NORTHEAST UTILITIES

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Docket No. 50-336
B12275
A05935

Office of Nuclear Reactor Regulation
Attn: Mr. Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Storage of Consolidated Spent Fuel

In May, 1986,(1) Northeast Nuclear Energy Company (NNECO) submitted to the NRC Staff a request to amend its operating license, No. DPR-65, for Millstone Nuclear Power Station, Unit No. 2, to allow the storage of consolidated spent fuel in the Unit No. 2 spent fuel storage pool. As a result of the NRC Staff review of this proposal, the NRC Staff forwarded to NNECO a Request for Additional Information.(2) The purpose of this letter is to provide the NRC Staff the requested information.

Question #1:

Figure 3.9-3 shows the minimum required fuel assembly exposure as a function of initial enrichment for storage in Region 2 as consolidated fuel. If fuel rods from different assemblies and of different enrichments can be consolidated in one cannister, what value of initial enrichment is assumed in complying with Figure 3.9-3?

Response:

For the cannister under consideration, the pin with the highest enrichment determines the enrichment assumed for compliance with Figure 3.9-3 of the license amendment request.(1)

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(1) J.F. Opeka letter to A.C. Thadani, dated May 21, 1986, "Millstone Nuclear Power Station, Unit No. 2 Proposed Change to Technical Specifications Storage of Consolidated Fuel.

(2) D.H. Jaffe letter to J.F. Opeka, dated July 25, 1986, "Request for Additional Information on Storage of Consolidated Fuel for Millstone Unit No. 2".

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Question #2

How is the reactivity effect of less than a full consolidated storage box (less than 352 rods) accounted for?

Response:

The consolidation process permits the placement of solid metal rods in positions where fuel rods are missing. For those instances were solid rods are not used, a limited number of fuel rods can be omitted based on the attached Figure 1. Using Figure 1, the reactivity effect of less than a full consolidated storage box can be established by determining the maximum number of fuel rods that can be omitted while maintaining K-eff at 0.95 or less.

Question #3:

What are the values of the biases and calculated uncertainties referred to for Regions 1 and 2, and how were they derived?

Response:

The value of the bias is 0.00138 and the 95/95 confidence level calculation uncertainty is 0.00714. The validation report is enclosed as Attachment I.

Question #4:

Explain in more detail how the Region 2 allowable burnup for each initial enrichment accounts for the underestimation of K-effective due to the assumption for uniform axial burnup.

Response:

The non-uniform burnup distribution which produced the highest difference in reactivity in the Region 2 spent fuel rack when compared with the uniform distribution results is shown on page 3-6 of the license amendment request.[1] The K-eff is 0.0114. This K-eff was used in Figure 3-3[1] to determine the burnup needed to accommodate the increase in reactivity due to non-uniform burnup. The burnup was found for each initial enrichment shown in Figure 3-3.[1] The actual maximum burnup was 1,400 MWD/T for a K-eff of 0.0114. For conservatism, the burnup correction for any initial enrichment was assumed to be 1,800 MWD/T. This value was added to the maximum uniform burnup requirement for each initial enrichment.

Question #5

If Figure 3-4 is based on an infinite array of consolidated fuel, justify why Figures 3-4 and 3-5 need not be derived based on the higher reactivity configuration of one storage pattern of consolidated fuel boxes surrounded by an infinite array of regular fuel assemblies.

Response:

Spent Fuel Pool Technical Specification 3/4.9.20, SPENT FUEL POOL, will ensure that the K-eff of the spent fuel pool will always be less than 0.95 for any mix of unconsolidated or consolidated fuel. The Technical Specification requires that the blocked cell remain until the Region 2 STORAGE PATTERN of the spent fuel pool racks has been filled. At this time, consolidated fuel can be placed in a previously cell-blocked location only if it is completely surrounded by consolidated fuel. In this way, the unconsolidated fuel will be next to consolidated fuel which is stored in a 3 out of 4 pattern. The reactivity of consolidated fuel adjacent to the unconsolidated fuel is less than K-eff 0.95 since it is 3 out of 4, and not 4 out of 4.

Question #6:

Technical Specification 5.6.1.d should include additional wording to clarify that consolidated fuel can be stored in the 4th location of the storage rack only if the surrounding locations are occupied by consolidated fuel storage boxes.

Response:

We propose that Technical Specification 5.6.1.d be modified to read as follows:

> "Region II of the spent fuel storage pool is designed to permit storage of consolidated fuel in the 4th location of the storage rack and ensure a K-eff less than or equal to 0.95. Placement of consolidated fuel in the 4th location is only permitted if all surrounding cells of the STORAGE PATTERN are occupied by consolidated fuel."

The attached revised page 5-5 reflects incorporation of this change.

Question #7:

The NRC Staff recommends that a Technical Specification Surveillance Requirement be incorporated for consolidated fuel to verify the integrity of the fuel and structural elements before movement or placement in the spent fuel pool.

NRC Staff Clarification to Question #7:

What method does NNECO propose to verify the integrity of the storage cannister after it has been loaded with fuel rods?

Response;

Section 4.6.2 of the license amendment request(1) describes the Quality Assurance requirements with which the design, procurement and fabrication of the consolidated fuel storage boxes will comply to ensure that all manufacturing and installation activities conform to the acceptable quality requirements throughout all areas of performance.

Static and impact analyses were performed to verify the adequacy of the consolidated fuel storage box design for all the service loads associated with both the consolidation operation and storage in the spent fuel racks. The results of the structural analysis demonstrates that the consolidation box and cover can be safely lifted and transported using the cover as the lift point. The consolidation box is designed such that it will not be overstressed when subjected to a tensile load of 6000 lbs. The insert assembly supporting the weight of the fuel rods can withstand an impact of 5 Gs.

The cover assembly is a spring-loaded self-locking device which has a visual indicator when the cover has been engaged and locked in place. Finally, the cover is dimensionally similar to the upper end fitting of the fuel assembly, thereby permitting the consolidated fuel storage box to be transported by the fuel handling tool/system.

Additionally, it should be noted that, prior to placement of a consolidated storage box in the spent fuel racks, the consolidation operation will have transported the fully loaded consolidated storage box to the temporary racks within the Cask Laydown Area to permit access by the fuel handling tool/system, demonstrating that the fully-loaded consolidated storage box can be transported and placed in the racks while maintaining its integrity.

These measures were introduced into the design of the consolidated fuel storage box so that there would be no increase in the probability of a fuel handling accident as a result of storing consolidated spent fuel. We consider these measures to be adequate without any augmentation of the previously proposed surveillance requirement.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Senior Vice President

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,060 + 700/-0 cubic feet.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a K_{eff} less-than-or-equal-to 0.95. The maximum fuel enrichment to be stored in these racks is 3.70 weight percent of U-235.

b) Region I of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations to ensure a K_{eff} less-than-or-equal-to 0.95 with the storage pool filled with unborated water. Fuel assemblies stored in this region may have a maximum fuel enrichment of 4.5 weight percent of U-235. Consolidated fuel storage boxes may also be stored in this region.

c) Region II of the spent fuel storage pool is designed and shall be maintained with a 9.0 inch center to center distance between storage locations to ensure a K_{eff} less-than-or-equal-to 0.95 with the storage pool filled with unborated water. Fuel assemblies stored in this region must comply with Figure 3.9-1 to ensure that at least 85% of the design burn-up has been sustained. The contents of consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3.

d) Region II of the spent fuel storage pool is designed to permit storage of consolidated fuel in the 4th location of the storage rack and ensure a K_{eff} less-than-or-equal-to 0.95. Placement of consolidated fuel in the 4th location is only permitted if all surrounding cells of the STORAGE PATTERN are occupied by consolidated fuel.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22'6".

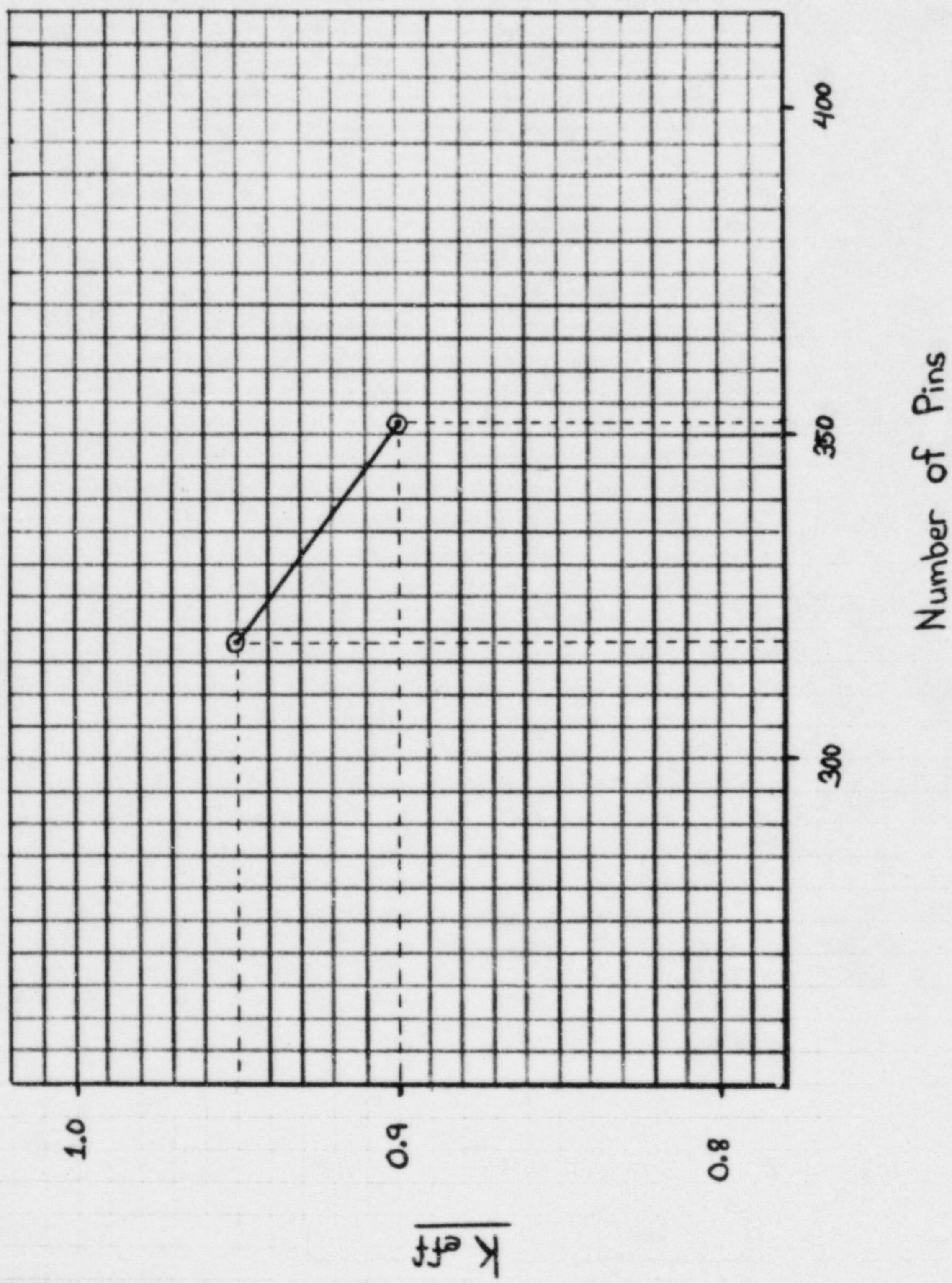
CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 384 storage locations in Region I and 962 storage locations in Region II for a total of 1346 storage locations.

Figure 1

Keff versus Number of Pins per Box

(for pins of all enrichments at minimum burnup or greater)



ATTACHMENT I

Qualification of Analytical
Methods Used In Spent Fuel
Storage Rack Analyses

Attachment I

QUALIFICATION OF ANALYTICAL METHODS USED IN SPENT FUEL STORAGE RACK ANALYSES

I. Purpose

The purpose of this attachment is to provide qualification of the calculational model and evaluation of calculational uncertainties and/or bias factors used in analyzing spent fuel storage racks, especially the HI-CAP™ racks employing steel boxes and super HI-CAPs containing boron carbide poison. This is based on the analysis of a variety of reactor and laboratory experiments. The methods of cross-section generation are essentially those of C-E's physics design procedures modified appropriately for use in four group transport, discrete ordinate method criticality calculations, and Monte Carlo codes.

II. Calculational Uncertainty and Bias

The results of the analysis of a series of UO_2 critical experiments are summarized in Table I. These are calculated using the methods described by Gavin (Reference 1) for CEPAK 2.3, which is used in present storage rack calculations. Table I includes the mean and standard deviation for this CEPAK model.

Although the spatial solution for the flux distribution was obtained by use of a diffusion theory code such as PDQ-7, transport corrections for the reflector and heterogeneous lattice effects were employed. Thus, for example, in Reference 8, the 4.3 w/o U-235 infinite lattice of close-packed assemblies in room temperature water had a K-eff of 1.4547 in PDQ and 1.4568 in DOT, the conservative bias in DOT of 0.0021 will be ignored. These calculations support use of the differential cross-section data base and broad group cross section generation codes.

Since fuel storage arrays do involve the spacing of the fuel assemblies at larger separation distances than in typical PWR reactor lattices, the predictive capability of the calculational model was tested on the following experiments. In these analyses done for this memo, the spatial flux solution was obtained directly with the transport code, ANISN. To assess the accuracy of the calculational model in predicting the multiplication factor of fuel assemblies having a separation distance sufficiently large so as to be isolated, analyses were carried out for a group of subcritical exponential experiments on clusters of 3.0 w/o UO_2 fuel pins clad with type 304 S.S. and moderated by H_2O (page 165 of Reference 9). The cluster sizes analyzed vary from 181 to 301 fuel rods so as to encompass the range of sizes typical of current PWR fuel assemblies. The multiplication factors for the lattices analyzed using axial bucklings deduced from the reported relaxation lengths are tabulated below.

No. of Fuel Rods	K-eff
181	0.9966
211	1.0011
235	0.9966
265	0.9988
301	0.9984

These results indicate that the calculational model predicts the multiplication factor for small clusters of fuel rods in a water environment to a high degree of accuracy, i.e., a bias of -.0017.

To ascertain whether the calculational mode can predict the reactivity characteristics of thick stainless steel plates and boron poisoned plates an analysis (Reference 10) was made of PNW experimental (Reference 11) critical separations of 2.35 w/o U-235 UO_2 subcritical clusters. The results using the Monte Carlo code KENO IV are shown in Table II.

Method of Calculation

The calculation methods for these experimental comparisons which are also used to determine reactivity for fuel rack storage, fuel shipping containers plus other fuel configurations found in fuel manufacturing areas are based on CEPAK 2.3 (Reference 1) cross sections. Using an appropriate buckling value and taking proper account of resonance absorption, three fast groups are collapsed from 55 fine energy mesh groups in FORM and the one thermal group is collapsed from 29 thermal energy groups in THERMOS. In addition, each component such as water gap, or poison plate has its thermal cross section determined by a slap THERMOS calculation employing the proper fuel environment. FORM and THERMOS are subprograms of CEPAK.

For one-dimensional analyses such as the BNL exponential experiments, the discrete ordinates code ANISN (Reference 12) is used. For two dimensional analyses, DOT-2W (Reference 13) is used. For three dimensional analyses (such as the critical separation experiments), KENO IV (Reference 14) is used.

<u>Results</u>

The above analyses indicate a bias between predicted mean and measured multiplication factors of +.00138 and a calculational uncertainty of .00714 at the 95/95 confidence level for the complete series of UO_2 experiments.

Thus, using CEPAK 2.3 cross sections, we conclude the following:

Total Number of Results	41
Mean Value ($\bar{\mu}$)	1.00138
Standard Deviation = 6	0.00337
σ Multiplier for 95/95 Confidence	2.11800
95/95 Confidence Level Uncertainty	0.00714
Bias ($\bar{\mu}$ - 1.0)	+.00138
Uncertainty minus Bias	.00575

It should be noted that the seven no boron steel cases have a bias of 0.00207 (i.e., the calculated value is .00207 greater than the critical K-eff value of unity) which is greater than the mean bias. The three boral cases have a bias of -0.00435 with unity particle self-shielding factor for the B_4C. Because of the size and distribution of the boron carbide particles, the boron allows more transmission than an equivalent homogeneous boron carbide mixture. Neutron transmission experiments conducted by the University of Michigan for Brooks & Perkins, Inc. (Reference 15) are consistent with using a 0.9 self-shielding factor in the third of four CEPAK neutron group and a 0.75 self-shielding factor in the thermal group. These self-shielding factors which are used in designing boron containing fuel racks make the bias for these boral cases +0.00008.

References:

1. P.H. Gavin, "CEPAK 2.3 Mod O," C-E Internal Report, 12/14/76.

2. T.C. Engelder, et al, "Spectral Shift Control Reactor, Basic Physics Program," B&W-1273, November 1963.

3. R.H. Clark, et al, "Physics Verification Program Final Report," B&W-3647-3, March 1967.

4. P.W. Davison, et al, "Yankee Critical Experiments," YAEC-94, April, 1959.

5. W.J. Eich and W.P. Rocacik, "Reactivity and Neutron Flux Studies in Multi-Region Loaded Cores," WCAP-1443, 1961.

6. F.J. Fayers, et al, "An Evaluation of Some Uncertainties in the Comparison Between Theory and Experiments for Regular Light Water Lattices, Brit. Nuc. Eng. Soc. J., 6, April 1967.

7. J.R. Brown, et al, "Kinetic and Buckling Measurements on Lattices of Slightly Enriched Uranium and UO_2 Rods in Light Water," WAPD-176, 1958.

8. J. Handschuh, L.C. Noderer, R.C. for "Compact Spent Fuel Storage Criticality Analysis for Arkansas Power and Light, Unit 2 at 68°F," C-E Internal Report, April 8, 1975.

9. G.A. Price, "Uranium - Water Lattice Compilation Part I, BNL Exponential Assemblies," BNL-50035 (T-449), December, 1966.

10. L.C. Noderer, "Analysis of Critical Separation of Low Enriched Subcritical Clusters," C-E internal Report, May 11, 1979.

11. S.R. Bierman, E.D. Clayton and B.M. Durst, "Critical Separation Between Subcritical Clusters of 2.35 w/o U-235 Enriched UO-2 Rods in Water with Fixed Neutron Poisons," PNL-2438, October, 1977.

12. Ward W. Engle, Jr., "A Users Manual for ANISN, a One Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering K-1693, March 30, 1967.

13. R.G. Soltose, R.K. Disney, A. Collier, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code," WANL-TME-1982, December, 1969.

14. L.M. Petrie and N.F. cross, "KENO IV, An Improved Monte Carlo Criticality Program" ORNL-4938, November, 1975.

15. James W. Bryson, John C. Lee and R. Robert Burn, "Neutron Transmission Through Boral Shielding Material: Theoretical Model and Experimental Comparison," University of Michigan, Dept. of Nuclear Engineering, Michigan Memorial-Phoenix Project, prepared for Brooks and Perkins, Inc., April, 1978.

TABLE I

Results of Analysis of Critical UO_2 Systems

No.	Lattice		B^2_{tot}	K_{eff}*
1	B&W (2)	I	.88-2	1.00121
2		II	.172-2	1.00534
3		X	.79-2	.99838
4		XIII	.701-2	1.00419
5		XX	.202-2	1.00550
6	B&W (3)	1	.861-2	1.00269
7		2	.420-2	1.00443
8	Yankee (4)	1	.408-2	1.00088
9		2	.531-2	1.00115
10		3	.633-2	1.00136
11	Yankee (5)	4	.688-2	1.00244
	Winfrith (6)			
12		R1-20	.660-2	1.00214
13		R1-80	.626-2	.99942
14		R3	.510-2	1.00422
15	Bettis (7)	1	.326-2	1.00053
16		2	.355-2	1.00046
17		3	.342-2	1.00106
	Average			1.00208
				±.00206

* Using calculated radial bucklings and measured axial bucklings.

TABLE II

Calculated k_{eff} Values
For Separation Experiments

Expt #	Type Poison Plate	K_{eff}	Monte Carlo σ (STD Deviation)
15	None	1.00227	.00534
04	None	0.99912	.00540
49	None	1.00221	.00473
18	None	1.00813	.00489
21	None	0.99589	.00461
28	304 S Steel 0.0 w/o Boron	1.00393	.00308
05*	304 S Steel 0.0 w/o Boron	1.00329	.00303
29	304 S Steel 0.0 w/o Boron	1.00271	.00302
27	304 S Steel 0.0 w/o Boron	1.00418	.00273
26	304 S Steel 0.0 w/o Boron	0.99811	.00279
34	304 S Steel 0.0 w/o Boron	0.99793	.00297
35	304 S Steel 0.0 w/o Boron	1.00436	.00290
32	304 S Steel 1.05 w/o Boron	0.99970	.00524
33	304 S Steel 1.05 w/o Boron	1.01173	.00491
38	304 S Steel 1.62 w/o Boron	1.00289	.00512
39	304 S Steel 1.62 w/o Boron	1.00208	.00506
20	Boral	0.99585	.00301
16	Boral	1.00020	.00288
17	Boral	0.99519	.00286

Mean K_{eff} Value	1.00157
Std. deviation	.00419