. Outlet temperature is maximized by assuming minimum RCS flow. The average core exit temperature at uprated conditions with nominal reactor coolant system (RCS) flow is 620.4 °F.

The maximum core average exit temperature for the minimum technical specification (TS) allowed RCS flow is 628 °F. Calculations supporting the response to RAI 21 used an exit temperature of 628 °F for blanketed fuel. The depletion calculations for no blanket fuel depleted in pre-uprate cycles used an outlet temperature of 620.6 °F. This temperature results from considering the minimum RCS flow with the pre-uprate power level of 3411 MWt.

Maximum fuel assembly exit temperature is dependent on assembly power and is higher than maximum core average temperature. Assembly power and exit temperature depend on fuel management strategy and vary with fuel burnup, burnable poison loading, and the number of fuel cycles in which the assembly is resident. For the calculation of isotopic number densities, it is not practical to simulate all possible power histories in detail. Rather, a conservative but constant value for assembly exit temperature is needed.

At any point in the lifetime of a fuel assembly, a lifetime average exit temperature can be determined from the lifetime average assembly relative power. The lifetime average assembly relative power is simply the assembly burnup divided by the total cycle burnup for the cycles the assembly has been used in up to that point in the lifetime of the assembly. The examples below will illustrate this calculation:

A) An assembly midway through its first cycle has a burnup of 12,000 MWd/MTU when the cycle burnup is 10,000 MWd/MTU. The lifetime average assembly relative power is 12,000/10,000 or 1.2. Using the core power (3650 MWt) x 1.2, the MPS3 core minimum flow, and the MPS3 nominal inlet temperature, the lifetime average assembly exit temperature can be calculated with a heat balance. For this example, the lifetime average assembly exit temperature at 12,000 MWd/MTU assembly burnup is 639.7 °F.

B) An assembly at the end of its third cycle has a burnup of 60,000 MWd/MTU. The total cycle burnup of the three cycles the assembly was used in is 60,000 MWd/MTU. The lifetime average assembly relative power is 60,000 / 60,000 or 1.0. Using the core power (3650 MWt) x 1.0, the MPS3 core minimum flow, and the MPS3 nominal inlet temperature, the lifetime average assembly exit temperature is 628 °F. This assembly would very likely have experienced higher than lifetime average exit temperatures in the first two cycles (similar to example A) and lower than lifetime average in the final cycle.

Using this method, a survey of the fuel assemblies in the four most recent MPS3 reload cores was performed to identify a bounding lifetime average exit temperature as a function of assembly burnup. Individual assembly values were calculated at mid and end of cycle for first and second burn fuel and at end of cycle for third burn fuel. A composite maximum power history was determined that bounds the fuel assemblies in the survey. The curve (linear between points) defined by these values does not represent the history of any individual assembly, but bounds the maximum lifetime average values of the assemblies surveyed at the assembly burnups. Table 10 shows the results of this survey:

Table 10 – Limiting Temperature as a Function of Assembly Burnup

Assembly Burnup (MWd/MTU)	Maximum Exit Temperature (°F)
0	648.7
28,000	648.7
50,000	643.6
60,000	635.8

It is not credible to assume lifetime average assembly exit temperatures significantly higher than those shown in Table 10.

WCAP-16721-P depletion calculations were performed using the maximum core average exit temperature (628 °F). In order to quantify the effect of assembly exit temperatures above core average, Table 10 bounding temperatures were combined with the limiting burnup profiles and axial nodalization identified in response to RAI 21, and the parameters in Table 11. Depletion calculations were performed using PARAGON, and discharged assembly reactivities were determined using SCALE Version 4.4 SFP models for all cases.

Table 11 – Depletion and Fuel Assembly Parameters

Parameter	WCAP-16721-P	RAIs 28 & 30, mid-enriched blankets
Axial Burnup Distribution	Profile 5 from DOE Topical Report	Limiting profiles identified in RAI 21
Number of axial zones modeled	4	24
Tin	556.4 °F	556.4 °F
Tout	628 °F	See Table 10
Soluble Boron present during depletion	constant 1000 ppm	constant 1000 ppm
Power	3650 MWt	3650 MWt
Theoretical Density of fuel	97.5%	97.5%
Fuel pellet shape	solid, right cylinder (i.e., no dishing or chamfering)	solid, right cylinder (i.e., no dishing or chamfering)
Design Basis Fuel Assembly	Westinghouse 17x17 STD	Westinghouse 17x17 STD
Fuel initial enrichments	3.0, 4.0, and 5.0 wt%	3.0, 4.0, and 5.0 wt%
Blankets modeled?	No	Yes

Because the response to RAI 21 demonstrated that fuel containing mid-enriched blankets is more limiting than fuel containing natural blankets, only mid-enriched blanket fuel is considered in this penalty calculation. The same conditions summarized in Table 11 were also used for the IFBA penalty calculation presented in response to RAI 28. The results of this combined exit temperature and burnup profile calculation are summarized in Table 12 for Region 2 and Region 3. Note that in Table 12 a negative value denotes an increase in reactivity.

Table 12 – Temperature and Burnup Profile Effects (Δk) for 3.0, 4.0, and 5.0 wt% Fuel

	BU (MWd/MTU)	Decay Time (yrs)	Region 2 Δk		BU (MWd/MTU)	Decay Time (yrs)	Region 3 Δk
3.0 wt%	15,000	0	-0.00244	3.0 wt%	25,000	0	-0.01364
	15,000	10	-0.00290		25,000	25	-0.01092
	20,000	0	-0.00723		30,000	0	-0.01500
	20,000	10	-0.00932		30,000	25	-0.01118
4.0 wt%	25,000	0	-0.00273	4.0 wt%	40,000	0	-0.00125
	25,000	10	-0.00124		40,000	25	+0.00918
	30,000	0	-0.00313		45,000	0	-0.00072
	30,000	10	-0.00030		45,000	25	+0.01290
5.0 wt%	40,000	0	-0.00243	5.0 wt%	55,000	0	+0.01049
	40,000	10	+0.00138		55,000	25	+0.02602
	45,000	0	-0.00199		60,000	0	+0.01453
	45,000	10	+0.00374		60,000	25	+0.03114

RAI 21 burnup shape and nodalization results indicated that some of the WCAP-16721-P K_{eff} results were non-conservative in the 20,000-30,000 MWd/MTU burnup range (Table 21-6, RAI 21). Combining increased moderator exit temperature with the conservative axial nodalization and burnup profiles results in larger reactivity penalties in the 20,000-30,000 MWd/MTU as shown in Table 12. At high burnups, the conservatism inherent in WCAP-16721-P burnup profile ("Profile 5", RAI 21) becomes much larger than the effect of increased moderator temperature.

The burnup limits in Region 1 are too low to be significantly affected by these factors for three reasons.

- WCAP-16721-P calculations were performed with a uniform profile. As indicated in RAI question 3, a uniform axial shape is considered conservative for burnup < 10,000 MWd/MTU. The maximum calculated burnup requirement in WCAP-16721-P for Region 1 is 5,743 MWd/MTU (WCAP Table 5-1). Therefore, there are no end effect concerns for Region 1.
- Depletion history effects accumulate gradually with increasing assembly burnup, and the burnup requirement in Region 1 is very low.
- Region 1 TS Figure 3.9-1, which is being retained, requires over 2,200 MWd/MTU more burnup at 5 wt% than required by the WCAP-16721-P analysis at 5 wt%. The additional burnup specified by TS Figure 3.9-1 over and above the WCAP-16721-P calculated minimum fuel assembly burnup provides margin to further illustrate that no burnup penalty is needed for Region 1.

Rather than attempt to identify off-setting conservatisms in the existing analysis for Regions 2 and 3, burnup penalties calculated from reactivity penalties are applied to the final burnup credit curves. Prior to this RAI response, the response to RAI 21 and RAI 26 included a reactivity penalty. In this response, reactivity penalties are also identified for RAI 28 and RAI 30. The penalty in RAI 30 includes and supersedes the penalty identified in RAI 21. The reactivity penalties identified in RAIs 26, 28, and 30 are summarized in Tables 13 and 14. An IFBA penalty was not explicitly determined for 4.0 wt% fuel, so a generic penalty is applied. The penalty chosen is 0.00450 Δk and is larger than the penalties calculated for either 3.0 or 5.0 wt% fuel. In addition to the penalties applied in response to the RAIs, the unallocated administrative margin is increased from 0.001 Δk_{eff} used in WCAP-16721-P to 0.005 Δk_{eff}. An additional 0.004 Δk_{eff} of unallocated administrative margin is included in Tables 13 and 14.

Table 13 – Summary of Region 2 Penalties (Δk)

	BU (MWd/MTU)	Decay Time (yrs)	Operating History RAI 26	IFBA RAI 28	Exit Temp. & Burnup Profile RAI 30	Additional Admin. Margin	Total
3.0 wt%	15,000	0	-0.00200	0	-0.00244	-0.00400	**-0.00844**
	15,000	10	-0.00200	0	-0.00290	-0.00400	**-0.00890**
	20,000	0	-0.00200	-0.00033	-0.00723	-0.00400	**-0.01356**
	20,000	10	-0.00200	0	-0.00932	-0.00400	**-0.01532**
4.0 wt%	25,000	0	-0.00200	-0.00450	-0.00273	-0.00400	**-0.01323**
	25,000	10	-0.00200	-0.00450	-0.00124	-0.00400	**-0.01174**
	30,000	0	-0.00200	-0.00450	-0.00313	-0.00400	**-0.01363**
	30,000	10	-0.00200	-0.00450	-0.00030	-0.00400	**-0.01080**
5.0 wt%	40,000	0	-0.00200	-0.00160	-0.00243	-0.00400	**-0.01003**
	40,000	10	-0.00200	-0.00112	+0.00138	-0.00400	**-0.00574**
	45,000	0	-0.00200	-0.00173	-0.00199	-0.00400	**-0.00972**
	45,000	10	-0.00200	-0.00175	+0.00374	-0.00400	**-0.00401**

Table 14 – Summary of Region 3 Penalties (Δk)

	BU (MWd/MTU)	Decay Time (yrs)	Operating History RAI 26	IFBA RAI 28	Exit Temp. & Burnup Profile RAI 30	Additional Admin. Margin	Total
3.0 wt%	25,000	0	-0.00200	-0.00298	-0.01364	-0.00400	**-0.02262**
	25,000	25	-0.00200	-0.00192	-0.01092	-0.00400	**-0.01884**
	30,000	0	-0.00200	-0.00343	-0.01500	-0.00400	**-0.02443**
	30,000	25	-0.00200	-0.00235	-0.01118	-0.00400	**-0.01953**
4.0 wt%	40,000	0	-0.00200	-0.00450	-0.00125	-0.00400	**-0.01175**
	40,000	25	-0.00200	-0.00450	+0.00918	-0.00400	**-0.00132**
	45,000	0	-0.00200	-0.00450	-0.00072	-0.00400	**-0.01122**
	45,000	25	-0.00200	-0.00450	+0.01290	-0.00400	**+0.00240**
5.0 wt%	55,000	0	-0.00200	-0.00194	+0.01049	-0.00400	**+0.00255**
	55,000	25	-0.00200	-0.00127	+0.02602	-0.00400	**+0.01875**
	60,000	0	-0.00200	-0.00221	+0.01453	-0.00400	**+0.00632**
	60,000	25	-0.00200	-0.00172	+0.03114	-0.00400	**+0.02342**

Table 15 shows eigenvalues used to calculate the burnup worth, the largest (most negative) applicable reactivity penalty identified in Tables 13 and 14, the burnup penalty applied, and the worth of the burnup penalty applied. The eigenvalues used to calculate the burnup worth are taken from Tables 4-9 and 4-10 of WCAP-16721-P. For each enrichment/region/burnup combination, the penalty is conservatively chosen as zero or the most negative of the 0 decay time value or the decay time value from Tables 13 and 14.

The Table 15 burnup worth was calculated by assuming a linear reactivity change over an interval of burnup near the point of potential non-conservatism. The difference in two SCALE Version 4.4 k_{eff} values was divided by the associated burnup difference to determine the reactivity worth associated with an increase in assembly average burnup. For example, the Region 2, 3.0 wt% burnup worth was calculated as:

$$(0.87720 - 0.94421)/(25 - 15) = -0.00670 \ \Delta k/GWd/MTU$$

The "BU Penalty Applied" column in Table 15 shows the total burnup penalty that will be applied to the burnup limit for each initial enrichment at all decay times. Finally, the "Worth of BU Penalty" is the burnup penalty multiplied by the burnup worth to demonstrate that the reduction in reactivity due to the penalty is greater than the maximum identified non-conservatism. Net burnup credit is not taken in Table 15, even though net credit is indicated in Table 14 for the 5 wt% cases in Region 3.

Monte Carlo uncertainties were not included in the determination of the burnup worth or the reactivity of the penalty because, as explained in response to RAI 16, the Monte Carlo uncertainty is already accounted for in the calculation of the total biases and uncertainties. Additionally, the conditions used to determine the reactivity penalty and the penalty selection method are both very conservative.

Table 15 – Burnup Worths and Penalties

Region 2

	BU (GWd/MTU)	k_{eff}	σ	BU Worth (Δk/GWd/MTU)	Max Reactivity (Δk)	BU Penalty Applied (GWd/MTU)	Worth of BU Penalty (Δk)
3.0 wt%	15	0.94421	0.00034	-0.00670	-0.01532	2.30	-0.01541
	25	0.87720	0.00029				
4.0 wt%	25	0.95012	0.00030	-0.00519	-0.01363	2.65	-0.01375
	35	0.89821	0.00030				
5.0 wt%	35	0.95615	0.00032	-0.00442	-0.01003	2.30	-0.01017
	45	0.91198	0.00031				

Region 3

	BU (GWd/MTU)	k_{eff}	σ	BU Worth (Δk/GWd/MTU)	Max Reactivity (Δk)	BU Penalty Applied (GWd/MTU)	Worth of BU Penalty (Δk)
3.0 wt%	25	0.91738	0.00033	-0.00539	-0.02443	4.70	-0.02533
	35	0.86343	0.00032				
4.0 wt%	35	0.93691	0.00032	-0.00460	-0.01175	2.60	-0.01196
	45	0.89088	0.00027				
5.0 wt%	45	0.94932	0.00032	-0.00422	0.0	0	0
	55	0.90713	0.00030				

The only penalty identified in Tables 13 and 14 that would affect the maximum fresh fuel enrichment is the additional administrative margin. The other penalties are not applicable because the reactivity of fresh fuel is not impacted by depletion conditions. In order to increase the amount of administrative margin associated with the determination of the maximum allowable fresh fuel enrichment, the enrichment uncertainties reported in Tables 4-6 and 4-7 of WCAP-16721-P were used. The enrichment uncertainties reported assumed a 0.05 wt% increase in initial enrichment. To find the enrichment associated with a change in reactivity of 0.004 Δk, 0.004 Δk was divided by the enrichment uncertainty and multiplied by 0.05 wt%. Maximum fresh fuel enrichments for Regions 2 and 3 are shown in Tables 16 and 17, respectively. The calculated Region 1 maximum fresh fuel enrichment of 3.79 wt% is bounded by the existing TS Figure 3.9-1 value (3.7 wt%).

The burnup versus enrichment curves provided in response to this RAI completely supersede the burnup versus enrichment curves previously provided. Tables 16 and 17 and Figures 2 and 3 show the new burnup versus enrichment curves. The calculations supporting these curves have been performed with conservatisms including high temperature, high soluble boron, maximum IFBA loading, higher than credible fuel density, and with the most conservative applicable burnup profile. In addition, allowances for depletion power history uncertainty, burnup worth uncertainty, and measured burnup uncertainty are conservatively treated as biases, and administrative margin is increased to 0.5% Δk. These factors provide assurance that the storage requirements presented in Tables 16 and 17 and Figures 2 and 3 are conservative.

Table 16
Minimum Required Assembly-Average Burnup versus Initial ^{235}U Enrichment and Decay Time for the "Region 2" Storage Configuration

Initial Enrichment (wt% ^{235}U)	Assembly Average Burnup (MWd/MTU)		
	0 yr Decay	5 yr Decay	10 yr Decay
1.79	0	0	0
3.00	19191	18107	17378
4.00	31811	29849	28808
5.00	44638	41952	40290

The required assembly burnup as a function of ^{235}U enrichment in the "Region 2" storage configuration is described by the following polynomials:

$$\text{Assembly Burnup (0 yr decay)} = +489.007\ e^3\ -5764.586\ e^2\ +34878.836\ e\ -46767.429$$

$$\text{Assembly Burnup (5 yr decay)} = +510.476\ e^3\ -5945.212\ e^2\ +34470.872\ e\ -45581.560$$

$$\text{Assembly Burnup (10 yr decay)} = +421.399\ e^3\ -5030.783\ e^2\ +31053.732\ e\ -41883.913$$

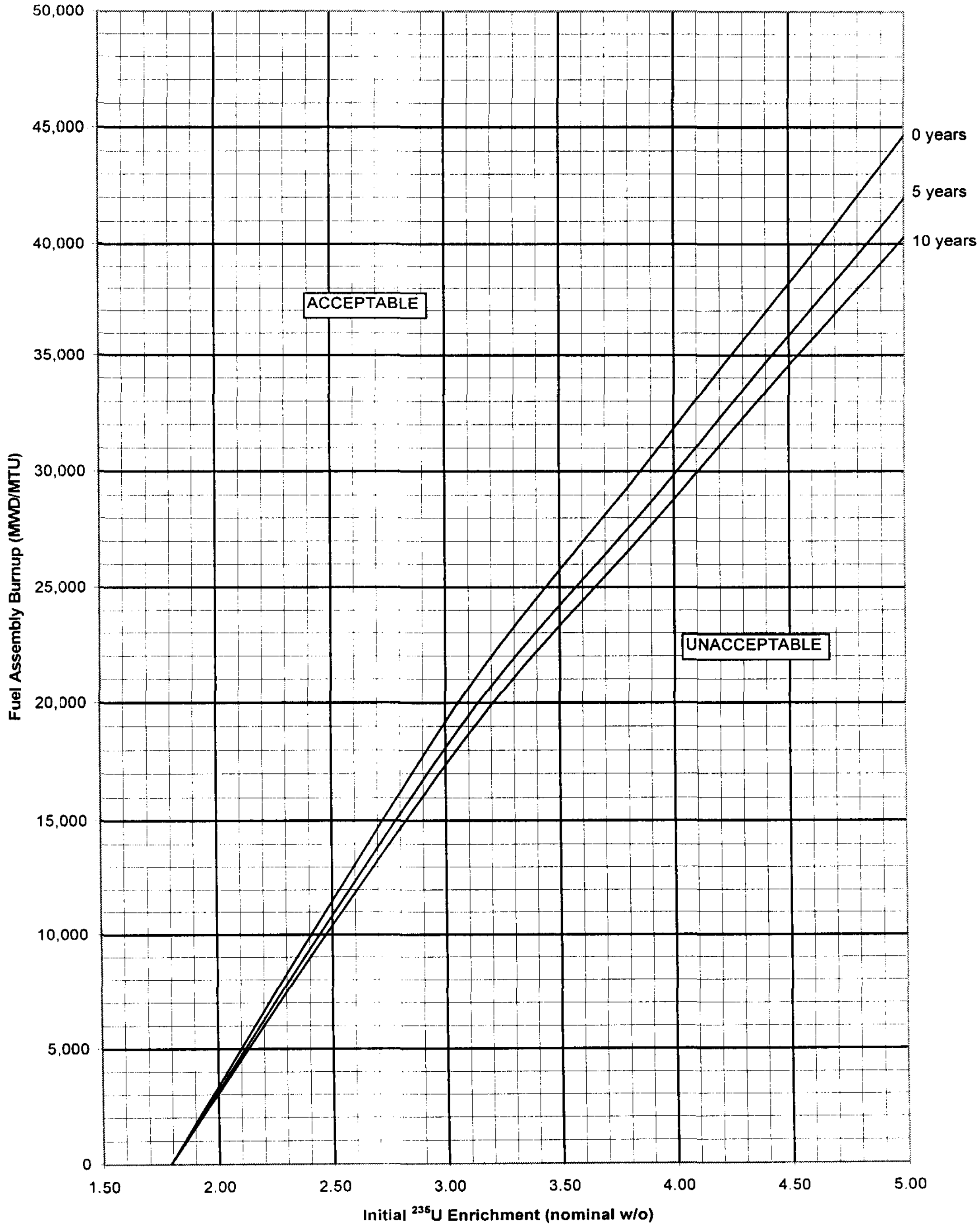

Figure 2 – Minimum Required Fuel Assembly Burnup versus Initial ^{235}U Enrichment for the "Region 2" Storage Configuration

NOTE: For assemblies from Post-Uprate (3650 MWt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 wt% U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel. Fuel batches A, B, C, and D may not be stored in Region 2.

Table 17
Minimum Required Assembly-Average Burnup versus Initial ^{235}U Enrichment and Decay Time for the "Region 3" Storage Configuration for Post Uprate Cores

Initial Enrichment (wt% ^{235}U)	Assembly Average Burnup (MWd/MTU)					
	0 yr Decay	5 yr Decay	10 yr Decay	15 yr Decay	20 yr Decay	25 yr Decay
1.43	0	0	0	0	0	0
3.00	30216	28304	27172	26413	25730	25374
4.00	43389	40617	38685	37440	36493	35685
5.00	55566	52057	49717	47978	46835	45874

The required assembly burnup as a function of ^{235}U enrichment in the "Region 3" storage configuration is described by the following polynomials:

$$\text{Assembly Burnup (0 yr decay)} = +522.404\,e^3 - 6766.842\,e^2 + 41211.965\,e - 46623.210$$

$$\text{Assembly Burnup (5 yr decay)} = +500.629\,e^3 - 6444.047\,e^2 + 38898.060\,e - 43910.737$$

$$\text{Assembly Burnup (10 yr decay)} = +564.139\,e^3 - 7010.171\,e^2 + 39711.046\,e - 44101.356$$

$$\text{Assembly Burnup (15 yr decay)} = +563.298\,e^3 - 7004.075\,e^2 + 39213.501\,e - 43399.874$$

$$\text{Assembly Burnup (20 yr decay)} = +554.180\,e^3 - 6860.665\,e^2 + 38282.980\,e - 42335.826$$

$$\text{Assembly Burnup (25 yr decay)} = +620.608\,e^3 - 7508.292\,e^2 + 39906.561\,e - 43527.461$$

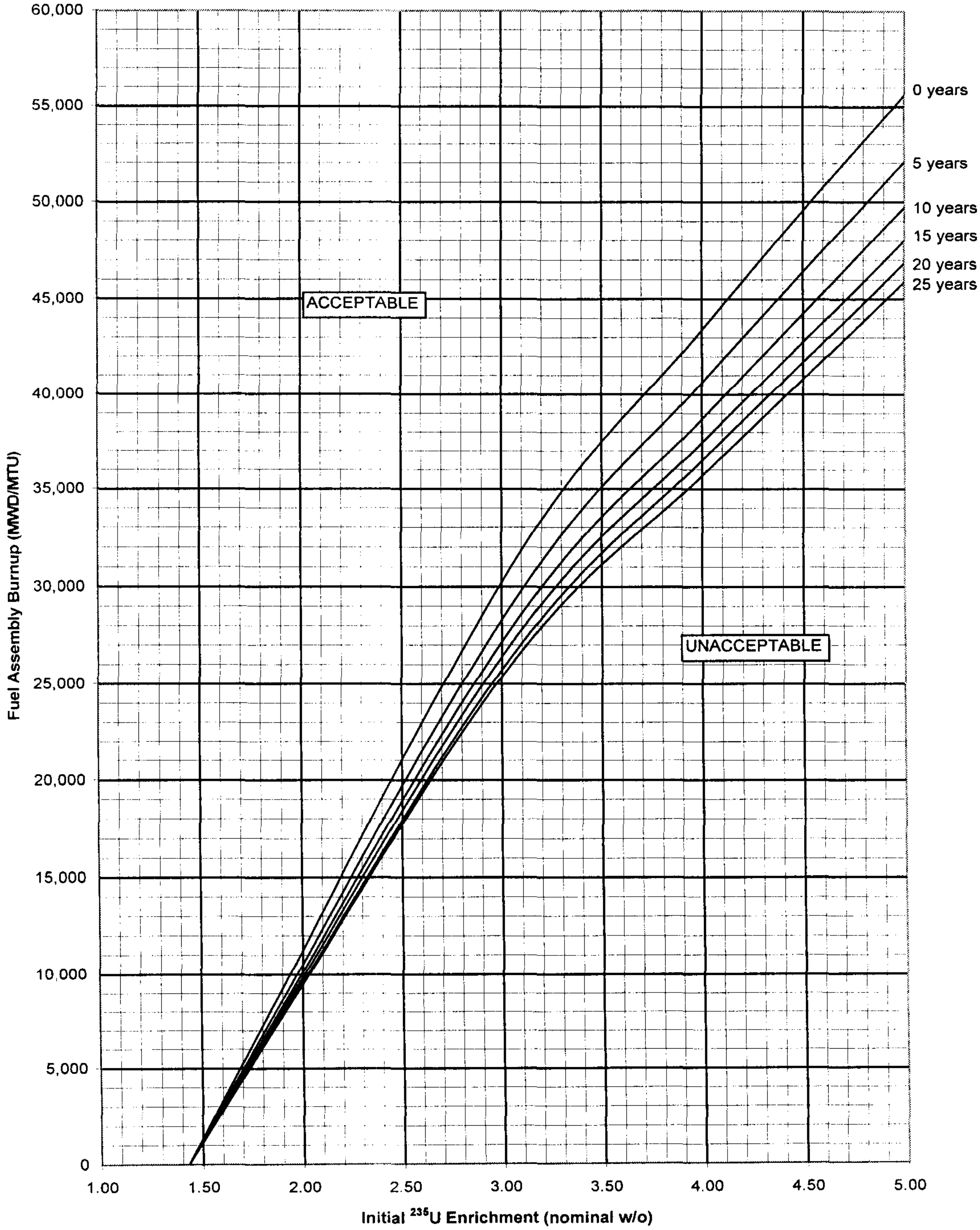

Figure 3 – Minimum Required Fuel Assembly Burnup versus Initial ^{235}U Enrichment for the "Region 3" Storage Configuration

NOTE: For assemblies from Post-Uprate (3650 MWt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 wt% U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel.

References:

1. C. E. Sanders and J. C. Wagner, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," NUREG/CR-6760, March 2002.

ATTACHMENT 2

RESPONSE TO RAI QUESTIONS 28 AND 30

REGARDING THE SPENT FUEL POOL LICENSE AMENDMENT REQUEST

UPDATED MARKUP OF TECNNICAL SPECIFICATIONS PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL - REACTIVITY

LIMITING CONDITION FOR OPERATION

3.9.13 The Reactivity Condition of the Spent Fuel Pool shall be such that k_{eff} is less than or equal to 0.95 at all times.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION: With k_{eff} greater than 0.95:

a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and

b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3, or 3.9-4, to a location for which that fuel assembly is allowed. (,) (or 3-9.5)

SURVEILLANCE REQUIREMENTS

4.9.13.1.1. Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment and burnup limits of Figure 3.9-3 by checking the fuel assembly's design and burn-up documentation. (, decay time,)

4.9.13.1.3. Ensure that all fuel assemblies (used exclusively in pre-uprate (3411 MWt) conditions which are) to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel assembly's design, decay time, and burn-up documentation. Ensure that all fuel assemblies used in post-uprate (3650 MWt) conditions which are to be placed in Region 3 fuel storage are within the enrichment, decay time, and burn-up limits of Figure 3.9-5 by checking the fuel assembly's design, decay time, and burn-up documentation.

November 28, 2000

FIGURE 3.9-3 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 2 Storage Configuration

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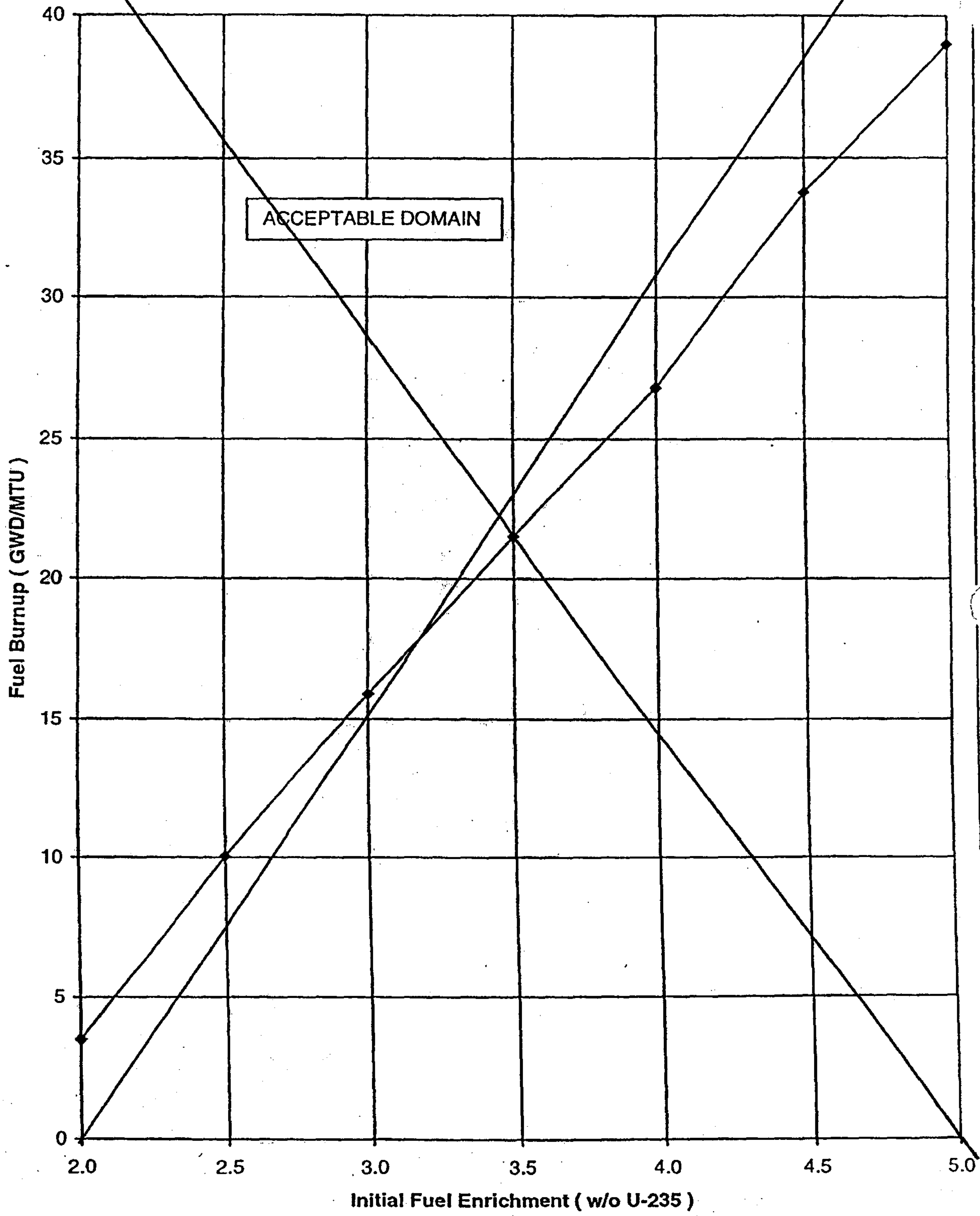
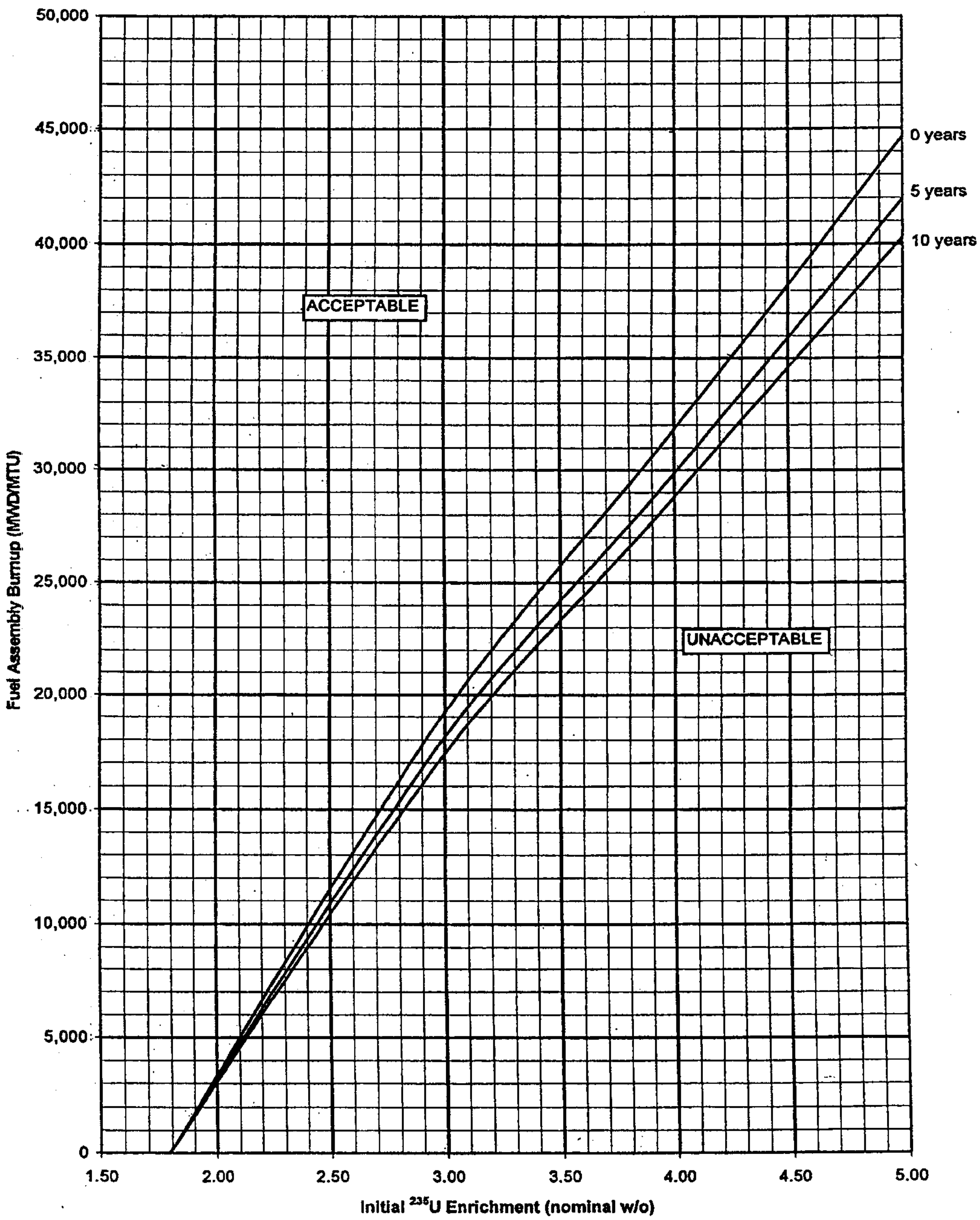


Figure 3.9-3 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 2 Storage Configuration



NOTE: For assemblies from Post-Uprate (3650 MWt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 w/o U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel. Fuel batches A, B, C, and D may not be stored in Region 2.

FIGURE 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for assemblies from pre-uprate (3411 MWt) cores

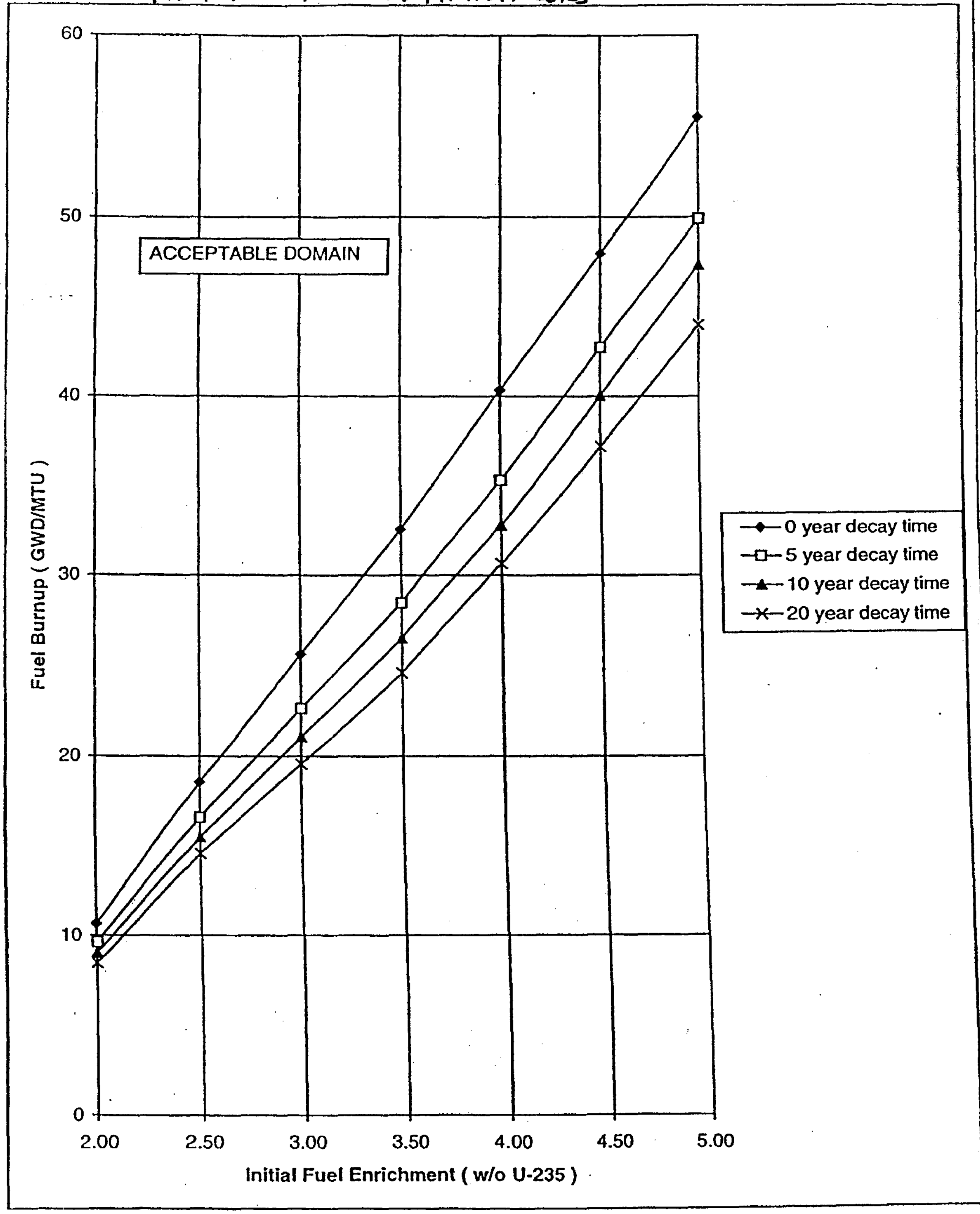
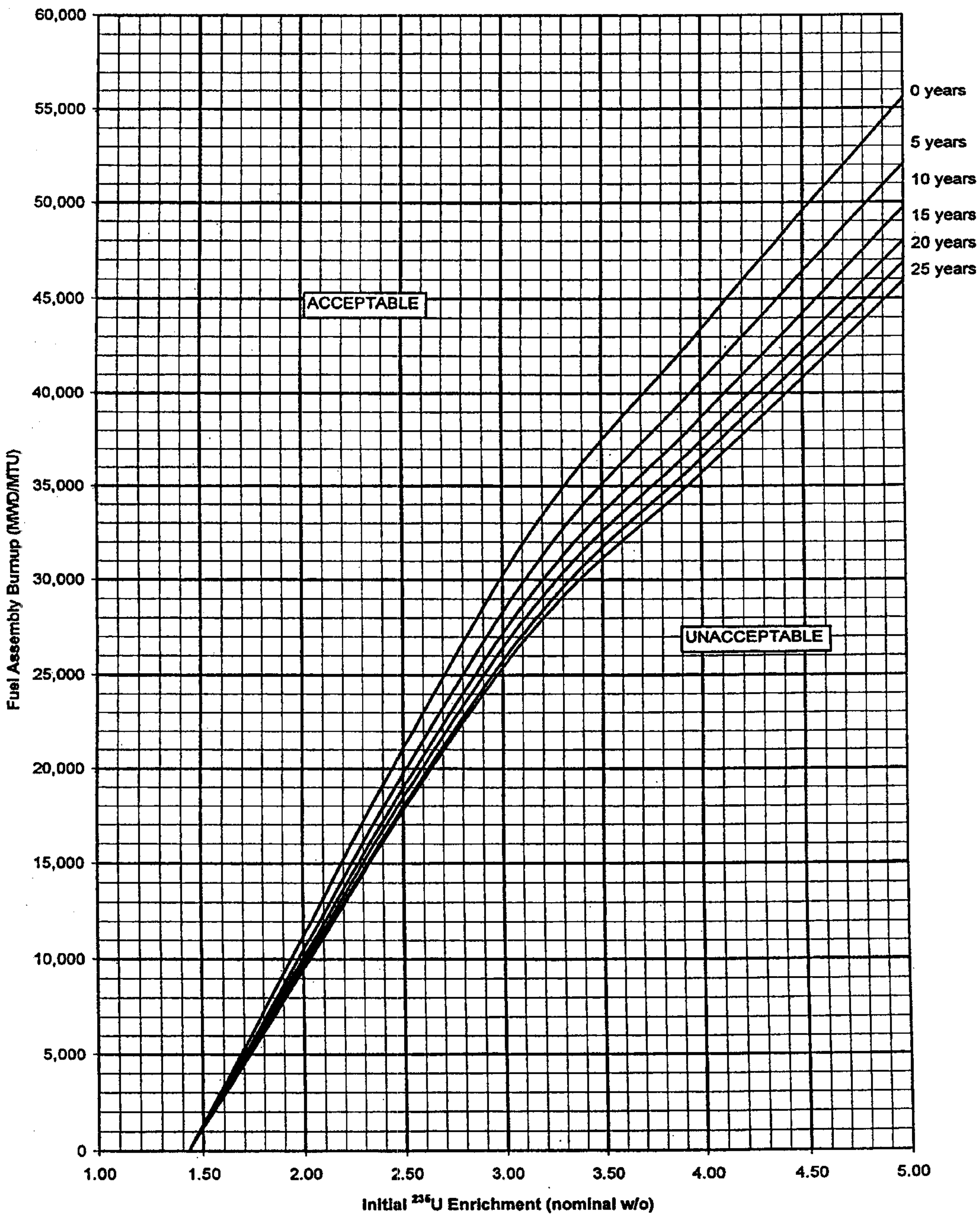


Figure 3.9-5 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Post-Uprate (3650 Mwt) Cores



NOTE: For assemblies from Post-Uprate (3650 MWt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 w/o U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water. The storage rack Regions are:

a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and can store fuel in 2 storage configurations:

(1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.

(2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup [and fuel decay time] as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.

c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4~~,~~ [for assemblies used exclusively in pre-uprate (3411 MWt) cores or Figure 3.9-5 for assemblies used in post-uprate (3650 MWt) cores,] fuel may be stored in all available Region 3 storage locations. ~~The Boraflex contained inside these storage racks is not credited.~~

DRAINAGE

5.6.2 The spent fuel storage pool is design and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

ATTACHMENT 3

RESPONSE TO RAI QUESTIONS 28 AND 30

REGARDING THE SPENT FUEL POOL LICENSE AMENDMENT REQUEST

MARKUP OF TECNNICAL SPECIFICATIONS BASES PAGE FOR INFORMATION ONLY

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

3/4.9.13 SPENT FUEL POOL - REACTIVITY

During normal Spent Fuel Pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than 0.95 in an unborated water environment.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 3-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, a maximum nominal 5 weight percent fuel enrichment, and the use of blocking devices in certain fuel storage locations, as specified by the interface requirements shown in Figure 3.9-2.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 4-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-1.

Maintaining Keff at less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-3.

and fuel decay time

Maintaining Keff at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

, 3.9-4, and 3.9.5

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3 ~~and 3.9-4~~ ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis.

3.9-4 for assemblies used exclusively in the pre-uprate (3411 MWt) cores and Figure 3.9-5 for assemblies used in the post-uprate (3650 MWt) cores.

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9-2, 3.9-3 ~~and 3.9-4~~ are complied with.

, 3.9-4, and 3.9.5 as well as restrictions specified in the Note on Figures 3.9-3 and 3.9-5

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region 1 3-OUT-OF-4 storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region 1 3-OUT-OF-4 storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region 1 3-OUT-OF-4 boundary to other storage regions and minimize the number of boundaries where a fuel misplacement incident can occur.