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October 17, 1985

Docket No. 50-336

B11777

Director of Nuclear Reactor Regulation
Attn: Mr. Edward J. Butcher, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Reply to Request for Additional Information on Spent Fuel Storage Capacity

In September, 1985(1) the Staff requested additional information concerning a Northeast Nuclear Energy Company (NNECO) request(2) to modify the Technical Specifications concerning the spent fuel storage capacity at Millstone Unit No. 2.

Attachment No. 1 to this letter provides the response, in a question and answer format, to the fourteen questions contained in the Staff's request for additional information.

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(1) E. J. Butcher letter to J. F. Opeka, "Request for Additional Information on Spent Fuel Storage Capacity Expansion for Millstone Unit No. 2", dated September 5, 1985.

(2) J. F. Opeka letter to E. J. Butcher, "Millstone Nuclear Power Station, Unit No. 2, Proposed Change to Technical Specification Modifications to Spent Fuel Storage Pool", dated July 24, 1985.

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1/40

We trust that the information provided is sufficient, and we remain ready to address any further questions as they arise to support expeditious processing of our pending amendment request.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Senior Vice President

E J Mroczka

By: E. J. Mroczka
Vice President

Docket No. 50-336

B11777

Attachment No. 1

Millstone Nuclear Power Station, Unit No. 2

Response to Request for Additional Information on

Spent Fuel Storage Capacity

October, 1985

Question RAB #1.1 Sources in the Spent Fuel Pool Water:

Provide a description of fission and corrosion product sources in the spent fuel pool (SFP) water from: (a) introduction of primary coolant into SFP water, (b) movement of fuel from the core into the pool, and (c) defective fuel stored in the pool. Include a listing of the radionuclides and their concentrations (expressed in uCi/mL) expected or measured during normal operations and refueling. The radionuclides of interest should include ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs.

Response:

The typically measured, post-shutdown levels of fission and corrosion products in the primary coolant during refueling outages are as follows:

Radionuclide	Concentration (uCi/ml)
Co-58	1.17 E-3
Co-60	2.033 E-4
Cr-51	1.965 E-4
Cs-134	3.815 E-5
Cs-137	5.784 E-5
I-131	6.016 E-5
Mn-54	9.77 E-6
Nb-97	1.713 E-4
Sr-92	1.337 E-5

The movement of fuel from the reactor vessel to the spent fuel pool has typically been shown to have a negligible effect on the levels listed above. Spent fuel pool samples do exhibit a peaking effect in isotopic concentrations immediately after opening the fuel transfer canal gates and allowing the free flow of reactor cavity water into the pool area. However, the radionuclide concentrations listed above are representative of the bounding values.

RAB #1.2 Airborne Radioactive Sources:

Provide a description of radioactive materials that may become airborne as a result of failed fuel and evaporation (e.g., ^{85}Kr and ^{3}H, respectively). The radionuclide description should include calculated or measured concentrations expected during normal operations and during refuelings.

Response:

The principle sources for the evaporation mechanism to generate airborne radiological hazards are the fission gases entrained in the spent fuel pool water. In this category, the isotopes Kr85 and H-3 are of the most concern but only H-3 is evident in appreciable quantities during normal operations and refuelings. The peak H-3 concentration in the spent fuel pool water during the 1985 refueling outage was measured as 3.2 E-2 micro-curies per milliliter.

The design maximum spent fuel pool water temperature under normal operation, with the increased amount of stored spent fuel, increases from 122°F to 131°F. This small increase in the water temperature will result in an increase in the evaporation rate from the pool. While this increase in the evaporation rate will result in an increase in tritium exposures to plant personnel working in the area of the spent fuel pool, the present levels of dose commitments from tritium exposure in the spent fuel building are below the levels required for tracking per 10 CFR 20. The calculated increase in tritium exposures, due to the increased fuel storage capacity, is approximately 28% and is not deemed as significantly increasing personnel dose commitments.

RAB #1.3 Miscellaneous Sources of Exposure:

Address the effects of more frequent replacement of demineralizer filters on cumulative dose equivalent if this is a factor that results from the modification.

Response:

The spent fuel pool cleanup filters are required to be changed when clogging of the filter element results in a high differential pressure across the filter. Historically, this high differential pressure occurs rarely during normal operations and three (3) to four (4) times during periods of increased activity in the spent fuel pool (e.g. a refuel outage). The proposed modification to the spent fuel pool will result in an increased inventory of spent fuel in the storage pool. However, as a general rule, the water borne contamination levels have been found to be a function of work activities in the pool and not a function of fuel inventory. Therefore, the modification of the spent fuel pool is not expected to result in a significant change to the spent fuel pool cleanup filter changeout periodicity.

Question RAB #2:

In Section 5.2.2, "Radiological Considerations", provide the basis, models, input data, and assumption for predicting the radiological impact of increasing the spent fuel pool capacity to 1112 assemblies.

Response:

The "Radiological Considerations" used in Section 5.2.2 for predicting the radiological impact of increasing the spent fuel capacity to 1112 assemblies draws its basis from previous results of dose rate predictions contained in the 1976 Millstone Unit No. 2 Spent Fuel Pool Rerack Submittal, Section 7.0.

The basis, models, input data and assumptions for originally predicting the radiological impact of increasing spent fuel pool storage capacity from 301 to 667 assemblies were as follows:

The additional spent fuel assemblies in the pool would result in an increase in dose rates in the spent fuel pool area due to a buildup of radionuclides in the pool water. To determine the amount of increase, a calculational model was devised which considered the leakage of the isotopes from the fuel to the pool water, the decontamination factor and flow rate of the spent fuel pool purification system the isotopic half-lives, and the decay time of the stored spent fuel.

Using the resulting model, the activity in the spent fuel pool was predicted for the original 301 assembly pool capacity and for the increased 667 assembly capacity. Activity due to isotopes of cobalt was assumed to be from radioactive crud in the pool water and therefore not affected by the number of assemblies stored in the pool.

From this model, the dose rate at the pool surface was found to increase from 1.55 to 1.66 millirem per hour when the pool capacity was increased from 301 assemblies to 667 assemblies. On the refueling platform, five feet above the center of the pool, the dose rate increased from 2.36 to 2.54 millirem per hour. At poolside, one foot from the pool wall and five feet above the surface, the dose rate increased from 1.43 to 1.54 millirem per hour.

These resulting small increases in dose rate were found to have a negligible effect on personnel exposure. For example, assuming an occupancy time of 400 man-hours per year at the refueling platform and 200 man-hours per year at poolside for refueling operations, the total incremental dose resulting from the original expansion of pool capacity from 301 to 667 assemblies was 0.100 man-rem per year.

Question RAB #3 Dose Rates from Fuel Assemblies, Control Rods, and Burnable Poison Rods:

a. Provide a description of the dose rate at the surface of the pool water from the fuel assemblies, control rods, burnable poison rods or any miscellaneous materials that may be stored in the pool. Additionally, provide the dose rate from individual fuel assemblies as they are being placed into the fuel racks. Information relevant to the depth of water shielding the fuel assemblies as they are being transferred into the racks should be specified. If the depth of water shielding over a fuel assembly while it is being transferred to a spent fuel rack is less than 10 feet, or the dose rate 3 feet above the spent fuel pool (SFP) water is greater than 5 mR/hr above ambient radiation levels, then submit a Technical Specification specifying the minimum depth of water shielding over the fuel assembly as it is being transferred to the fuel rack and the measures that will be taken to assure that this minimum depth will not be degraded.

Response:

a. The current inventory of spent fuel, control rods, burnable poison rods and waterborne fission and corrosion product contamination results in spent fuel pool dose rates of:

- 15 millirem per hour on pool surface at the center of the pool
- 12 millirem per hour three (3) feet above pool surface at the center of the pool
- 7 millirem per hour on the fuel handling bridge at the center of the pool

The predominant contributor to these dose rates is waterborne radioactivity.

The minimum water coverage over irradiated fuel in movement is approximately 9½ (nine and one-half) feet above the irradiated fuel stack. This was determined by actual measurement in the Spent Fuel Pool. This assumes the fuel handling tool is at its upper electric limit and the spent fuel pool water level is at minimum allowable level. Typically, irradiated fuel in movement would have greater than 9½ feet of water over the irradiated fuel stack since we do not move fuel with the fuel handling hoist at the Upper Electric Limit nor do we typically have Spent Fuel Pool Level at the minimum level.

No noticeable difference in radiation levels have been measured above the fuel pool when irradiated fuel movement is in progress with the water coverage described above.

Further, when the new spent fuel pool racks are fully installed in the fuel pool with all old racks fully removed, additional water coverage over irradiated fuel may be provided since the new racks will be at a lower elevation in the fuel pool. The old racks are supported by a steel understructure which will also be removed. The new racks are freestanding

on the pool floor liner and do not need this understructure. Up to one (1) foot of additional shielding can then be utilized by changing the fuel handling tool/sling arrangement to allow the fuel to be moved with more water coverage. After the rerack is complete, as built measurements will be taken to assess how much increased water coverage may be provided.

Based on these considerations, the need for a technical specification minimum water coverage is not considered necessary.

Question RAB #3b:

Address the dose rate changes at the sides of the pool concrete shield walls, where occupied areas are adjacent to these walls, as a result of the modification. Increasing the capacity of the pool may cause spent fuel assemblies to be relocated closer to the concrete walls of the pool, resulting in an increase of radiation levels in occupied areas. Please evaluate this potential problem.

Response:

The proposed spent fuel rack design does provide for the storage of spent fuel assemblies approximately fourteen (14) inches closer to the walls of the spent fuel pool. However, the minimum thickness of the high density concrete walls and floor of the spent fuel pool is six (6) feet. The fractional change in the total shield transmission factor when going from a six (6) foot concrete and eighteen (18) inch water composite shield to a six (6) foot concrete and three (3) and five sixteenths (5/16) inch water shield is insignificantly small.

Question RAB #4:

Provide information on the dose rates at the surface of SFP water resulting from radioactivity in the water. Include: (1) dose rate levels in occupied areas and along the edges and center of the pool and on the fuel handling crane; (2) effects of crud buildup; and (3) based on refueling water activity, the dose rates before, during, and after refueling.

Response:

The typically measured post shutdown levels of fission and corrosion products in the primary coolant during refueling outages were given in the response to Question RAB #1. As stated, samples of spent fuel pool water exhibit a peaking effect in isotopic concentrations immediately after opening the fuel transfer canal gates and allowing the free flow of reactor cavity water into the pool area. The peak tritium concentration in the spent fuel pool water during the 1985 refueling outage was measured as 3.2 E-2 uCi per ml. The measured dose rates given in the response to Question RAB #3 are predominantly a result of waterborne radioactivity.

Question RAB #5 Dose Rates from Airborne Isotopes:

Based on the source terms, provide the dose rates from submersion and dose commitments from inhalation of airborne activity for exposure to the concentrations of ^{85}Kr and ^{3}H.

Response:

As explained in the response to Question RAB #1.2, the principle isotope of concern with respect to airborne hazards is H-3.

The design maximum spent fuel pool water temperature under normal operation, with the increased amount of stored spent fuel, increases from 122°F to 131°F. This small increase in the water temperature will result in an increase in the evaporation rate from the pool. While this increase in the evaporation rate will result in an increase in tritium exposures to plant personnel working in the area of the spent fuel pool, the present levels of dose commitments from tritium exposure in the spent fuel building are below the levels required for tracking per 10 CFR 20. The calculated increase in tritium exposures, due to the increased fuel storage capacity, is approximately 28% and is not deemed as significantly increasing personnel dose commitments.

Question RAB #6 (1):

(1) Discuss the manner in which occupational exposure will be kept ALARA during the modification. Include the need for and the manner in which cleaning of the crud on SFP walls will be performed to reduce exposure rates in the SFP area.

Response:

The modification procedures for reracking the spent fuel pool will incorporate a number of ALARA controls including:

- the establishment of visible work zone barriers for diving operations in the pool at a distance no closer than the seven (7) foot line from the nearest filled spent fuel rack.

- the use of remote operations, where practical, for rack removal and replacement operations.

- the moving and storage of spent fuel assemblies between rack removal steps in such a manner as to ensure the older available spent fuel assemblies are located nearest the seven (7) foot fuel line.

- the hydrolasing and cleaning of old spent fuel racks in another location, outside the spent fuel pool, in order to keep pool water contamination levels and manhours spent in and around the pool to a minimum.

There is no appreciable level of contamination on the walls of the spent fuel pool which would present a measurable dose rate to the personnel involved in diving operations, therefore no cleaning of the pool walls is planned.

Question RAB #6 (2):

Discuss vacuum cleaning of SFP floors if divers are used and the distribution of existing spent fuel stored in racks to allow maximum water shielding to reduce dose rates to divers.

Response:

The floor of the spent fuel pool will be vacuumed as old racks and components are removed, thereby allowing access to these areas.

The diver work area will be established such that there is a seven (7) foot wide buffer zone between the diver and the nearest filled spent fuel rack. Additionally, the older available spent fuel assemblies will be placed in the storage locations nearest the seven (7) foot buffer zone.

Question RAB #6 (3):

Describe plans for cleanup of the SFP water to minimize radioactive contamination and to ensure fuel pool clarity and underwater lighting acceptance criteria to help ensure good visibility.

Response:

The fuel pool cooling and cleanup filters will be maintained in an operating condition throughout the reracking process. The filters are changed at a pre-established high differential pressure reading, and are expected to reach this high differential pressure at a frequency somewhat greater than the previously experienced three (3) or four (4) times during normal refueling outage periods of heavy pool work activity.

In addition to the use of the normal fuel pool cleanup systems, water clarity will be aided by local, high volume water vacuuming during evolutions which would generate increased levels of waterborne contaminants.

Existing station procedures require that the diver and work area must be clearly visible to Health Physics personnel providing the controls.

Question RAB #6 (4):

Discuss underwater radiation surveys that will be made before any diving operation. These surveys should be performed during or after any fuel movement or movements of any irradiated components stored in the pool.

Response:

The existing Health Physics Procedure specifying requirements for underwater spent fuel pool work requires that radiation surveys be performed prior to any diving operations on a daily basis. These surveys are performed using two different survey instruments and a complete survey of the work area is required after any movement of fuel or any irradiated component. Results of the survey are documented on detailed survey maps and clear work area boundaries, entry and exit routes are designated for the diver to minimize exposure.

Additionally, the procedure requires that an operable survey instrument monitoring dose rates be maintained in close proximity to the diver and that the entire work area be surveyed twice per shift.

Question RAB #6 (5):

State your intent to equip each diver with a calibrated alarming dosimeter and personnel monitoring dosimeters, which should be checked periodically to ensure that prescribed dose limits are not being exceeded.

Response:

The existing Health Physics Procedure specifying requirements for underwater spent fuel pool work requires that each diver be prepared by establishing at least two waterproof packets containing a TLD, a High Range Dosimeter, and a Low Range Dosimeter. One monitoring packet is to be placed on the part of the body expected to receive the highest radiation dose. The other monitoring packet is to be placed on the upper chest. Each diver will wear extremity rings and/or additional TLD's as required and an audible alarm dosimeter with fresh batteries will be placed inside each diving helmet.

Prior to a dive, a maximum allowable stay time, based on radiation levels, before the diver must surface to have dosimeters read will be established by Health Physics personnel and explained to each diver. During a dive each diver will be equipped with a safety line and continuous voice communication with surface personnel.

Question RAB #6 (6):

Discuss any preplanning of work by divers as required.

Response:

The existing Health Physics Procedure specifying requirements for underwater spent fuel pool work requires that Health Physics Personnel designate clear work area boundaries, entry and exit routes, and discuss the job, radiation levels and pool conditions with each diver prior to a dive. Discussion of the job with the divers will entail use of a spent fuel pool model developed by NU specifically as a training aid. In addition, full size and half size "mock-up" models will be utilized in conjunction with the engineering drawings to identify the scope of the individual activities associated with each particular dive. Additionally, an allowable stay time is determined by Health Physics personnel and explained to each diver before a dive.

Question RAB #6 (7):

Discuss your provision for surveillance and monitoring of the spent fuel pool work area by Health Physics personnel during the modification.

Response:

The existing Health Physics Procedure specifying requirements for underwater spent fuel pool work requires that Health Physics personnel provide continuous Health Physics coverage and an approved Radiation Work Permit is required for all diving. Additionally, with a diver in the water, an operable survey instrument monitoring dose rates in close proximity to the diver is required as is a periodic (i.e. twice per shift) survey of the entire work area. The procedure requires that whenever there is a change in on-the-job Health Physics coverage, the technician going off duty must thoroughly brief the person coming on duty about the status of the work and the radiologicai conditions. The procedure also requires that each diver be equipped with a safety line and continuous voice communication with surface personnel and that all diver support personnel understand required diving vendor emergency diver rescue procedures.

Question CPB #1:

You have not described or provided detailed results of the benchmark calculations for your methods which provide values of uncertainties and biases when calculating (1) fuel parameters including burnup, (2) your fuel pool configurations, (3) boron as used in Region I, (4) burnup effects as in Region II. Please describe these analyses and their applicability to your configurations and parameters, and present relevant results and conclusions on uncertainties. In particular, discuss the effects of axially non uniform burnups and reactivity parameters and their uncertainty (including extreme ranges which might be encountered). If some of this information has been presented (and approved) in other applications these may be referenced, but include a summary, basis for applicability, and results for your configurations.

Response:

Appendix A attached is a copy of the Combustion Engineering validation report showing how the bias and uncertainties are determined. The critical experiments cover a wide range of parameters, including experiments with boron between fuel clusters as in Region I of the Millstone Unit No. 2 racks. The critical experiments also include steel between fuel clusters as in Region II.

The CEPAK lattice program is used to calculate the fission product and transuranium isotopic content in burned fuel. The attached figure (Figure 1.1) shows good agreement between the predicted and measured plutonium isotopic concentrations for a Yankee-Rowe asymptotic fuel pin.

Fission products are calculated by the CINDER module in CEPAK. Fourteen (14) fission product chains representing the more important fission products are used instead of the sixty-nine (69) chains employed in WAPD-TM-334*; the difference in fission product poisoning between the condensed and detailed fission product chain is treated as a lumped fission product for each fissioning nuclide.

The possible reactivity changes in the fuel subsequent to removal of the fuel from the reactor for storage in the spent fuel racks are also calculated. Fuel nuclide concentrations appropriate to the end of the 3000 hour cooling period are used to calculate the microscopic cross sections for use in the DOT analyses of the effective multiplication of Region II.

The maximum reactivity change due to the axial burnup shape was determined by analyzing the two extreme axial shapes that have been observed in reactor core analysis (cosine axial shape, and a flat distribution). The two extreme distributions were picked from axial shapes taken from reactor core analysis; as expected, the cosine-like shape occurred at the end of Cycle 1 as shown in Figure 1. The flattest distribution occurred at the end of what is considered an equilibrium cycle (Figure 2). The axial average for both of these burnup shapes

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*T. R. England, "CINDER - A Point Depletion and Fission Product Program," WAPD-TM-334, Revised June 1964.

Question CPB #1 (Cont.)

was calculated. The average burnup became the uniform burnup case, which did not take into consideration the axial shape. The reactivity was calculated for both the uniform burnup and the actual burnup profiles (Figure 1 and 2). The effective multiplication factors were calculated using KENO assuming the entire Region II rack was full of fuel with the same burnup shape. The effective multiplication factors obtained using the KENO code are as follows:

a) Assembly discharged at EOC1

Uniform Burnup Distribution	Non-Uniform Burnup Distribution
0.97575 ± 0.00306	0.96584 ± 0.00353

b) Assembly discharged at EOC6 (Equilibrium)

Uniform Burnup Distribution	Non-Uniform Burnup Distribution
0.99720 ± 0.00145	1.00834 ± 0.00164

The analysis shows that the assumption of uniform burnup axially may result in underestimating the multiplication factor in some cases, and overestimating it in other cases. For the Millstone Unit No. 2 Region II spent fuel racks, the resulting reactivity difference for the EOC6 was 0.0114 delta K effective, which was conservatively determined to be equivalent to 1800 MWd/t. The Region II allowable (average) burnup for each initial enrichment reflects this reactivity penalty.

APPENDIX A

QUALIFICATION OF ANALYTICAL METHODS USED IN SPENT FUEL STORAGE RACK ANALYSES

I. Purpose

The purpose of this memo 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 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	0.9968
301	0.9964

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 slab THERMOS calculation employing the proper fuel environment. FORM and THERMOS are sub-programs 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.

Results

The above analyses indicate a ~~mean~~ error between predicted and measured multiplication factors of -.00135 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 ($\overline{\mu}$)	1.00138
Standard Deviation = σ	0.00337
Multiplier for 95/95 confidence	2.118
95/95 Confidence Level Uncertainty	0.00714
Bias ($\overline{\mu}$ -1.0)	+.00138
Uncertainty Minus Bias	.00575

It will 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:

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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-2433, 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. Sottesy, 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 & Perkins, Inc. April 1978.

LCN:njm

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	1.00410
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	1.00098
9		2	.531-2	1.00115
10		3	.633-2	1.00130
11	Yankee (5)	4	.688-2	1.00244
	Winfrith (6)			
12		R1-20	.660-2	1.00214
13		R1-80	.626-2	.99942
14		E3	.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
25	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	[illegible]	.00524
33	304 S Steel 1.05 w/o Boron	[illegible]	.00491
38	304 S Steel 1.62 w/o Boron	1.00239	.00512
39	304 S Steel 1.62 w/o Boron	1.00208	.00506
20	Boral	0.99555	.00301
16	Boral	1.00020	.00288
17	Boral	0.99519	.00286

Mean K_{eff} Value 1.00157

Std. deviation .00419

CYCLE 1

AXIAL POWER DISTRIBUTION

BOX 62

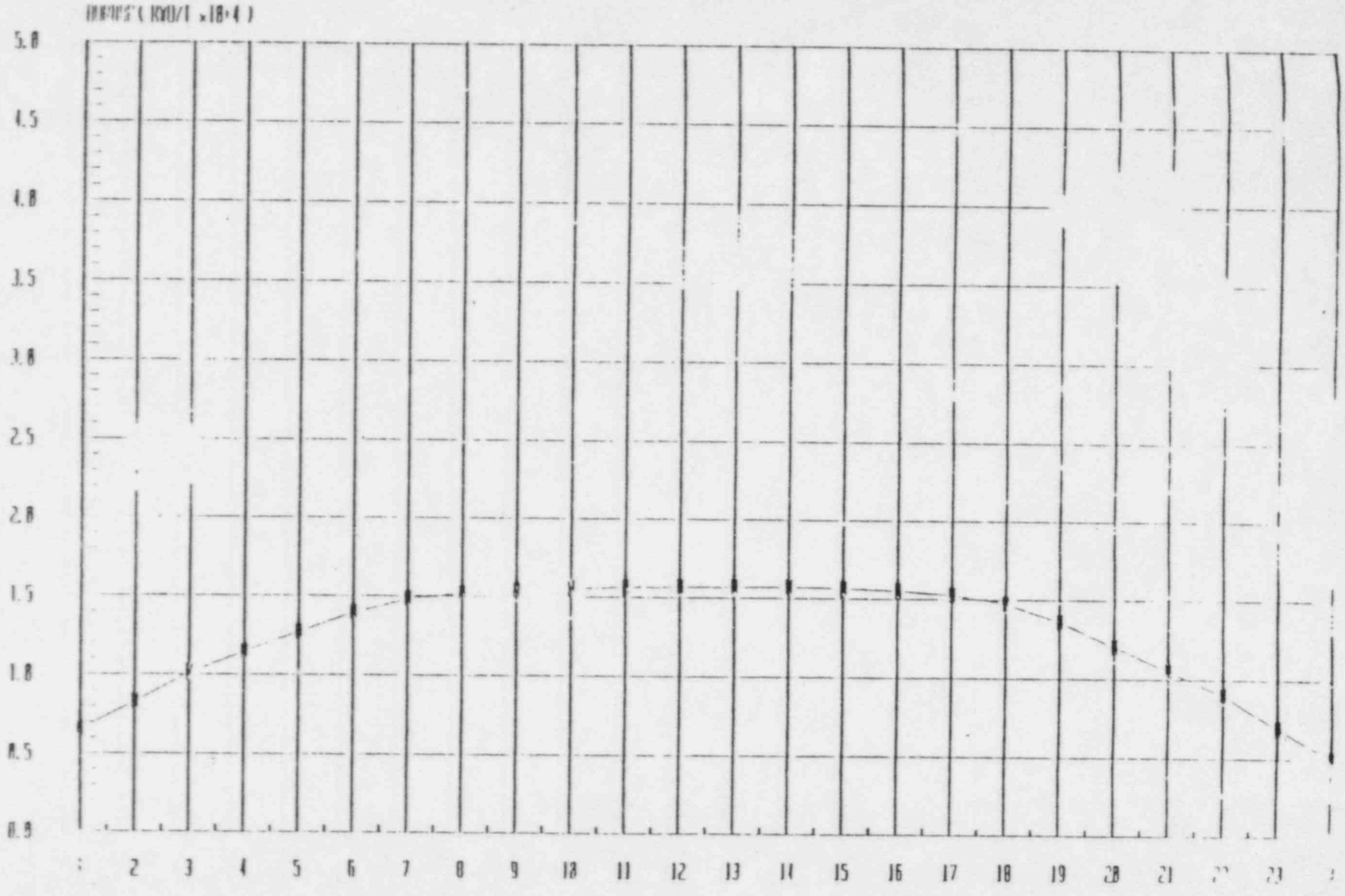
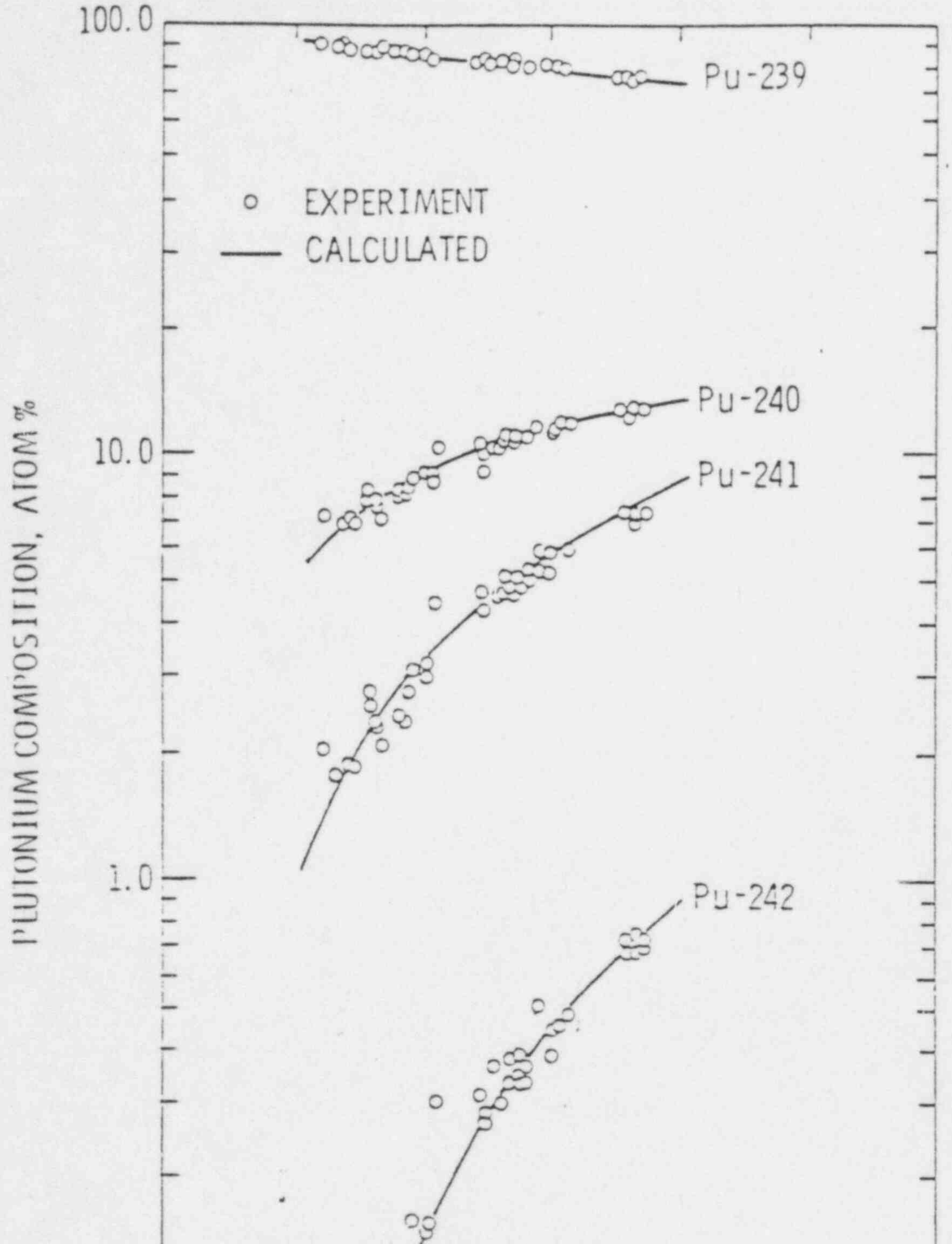
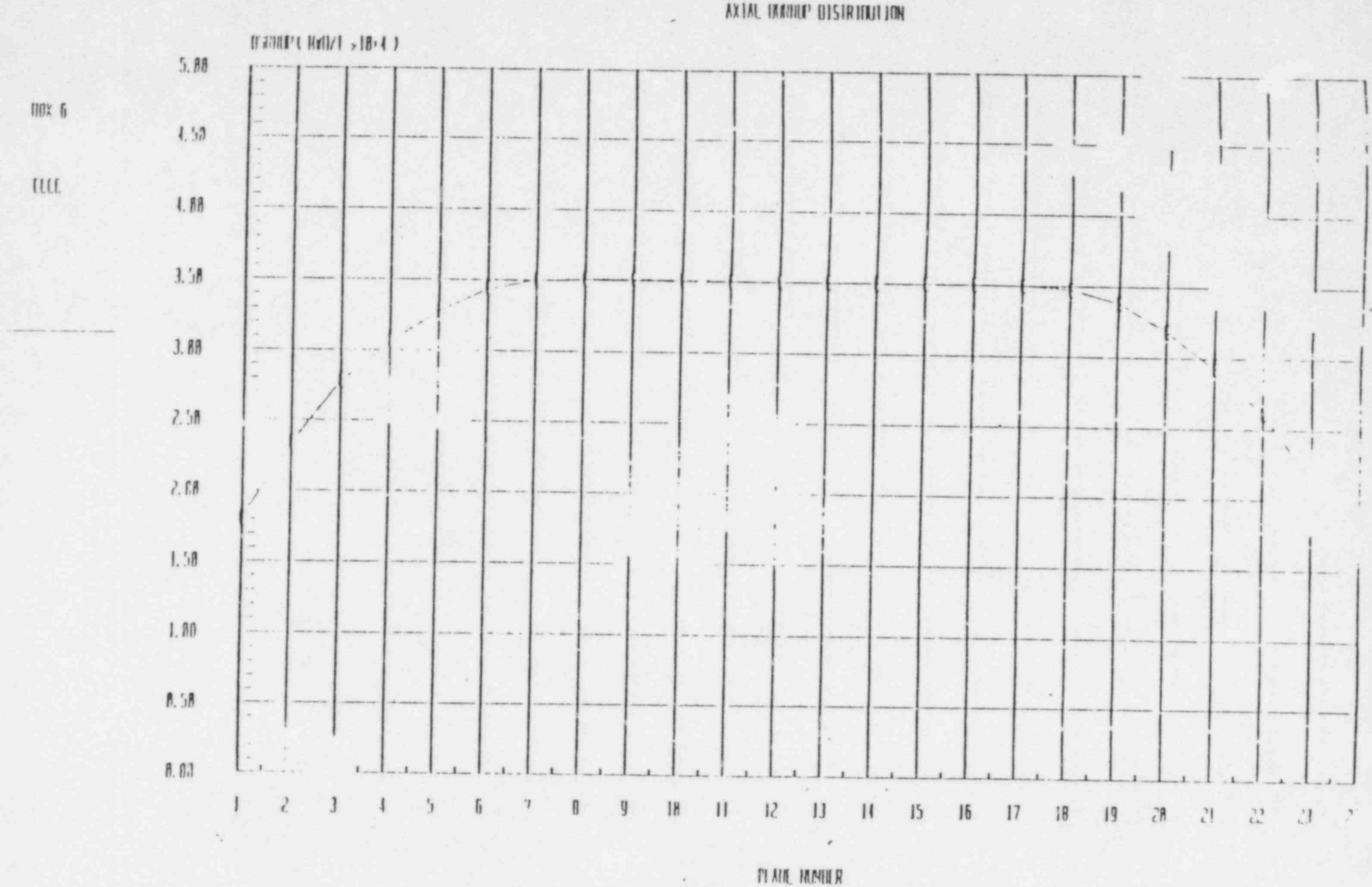


Figure 1



PLUTONIUM ISOTOPIC COMPOSITION VS FUEL DEPLETION IN THE ASYMPTOTIC SPECTRUM FOR YANKEE-ROWE

FIGURE 1.1


AXIAL BURNUP DISTRIBUTION
5.00
4.50
4.00
3.50
3.00
2.50
2.00
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23
PLANE NUMBER

Figure 2

Question CPB #2:

Your list of reactivity uncertainties contains no values for variation of B10 (via boroflex dimensions and content) and fuel parameters (enrichment, density, etc.). Please explain.

Response: 2.

The B-10 areal density is a minimum, no credit was taken for the poison effect of non-B10 material, and nominal dimensions were used. The Boroflex supplier's Certificate of Compliance is attached.

The fuel enrichment is the maximum allowed in storage. The fuel density used in the analysis is a nominal fresh fuel stacking density - this is conservative relative to exposed fuel densities. Nominal values of other fuel assembly parameters were used since their variation was considered statistically non-significant for this analysis.

bisco

November 27, 1984

Composite Certificate of Compliance

bisco products, inc.
1125 howard st.
elk grove village, illinois 60007
(312) 640-1840

C.E. Power Systems Group
Combustion Engineering, Inc.
Prospect Hill Road
Windsor, CT 06095

Attention: W.J. Doyle (Code 5205-EG04)

Dear Mr. Doyle:

Bisco Products, Inc. certifies that the material described herein meets the requirements of your Purchase Order 9471856-11784-1 for 1,550 Sheets of Boraflex™ Neutron Absorbing Material producted to the following limits:

> 0.110" ± 0.007" thick x 141" + 1/4" - 1/8" long x 8 1/8" ± 1/16" wide. 0.030 g/cm^2 minimum B_{10} loading.

In addition to the test reports included, Bisco Products, Inc. submits the following to comply with the contractual requirements of your purchase order.

Appendix A	Listing of all specification numbers, and revision numbers applicable to this material. Originally submitted on Bisco Document Transmittal M-038, dated 7/17/84.
Appendix B	Summary of all TCRS & RARS submitted to C.E. Power Systems Group.
Supplement A	Bisco Technical Report N-4

This material has been shipped to you via Bisco Delivery Ticket No. 8918, Job No. 00459/573.

Respectfully,

BISCO PRODUCTS, INC.

J.D. Liwo
Director of Quality

Enclosures (3)
Q-84-68
cc: J. Anderson
K. Hoffman
J. Woolley
D. Kroman
C. Klaver

Question CPB #3:

Is there any contained burnable poison in analyzed fuel?

Response:

Some fuel assemblies, from the original core, now stored in the Millstone Unit No. 2 Spent Fuel Pool contain burnable poisons. These burnable poison pins have essentially no B-10 since these poison pins were depleted during the initial fuel cycle.

Analyses supporting the criticality calculations for the spent fuel racks treat these fuel assemblies in a highly conservative manner by assuming that fuel pins are present rather than poison pins. Thus, no credit has been assumed in the design and analysis of the spent fuel racks for any burnable poisons that might be present in the fuel inventory.

Question CPB #4:

Please discuss the pool temperature range studied for criticality, the temperature values used for conclusions on meeting pool criticality limits, and the demonstration that these constitute a maximum reactivity condition.

Response:

For Region I, lower pool temperatures constitute a maximum reactivity condition. For Region II, higher pool temperatures constitute a maximum reactivity condition.

The design temperature used for Region I was 90°F while 120°F was used for Region II. The difference between 90°F and 68°F (for Region I), and the difference between 120°F and 236°F (for Region II) was assigned in the uncertainty analysis. Calculations performed relative to Question CPB #4 resulted in a revision to Section 3.1 of our submittal. The Region I criticality calculations were revised to temperatures below 90°F, to a minimum temperature of 68°F, and resulted in a delta K_{eff} Temperature Change uncertainty value of +0.0031 rather than the previously indicated +0.0042.

Thus, for Region I the resulting neutron multiplication factor (K_{eff}) is 0.943 including all uncertainties and calculational biases. This is in contrast to the value of 0.939 which appears in Section 3.1.4 on page 8 of the original submittal.

Question CPB #5:

You imply credit for the double contingency principle via pool boron for accidents in Region I as well as in Region II (although Region II is limiting) but you have no Technical Specifications for boron in Region I for assembly movement as you do in Region II. Please discuss and provide a Region I Specification unless otherwise justified.

Response:

The spent fuel rack design for Region I is based upon the physics principle of a "neutron flux trap". This flux trap is provided by the use of a neutron absorber or poison material (Boroflex). The Region I racks are designed to store Millstone 14x14 fuel with an initial enrichment of 4.5 weight percent U-235. There are, therefore, no limitations on storage of fuel in these poison racks.

The spent fuel rack design for Region II is based on the criticality acceptance criteria specified in Revision 2 of Regulatory Guide 1.13 which allows credit for reactivity depletion in spent fuel. Region II consists of fourteen (14) modules of non-poisoned spent fuel racks. Fuel assemblies are stored in a three-out-of-four logic pattern. The fourth location of the storage configuration remains empty to provide the flux trap to maintain the required reactivity control. Blocking devices are used to prevent inadvertent placing of a fuel assembly in the fourth location.

The Technical Specification on pool boron is provided to ensure that, in the event of a fuel assembly drop accident or an inadvertent placement of a Region I fuel assembly into a Region II rack location, the K_{eff} will remain less than or equal to 0.95. Because of the poison rack design used in Region I, no such Technical Specification is required for Region I.

Question CPB #6:

It has been our policy in past reviews of spent fuel pools when credit for burnup and multiple regions is involved that reactor unloading involve moving the fuel to be stored first to the fresh fuel region (in your case Region I). Then when the reload is complete, a careful analysis and check of the burnup records is made. Only then is fuel moved into the burnup credit regions (Region 2). Please indicate if this procedure can and will be followed in your operations. Technical Specifications should reflect the transfer and record keeping process. Please provide a discussion of the procedures, limitations, record keeping and accountability measures taken throughout the life of the facility that support the assumption that the inadvertent misplacement of fuel assemblies is not expected to occur.

Response:

Procedures are currently being written to address the questions and concerns raised. The outline of the procedure to be followed for placing a fuel assembly into a Region II rack is shown in Table I. This Table describes the thought process and procedural requirements to be performed prior to allowing a fuel assembly to be placed into a Region II rack location.

The two (2) principal problems to be focused on when using a 2 Region Rack Concept are:

(1) Ensuring that accurate burnups are determined for each fuel assembly to be located in Region II and

(2) Ensuring that the correct fuel assembly is placed into Region II, once a decision has been made that a fuel assembly is qualified for Region II storage.

The procedural requirement that a TV camera or binocular inspection of the serial number engraved on each fuel assembly be performed <u>after</u> grappling the fuel but <u>prior</u> to inserting the fuel into the Region II rack should avoid the possibility of picking up the wrong fuel assembly.

Ensuring that correct burnups are used in the evaluation of whether a fuel assembly is qualified for Region II will be done by:

(1) Using Measured Assembly Burnups

(2) Comparing Measured Burnups to Predicted Burnups for a consistency check

(3) 2 reactor engineering personnel will independently calculate all numbers used in evaluating a fuel assembly for use in Region II

(4) Reactor Engineer approval is required

(5) Existing Special Nuclear Material Procedures and Material Transfer Forms will be utilized

Question CPB #6 (Cont.):

The Engineering Forms described in this Table will be retained for the life of the plant.

Procedures being written will state that the preferred method of fuel storage for fuel coming out of the reactor vessel is to place the fuel in the Region I racks and later consider transfer to the Region II racks for qualified assemblies. However, we can conceive of situations where movement of qualified fuel to Region II may be required directly from the core. To preserve this option, our procedures will allow, at the Reactor Engineer's discretion, such moves from the Reactor Vessel to the Region II racks of qualified fuel assemblies if no alternative exists. Should this be necessary, all procedural requirements previously described would still be applicable. Also, pursuant to proposed Technical Specification 3.9.18, surveillance will be performed prior to fuel movement, to ensure 800 ppm boron is available in the Spent Fuel Pool as a precondition to a postulated fuel movement accident.

TABLE I

Outline of Procedure for Allowing Fuel into Region II

	Step	Comments
(1)	Determine Fuel Assembly Initial Enrichment & Measured accumulated average burnup. (Performed by site Reactor Engineering.)	Initial Enrichments from DOE/NRC-741 Forms. Measured Burnups from Incore Monitoring System.
(2)	Record Data on Engineering Form for that Fuel Assembly.	Each Fuel Assembly has its own form as a permanent record of its status.
(3)	Calculate allowable burnup per Proposed Technical Specification Figure 3.9-1 and record on Engineering Form.	
(4)	Determine, as an independent check, that measured Fuel Assembly burnup is close to predicted Fuel Assembly Burnup.	Predicted Fuel Assembly Burnups are available from the vendor.
(5)	Determine if Fuel Assembly meets Region II reactivity requirements. That is, measured burnup from Step (1) is greater than allowable burnup from Step (3).	

TABLE I (Continued)

Step	Comments
(6) Perform an independent check of Steps 1 thru 5. (To be performed by different person from Reactor Engineering than did steps 1 thru 5.)	
(7) Reactor Engineer approves/disapproves fuel assembly for use in Region II based on steps 1 thru 6.	
(8) If assembly is allowed to be put into Region II, it will be added to Engineering Form which lists all fuel allowed in Region II.	
(9) Material Transfer Form written which allows/directs fuel to be moved to Region II per current Special Nuclear Material Procedures.	Fuel Assembly not allowed in Region II unless it is on the engineering form described in Step (8).
(10) Fuel Assembly to be moved into a Region II rack must have its serial number verified using TV camera or binoculars prior to insertion into a Region II rack location.	This verification to be performed after grappling fuel assembly but before insertion into the Region II racks.

Question CPB #7:

There is apparently a change, related to your Technical Specification 5.6.1.a, for new fuel (dry) storage, (i.e., a limit on fuel enrichment). However, there is no discussion of this area in your submittal. Please provide the analysis and justification for the limit. Please note that for dry storage both full flooding, with 0.95 criteria (including uncertainties), and maximized low moderator density (e.g., from fire fighting), with a 0.98 multiplication criteria (with uncertainty), need to be considered unless approved provisions for preventing such conditions exist.

Response:

There is no change in the design of the new fuel (dry) storage racks. Section 9.8 of the Millstone Unit No. 2 FSAR Page 9.8-6 states the original design of the new fuel (dry) storage racks to allow up to 3.7 w/o (U-235) enrichment.

The existing Spent Fuel Pool racks allow up to a 3.35 w/o U-235 enrichment. Our existing technical specifications (Section 5.6) state that 3.35 w/o U-235 is the maximum fuel enrichment to be stored in the Spent Fuel Pool racks. Our current technical specifications do not specify a maximum enrichment for the new fuel (dry) storage racks since the existing spent fuel pool racks are limiting, (i.e. 3.35 w/o maximum enrichment vs 3.7 w/o maximum enrichment).

With the proposed new design of the spent fuel pool racks, the new fuel (dry) storage racks would become limiting for the maximum allowed fuel enrichment. The proposed new design of the spent fuel racks allow up to a 4.5 w/o enrichment. Since this is greater than the 3.7 w/o enrichment design limit on the new fuel (dry) storage racks, we specifically state in the proposed technical specifications that 3.7 w/o U-235 is the maximum enrichment allowed in the new fuel (dry) storage racks.

In summary, there is no change in design, layout or use of the new fuel (dry) storage racks. The technical specifications have been proposed to be modified to specifically state the current existing design limits of our new fuel (dry) storage racks.

Question AEB #1:

In accordance with guidelines of SRP 15.7.4, evaluate the radiological consequence of a cask handling accident that damages stored fuel. Include in the evaluation the assumption used and the basis for each assumption.

Response:

Attached is a copy of Calculation Number 81-166-86RA "Millstone Unit 2 Analysis of Cask Drop in the Spent Fuel Pool (Reracked)".

Millstone Unit 2 Analysis of Cask Drop in the Spent Fuel Pool (Reracked)

QA Category I
Calc No: 81-166-86RA
1∅ Pages

Rev. ∅

Review Method:
Full Review in accordance with NEO 5.06 Rev. 0

Prepared:
David W Margill 6-25-85

Reviewed:
(Full Review)
Laraay Jean Rayburn 7-10-85

Approved:
John W. Dorvalis 7/18/85

Page*

TABLE OF CONTENTS

LIST OF TABLES

DWM 6-25-85

81-166-86RA
2 10

PURPOSE:

This document is to demonstrate that a spent fuel cask drop in the Millstone Unit 2 spent fuel pool will at worst result in a release "well within" 10 CFR Part 100 guidelines. This is shown to be true with the existing racks or with the racks associated with Proposed Technical Specification Change Request #2-5-85. (ref 1).

DISCUSSION:

Millstone Unit 2 PTSCR No 2-5-85 proposes changes to the Tech Specs as a result of new fuel storage racks. These racks are designed to increase the storage capacity of the spent fuel pool from 667 to 1112

DWM 6-25-85

81-166-86RA
3 10

locations. The Nuclear Review Board has requested a radiological evaluation of the new racks under spent fuel cask drop conditions. The old racks will be evaluated for ~~pup~~ purposes of calculating the increase/decrease in dose, per 10 CFR 50.~~59~~ DWM and 50.~~92~~ DWM

There are administrative limits as to where an assembly may be placed in the spent fuel pool as a function of the amount of time it has been subcritical. These limits are intended to remain in effect with the new racks. The limit is defined in ~~the~~ Millstone 2 Safety Technical Specification 3.9.16 (ref. 2):

"All fuel within a distance L from the center of the spent

DWM 6-25-85

81-166-86RA
4 10

fuel pool cask ~~set down~~ set-down area shall have decayed for at least 120 days. The distance L equals the major dimension of the shielded cask."

It has been shown in reference 3 that, for either set of racks, damage cannot occur to fuel assemblies at a distance further than the distance L. Thus, only fuel which has decayed at least 120 days may be damaged during the postulated accident, for either set of racks.

Guidance for this analysis appears in NUREG 0612 (ref 4). Figure 2.1-1 of this reference shows that for decay times of 120 days or more, the .

limiting dose is the whole body dose.

ANALYSIS:

The fundamentals of this calculation have been done in NUREG 0612 (ref 4). This analysis will demonstrate that the assumptions used in NUREG 0612 are satisfied by Millstone 2. Ratios are applied to calculate conservative dose estimates.

Table I compares the assumptions used in NUREG 0612 (Table 2.1-2) with the values used for Millstone 2. Note that no credit is taken for filtration.

The results in NUREG 0612 can now be applied to Millstone 2. From NUREG 0612, Table 2.1-1, it can be seen that the minimum number of 120 day

DWM 6-25-85

81-166-86RA
6 10

TABLE I

COMPARISON OF CASK DROP ASSUMPTIONS

	NUREG 0612	Millstone 2
Power Level (MW_t)	3000	2700
0-2 Hour χ/Q, EAB (sec/m^3)	$1.0 * 10^{-3}$	$5.4 * 10^{-4}$ (ref 6)
0-2 Hour χ/Q, LPZ (sec/m^3)	$1.0 * 10^{-4}$	(Not Calculated)
Peaking Factor	1.2	1.2
No. of Assy in Core	193	217
Pool Water Decon for Iodine	100	100

subcritical PWR assemblies that need to be damaged for the postulated cask drop to approach one-quarter of 10CFR100 dose limits is 7500. To apply this to Millstone 2, this number is ~~ratioed~~ corrected for power level and χ/Q:

$$7500 \text{ assy} * \frac{3000 \text{ MW}_t}{2700 \text{ MW}_t} * \frac{1.0 * 10^{-3} \text{ sec/m}^3}{5.4 * 10^{-4} \text{ sec/m}^3} = 15,400.$$

When dealing with this many assemblies each can be construed as having an equal contribution to the dose, and since the total number of 15,400 assemblies approaches ~~the~~ 1/4 of 10 CFR 100 limits (as noted above, this is 6.25 rem whole body), each assembly can be conservatively calculated to contribute ~~0.85~~ 0.41 (DM) mrem.

It has been conservatively calculated

DWM 6-25-85

81-166-86RA
8 10

(ref 5) that the number of locations in the Millstone 2 spent fuel pool that are within the distance L is 352 locations for the current racks and 587 locations in the proposed design. By assuming that all these locations have assemblies which are damaged during a cask drop (very conservative), dose estimates can be computed. These are given in Table II.

<u>SUMMARY</u>:

The conservatively calculated dose at the site boundary due to the postulated cask drop is ~~487~~ 241 DM mrem (whole body) with new racks in the spent fuel pool. This is "well within" 10CFR100 guidelines. It represents an increase of ~~195~~ 97 DM mrem over the (conservatively) estimated dose using the current racks.

DWM 6-25-85

81-166-86RA
9 10

TABLE II

CONSERVATIVE OFFSITE DOSE ESTIMATES FOLLOWING A CASK DROP

	No. of Assy Damaged	Whole Body Dose (mrem)
Current rack design	352	144
Proposed rack design*	587	241
Increase	235	97

*This is based 3 of 4 cell occupancy ($782 \times \frac{3}{4} = 587$) of the racks, as noted in refs (1) and (5).

Dean 6-25-85

81-166-86RA
10 10

REFERENCES

1) "Millstone Unit 2 Proposed Technical Specification Change Request # 2-5-85: Spent Fuel Pool Rerack". Also, the associated Safety Evaluations:
   a) Mechanical, Material, Seismic, Structural, signed GN Betancourt on 6-3-85
   b) Structural, signed WJ Briggs 5-17-85
   c) Technical Review, SJ Weyland, 5-7-85
   d) Safety Evaluation - Nuclear C.S. Banworth, 5-17-85

2) Millstone Unit 2 Safety Technical Specification No. 3-9.16, dated 6-30-77

3) Memo, R.N. Smart to D.W. Marzilli, "Cask Drop MP2 Spent Fuel Pool Rerack", July 10, 1985. GCE-85-435

4) NUREG 0612, Control of Heavy Loads at Nuclear Power Plants, USNRC July 1980

5) Memo, GN Betancourt to RA Crandall, "Millstone Unit 2 TSCR 2-5-85 Spent Fuel Pool Rerack Cask Drop Accident", June 20, 1985. PSE-85-537

6) QA Cat I Calc # xxxx-5-PS, "χ/Q's at Site Bndry for Release from MP2 Aux Bldg.", Jan 19, 1978. No Revision # attached.